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

August 30, 2016

Mr. Anthony J. Vitale
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 – INTEGRATED INSPECTION REPORT 05000247/2016002 AND 05000286/2016002

Dear Mr. Vitale:

On June 30, 2016, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Indian Point Nuclear Generating (Indian Point), Units 2 and 3. The enclosed inspection report documents the inspection results, which were discussed on August 4, 2016, with Larry Coyle and other members of your staff. Based on additional information provided, the inspectors conducted an updated exit meeting on August 30, 2016 with John Kirkpatrick, Plant Operations General Manager and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents three NRC-identified findings of very low safety significance (Green). These findings involved violations of NRC requirements. However, because of the very low safety significance, and because they are entered into your corrective action program, the NRC is treating these findings as non-cited violations, consistent with Section 2.3.2.a of the NRC Enforcement Policy. If you contest any non-cited violation 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 Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Senior Resident Inspector at Indian Point. In addition, if you disagree with the cross-cutting aspect assigned to any 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 Senior Resident Inspector at Indian Point.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390 of the NRCs "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's Public Document Room or from the Publicly Available Records component of the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Glenn T. Dentel, Chief
Reactor Projects Branch 2
Division of Reactor Projects

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

Enclosure:
Inspection Report 05000247/2016002 and 05000286/2016002
w/Attachment: Supplementary Information

cc w/encl: Distribution via ListServ

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390 of the NRCs "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's Public Document Room or from the Publicly Available Records component of the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Glenn T. Dentel, Chief
Reactor Projects Branch 2
Division of Reactor Projects

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

Enclosure:
Inspection Report 05000247/2016002 and 05000286/2016002
w/Attachment: Supplementary Information

cc w/encl: Distribution via ListServ

DISTRIBUTION: (via e-mail)
DDorman, RA
DLew, DRA
MScott, DRP
DPelton, DRP
RLorson, DRS
PKrohn, DRS
GDentel, DRP
NFloyd, DRP
JSchussler, DRP
KReid, DRP
BHaagensen, DRP, SRI
GNewman, DRP, RI
SRich, DRP, RI
DHochmuth, DRP, AA
JBowen, RI OEDO
RidsNrrPMIndianPoint Resource
RidsNrrDorlLpl1-1 Resource
ROPReports Resources

DOCUMENT NAME: G:\DRP\BRANCH2\A - INDIAN POINT\IP2&3 INSPECTION REPORTS\2016\2016-002\IP2&3 2016002 FINAL 8 30 2016.DOCX
ADAMS Accession No. ML16243A245

☑ SUNSI Review		☑ Non-Sensitive ☐ Sensitive		☑ Publicly Available ☐ Non-Publicly Available	
OFFICE	RI/DRP	RI/DRP	RI/DRS	RI/DRP	RI/DRP
NAME	BHaagensen/bh	NFloyd/nf	MGray/mg	GDentel/gtd	MScott/dlp for
DATE	8/29/16	8/24/16	8/30/16	8/30/16	8/30/16

OFFICIAL RECORD COPY

U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos.	50-247 and 50-286
License Nos.	DPR-26 and DPR-64
Report Nos.	05000247/2016002 and 05000286/2016002
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	Indian Point Nuclear Generating Units 2 and 3
Location:	450 Broadway, GSB Buchanan, NY 10511-0249
Dates:	April 1, 2016, through June 30, 2016
Inspectors:	B. Haagensen, Senior Resident Inspector G. Newman, Resident Inspector S. Rich, Resident Inspector S. Galbreath, Reactor Inspector J. Furia, Senior Health Physicist N. Floyd, Senior Project Engineer K. Mangan, Senior Reactor Inspector J. Poehler, Senior Materials Engineer
Approved By:	Glenn T. Dentel, Chief Reactor Projects Branch 2 Division of Reactor Projects

TABLE OF CONTENTS

SUMMARY

Inspection Report 05000247/2016002 and 05000286/2016002; 04/01/2016 – 06/30/2016; Indian Point Nuclear Generating (Indian Point), Units 2 and 3; Operability Determinations and Functionality Assessments, Refueling and Other Outage Activities, and Follow Up of Events and Notices of Enforcement Discretion.

This report covered a three-month period of inspection by resident inspectors and announced inspections performed by regional inspectors. The inspectors identified three findings of very low safety significance (Green), which were non-cited violations (NCVs). The significance of most findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, “Significance Determination Process,” dated April 29, 2015. Cross-cutting aspects are determined using IMC 0310, “Aspects within the Cross-Cutting Areas,” dated December 4, 2014. All violations of U.S. Nuclear Regulatory Commission (NRC) requirements are dispositioned in accordance with the NRC’s Enforcement Policy, dated February 4, 2015. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 6.

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Green NCV of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because Entergy did not adequately accomplish the actions prescribed by procedure EN-OP-104, “Operability Determination Process,” for a degraded condition associated with the Unit 3 baffle-former bolts. Specifically, Entergy incorrectly concluded that no degraded or non-conforming condition existed related to the Unit 3 baffle-former bolts and exited the operability determination procedure. Entergy subsequently performed the remaining steps in the procedure and provided appropriate justification for their plans to examine the baffle-former bolts at the next Unit 3 refueling outage (RFO). Entergy’s immediate corrective actions included entering the issue into its corrective action program (CAP) as CR-IP3-2016-01961 and documenting an operability evaluation to support the basis for operability of the baffle-former bolts and the emergency core cooling system (ECCS).

  This performance deficiency is more than minor because it was associated with the equipment performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). In accordance with IMC 0609.04, “Initial Characterization of Findings,” and Exhibit 2 of IMC 0609, Appendix A, “The Significance Determination Process for Findings At-Power,” issued June 19, 2012, the inspectors screened the finding for safety significance and determined it to be of very low safety significance (Green), because the finding did not represent an actual loss of system or function. After inspector questioning, Entergy performed an operability evaluation, which provided sufficient bases to conclude the Unit 3 baffle assembly would support ECCS operability. This finding is related to the cross-cutting aspect of Problem Identification and Resolution, Operating Experience, because Entergy did not effectively evaluate relevant internal and external operating experience. Specifically, Entergy did not adequately evaluate the impact of degraded baffle bolts at Unit 3 when relevant operating experience was identified at Unit 2. [P.5 – Operating Experience] (Section 1R15)

- Green. The inspectors identified a Green NCV of Technical Specification (TS) 5.4.1, "Procedures," for Entergy's failure to implement procedure OAP-007, "Containment Entry and Egress." Specifically, workers transiting the inner and outer crane wall sections of containment failed to maintain at least one (of two) flow channeling gate closed to ensure availability of the containment sumps to provide suction for the ECCS. Entergy immediately coached the gate monitor and restored the gates to an acceptable position. Entergy generated CR-IP2-2016-04036 to address this issue.

  This performance deficiency is more than minor because it was associated with the configuration control (shutdown equipment lineup) attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). A detailed risk assessment was conducted and determined that the change in core damage frequency was determined to be 7E-9, therefore, this issue represents a Green finding. This finding had a cross-cutting aspect in the area of Human Performance, Avoid Complacency, because Entergy did not consider potential undesired consequences of actions before performing work and implement appropriate error-reduction tools. Specifically, the work crew did not understand the requirements and potential consequences prior to commencing work and the gate monitor did not enforce these requirements to maintain at least one gate locked or pinned closed as required by OAP-007. [H.12 – Avoid Complacency] (Section 1R20)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a Green NCV of 10 CFR 50.65(b)(1) for Entergy's failure to include a function of a safety-related system within the scope of the maintenance rule program. Specifically, Entergy failed to include the feedwater isolation function performed by the main boiler feedwater pumps (MBFPs) discharge valves, MBFPs, and feedwater regulating valves, which are required to remain functional during and following a design basis event to mitigate the consequence of the accident within the scope of the maintenance rule monitoring program. Entergy initiated corrective actions to include the feedwater isolation function performed by the MBFP discharge valves, MBFPs, and feedwater regulating valves within the maintenance rule monitoring program. Entergy entered this issue into the CAP as CR-IP2-2016-03963.

  This performance deficiency is more than minor because it was associated with barrier performance attribute of the Barrier Integrity cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to properly scope the feedwater isolation function prevented Entergy from identifying that equipment reliability was no longer effectively controlled through preventive maintenance. In accordance with IMC 0609.04, "Initial Characterization of Findings," and Exhibit 2 of IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power," issued June 19, 2012, the inspectors determined that the finding was of very low safety significance (Green) because the finding did not represent an actual open pathway in the physical integrity of reactor containment, containment isolation system, and heat removal components. This finding does not have a cross-cutting aspect since the failure to scope this equipment into the maintenance rule program was not recognized when Entergy combined the maintenance rule basis documents for Units 2 and 3 in 2012 and, as a result, is not indicative of current licensee performance. (Section 4OA3)

REPORT DETAILS

Summary of Plant Status

Unit 2 began the inspection period during RFO 2R22 which lasted 102 days. Upon completion of the outage, the operators restarted Unit 2 on June 14, 2016, and increased power slowly to 93 percent for fuel preconditioning. On June 23, 2016, the operators shutdown the reactor to repair a service water leak on the 21 component cooling water (CCW) heat exchanger (Hx) inlet line and replace switchyard breaker 9. Unit 2 returned to 100 percent power on June 29, 2016. Unit 2 remained at or near 100 percent power for the remainder of the inspection period.

Unit 3 began the inspection period at 100 percent power. On April 26, 2016, a failed controller caused both heater drain pumps to trip; and the operators reduced power rapidly, stabilizing the unit at 48 percent power. Operators returned Unit 3 to 100 percent power on April 27, 2016, and remained at or near 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment

Partial System Walkdowns (71111.04Q – 5 samples)

a. Inspection Scope

The inspectors performed partial walkdowns of the following systems:

Unit 2

- Spent fuel pool cooling system following core offload on May 19, 2016
- Shutdown cooling system following core reload on June 6, 2016
- CCW system following maintenance on June 28, 2016

Unit 3

- 32 emergency diesel generator (EDG) following maintenance on May 9, 2016 (this sample was part of an in-depth review of the EDG system)
- Residual heat removal pumps following CCW system testing on May 20, 2016

The inspectors selected these systems based on their risk-significance relative to the reactor safety cornerstones at the time they were inspected. The inspectors reviewed applicable operating procedures, system diagrams, the updated final safety analysis report (UFSAR), TSs, work orders (WOs), condition reports (CRs), and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have impacted system performance of their intended safety functions. The inspectors also performed field walkdowns of accessible portions of the systems to verify system components and support equipment were aligned correctly and were operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no deficiencies. The inspectors also reviewed whether Entergy had properly identified equipment issues and entered them into the CAP for resolution with the appropriate significance characterization. Documents reviewed for each section of this inspection report are listed in the Attachment.

b. Findings

No findings were identified.

1R05 Fire Protection

Resident Inspector Quarterly Walkdowns (71111.05Q – 6 samples)

a. Inspection Scope

The inspectors conducted tours of the areas listed below to assess the material condition and operational status of fire protection features. The inspectors verified that Entergy controlled combustible materials and ignition sources in accordance with administrative procedures. The inspectors verified that fire protection and suppression equipment were available for use as specified in the area pre-fire plan (PFP) and passive fire barriers were maintained in good material condition. The inspectors also verified that station personnel implemented compensatory measures for out-of-service (OOS), degraded, or inoperable fire protection equipment, as applicable, in accordance with procedures.

Unit 2

- Containment, 95-foot elevation, during baffle bolt repair activities with hot work in progress (PFP-203 was reviewed) on June 2, 2016
- Residual heat removal pump rooms in primary auxiliary building (PAB), 15-foot elevation (PFP-204 was reviewed), on June 6, 2016
- CCW pump room, 68-foot elevation (PFP-209 was reviewed), on June 25, 2016
- PAB, 80-foot elevation, CCW heat exchanger area with hot work in progress (PFP-211 was reviewed) on June 25, 2016

Unit 3

- 32 EDG room, 10-foot elevation (PFP-354 was reviewed), on May 9, 2016
- 480V switchgear room, 15-foot elevation (PFP-351 was reviewed), on June 30, 2016

b. Findings

No findings were identified.

1R07 Heat Sink Performance (71111.07A – 1 sample)

a. Inspection Scope

The inspectors reviewed the 32 EDG jacket water and lube oil heat exchanger to determine its readiness and availability to perform its safety functions. The inspectors reviewed the design basis for the component and verified Entergy's commitments to NRC Generic Letter 89-13, "Service Water System Requirements Affecting Safety-Related Equipment." The inspectors observed the annual cleaning and inspection of the heat exchangers and reviewed the results of previous inspections of the Unit 3 EDG heat exchangers. The inspectors discussed the results of the most recent inspection with engineering staff. The inspectors verified that Entergy initiated appropriate corrective actions for identified deficiencies. The inspectors also verified that the number of tubes plugged within the heat exchanger did not exceed the maximum amount allowed.

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08P – 1 sample)

a. Inspection Scope

Inspectors from the NRC Region I Office, specializing in materials and inservice examination activities, observed portions of Entergy's activities involving baffle-former bolt examinations and replacements during Unit 2 RFO 2R22. The inspectors reviewed work documentation and examination procedures and results, and discussed these activities with Entergy. The inspectors were on-site from April 27 to April 28, 2016, and on May 23, 2016. The inspectors verified that Entergy completed baffle-former bolt examinations in accordance with their approved procedures which implemented activities described in the Materials Reliability Program (MRP)-227-A, "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," as they relate to this component. Specifically, the inspectors reviewed the results of the visual and volumetric examinations of the baffle-former bolts, including capabilities, limitations, and acceptance criteria that were performed during the current RFO.

Non-Destructive Examination Activities

The inspectors reviewed the ultrasonic testing (UT) procedure used for the examination of the Unit 2 baffle-former bolts to verify the procedure was in accordance with the applicable guidance in MRP-227-A and MRP-228. The inspectors reviewed the UT data records and the detailed UT channel analysis for a sample of baffle-former bolts to verify the examinations and evaluations were performed in accordance with approved procedures and applicable guidance. The inspectors reviewed video recordings of the visual examinations of the baffle-former bolts during the current RFO. The inspectors also reviewed recorded video of visual examinations performed in 2006 at Unit 2, completed as part of the existing inservice inspection program for the 10-year reactor vessel examinations, to independently assess the past conditions of the baffle-former bolts and assembly.

The inspectors reviewed certifications of the UT technicians performing the ultrasonic examinations to verify the examinations were performed by qualified individuals and to verify the results were reviewed and evaluated by certified level III non-destructive examination personnel.

Baffle-Former Bolt Replacement Activities

The inspectors reviewed the baffle-former bolt replacement activities performed as part of a corrective action to resolve the degraded condition identified at Unit 2. The inspectors observed a sample of in-process bolt removal activities, which included lock bar milling and bolt hole machining. The inspectors reviewed the documentation for in-process and completed bolt installation activities and verified that loose parts generated as part of the bolt replacements were properly tracked. The inspectors verified that bolt replacement activities were performed in accordance with approved procedures. The inspectors also reviewed the Engineering Change (EC) package associated with the new baffle-former bolt design. This review is documented in Section 1R18 of this report. After completion of the bolt replacement activities, the inspectors reviewed the video of the final visual examination of the baffle assembly to verify that the baffle-former bolt work was accomplished as planned and that there were no visual indications of deficiencies.

b. Findings

No findings were identified.

Update to URI 05000247/2016001-01, Baffle-Former Bolts with Identified Anomalies

This inspection was conducted to follow-up on NRC Unresolved Item (URI) 05000247/2016001-01, Baffle-Former Bolts with Identified Anomalies, to determine whether there was a performance deficiency associated with the degraded baffle-former bolt condition discovered at Unit 2. The inspectors plan to review additional technical information from Entergy as it becomes available, including any revisions to the root cause evaluation. The URI remains open until review of this additional information is completed. **(URI 05000247/2016001-01, Baffle-Former Bolts with Identified Anomalies)**

1R11 Licensed Operator Requalification Program (71111.11Q – 5 samples)

Unit 2

.1 Quarterly Review of Unit 2 Licensed Operator Requalification Testing and Training (71111.11Q – 1 sample)

a. Inspection Scope

The inspectors observed Unit 2 licensed operator simulator training on May 24, 2016, which included reactor coolant pump seal failure with loss of normal heat sink requiring implementation of feed and bleed cooling. The inspectors evaluated operator performance during the simulated event and verified completion of risk significant operator actions, including the use of abnormal and emergency operating procedures. The inspectors assessed the clarity and effectiveness of communications, implementation of actions in response to alarms and degrading plant conditions, and the oversight and direction provided by the control room supervisor. The inspectors verified the accuracy and timeliness of the emergency classification made by the shift manager and the TS action statements entered by the shift technical advisor. Additionally, the inspectors assessed the ability of the crew and training staff to identify and document crew performance problems.

b. Findings

No findings were identified.

.2 Quarterly Review of Unit 3 Licensed Operator Requalification Testing and Training (71111.11Q – 1 sample)

a. Inspection Scope

The inspectors observed a Unit 3 licensed operator simulator requalification training evaluated scenario on May 24, 2016, which included failure of a pressurizer pressure instrument, charging pump trip, loss of 480V safety bus 5A, a small break loss-of-coolant accident (LOCA), and entry into FR-C.2 core cooling. The inspectors evaluated operator performance during the simulated event and verified completion of risk significant operator actions, including the use of abnormal and emergency operating procedures. The inspectors assessed the clarity and effectiveness of communications, implementation of actions in response to alarms and degrading plant conditions, and the oversight and direction provided by the control room supervisor. The inspectors verified the accuracy and timeliness of the emergency classification made by the shift manager and the TS action statements entered by the shift technical advisor. Additionally, the inspectors assessed the ability of the crew and training staff to identify and document crew performance problems.

b. Findings

No findings were identified.

.3 Quarterly Review of Licensed Operator Performance (71111.11Q – 3 samples)

a. Inspection Scope

The inspectors conducted a focused observation of operator performance in the main control room. The inspectors observed pre-job briefings and control room communications to verify they met the criteria specified in Entergy's administrative procedure EN-OP-115, "Conduct of Operations." Additionally, the inspectors observed restoration activities to verify that procedure use, crew communications, and coordination of activities between work groups similarly met established expectations and standards.

Unit 2

- Plant startup from RFO 2R22 on June 16, 2016 including response to a turbine trip without a reactor trip and the subsequent turbine-generator synchronization and transfer of plant electrical loads from offsite power to the unit auxiliary transformer.
- Reactor startup and grid synchronization conducted on June 27, 2016.

Unit 3

- Operator response to the feedwater transient which occurred on April 26, 2016

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12Q – 4 samples)

.1 Routine Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the samples listed below to assess the effectiveness of maintenance activities on SSCs performance and reliability. The inspectors reviewed system health reports, CAP documents, maintenance WOs, and maintenance rule basis documents to ensure that Entergy was identifying and properly evaluating performance problems within the scope of the maintenance rule. For each SSC sample selected, the inspectors verified that the SSC was properly scoped into the maintenance rule in accordance with 10 CFR 50.65 and verified that the (a)(2) performance criteria established by Entergy was reasonable. As applicable, for SSCs classified as (a)(1), the inspectors assessed the adequacy of goals and corrective actions to return these SSCs to (a)(2). Additionally, the inspectors ensured that Entergy was identifying and addressing common cause failures that occurred within and across maintenance rule system boundaries.

- Unit 2 EDGs
- Unit 3 EDGs (this sample was part of an in-depth review of the EDG system)
- Units 2 and 3 CVCS

b. Findings

No findings were identified.

URI Opened, CVCS Goal Monitoring Under the Maintenance Rule

Introduction

The inspectors identified issues of potential concern with Entergy's application of 10 CFR 50.65(a)(1), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Plants," (the maintenance rule) in regards to the reliability of the Unit 2 CVCS system. These concerns included the establishment of appropriate (a)(1) goals and whether appropriate justification was established that the corrective actions to address identified maintenance weaknesses were effective prior to removal from (a)(1) status. Specifically, Entergy may have established restrictive goals without defensible justification and may not have demonstrated their chosen goal before ending the goal monitoring interval.

Description

The maintenance rule requires that licensees shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these structures, systems, and components are capable of fulfilling their intended functions. These goals shall be established commensurate with safety and, where practical, take into account industrywide operating experience. When the performance or condition of a structure, system, or component does not meet established goals, appropriate corrective action shall be taken. EN-DC-206, "Maintenance Rule (a)(1) Process," provides the requirements and processes for managing SSCs for which (a)(2) monitoring has not demonstrated effective maintenance. EN-DC-206 specifies that (a)(1) action plans should not be closed until effectiveness of all corrective actions has been demonstrated by meeting performance goals through the monitoring period (or by other means specified in the action plan).

Since 2013, there have been several repeat functional failures of equipment in the CVCS resulting in a failure to meet the performance criterion for reliability. These failures included:

- A failure of the 23 charging pump on August 6, 2013, after the internal oil pump discharge tubing broke causing the pump to trip on low oil pressure and a loss of charging. The 21 charging pump had tripped for the same reason in 2010.
- A failure of the 22 charging pump on January 14, 2014, due to cracked internal check valves caused by an inadequate fill-and-vent that left air in the pump following maintenance. The 21 charging pump had failed due to the same cause in 2013.
- A failure of the Unit 2 valve FCV-110A, boric acid flow control valve, to fully open on January 5, 2015. The valve had insufficient insulation; and as a result, boron crystalized above the valve plug and blocked its movement. The Unit 3 FCV-110A had failed in the same way in 2011, with earlier failures of other valves for the same cause going back to 1997.

In each case, the CVCS for Unit 2 was already (a)(1), so Entergy either updated the existing (a)(1) action plan or created another one to operate in parallel with the existing one. Upon reviewing the associated (a)(1) action plans, the inspectors noted that in each example Entergy's goals may not have been in accordance with EN-DC-206(a)(1) Process. It specifies that monitoring intervals should be at least six months for normally operating SSCs, at least three surveillances for SSCs monitored by surveillance and long enough to detect recurrence of the applicable failure mechanism. It also states that performance goals that provide reasonable assurance that the SSC is capable of performing its intended functions should be monitored throughout the time the SSC is classified (a)(1). EN-DC-206 defines an SSC as any discreet component grouping that has caused a monitoring failure, including any applicable extent of condition. In the examples provided, NRC inspectors challenged whether Entergy either chose a shorter monitoring interval or a goal that did not include the applicable extent of condition. Specifically:

- The (a)(1) action plan for the broken oil tubing had a goal of no noticeable decrease in 23 charging pump's running oil pressure for the next three quarterly surveillances. The chosen monitoring interval met the procedural expectation, but Entergy limited the monitoring to the 23 charging pump without written justification, when the 21 charging pump had failed previously for the same reason and the other pumps were susceptible to the same failure mechanism. During the monitoring interval, the 21 charging pump experienced low oil pressure. When Entergy performed repairs on the 21 charging pump for an unrelated issue, they discovered that the oil tubing had failed in the same way the 23 charging pump oil tubing had failed, although it had not yet caused a pump trip.
- The (a)(1) action plan for the cracked check valves had a goal of no check valve failure for six months for the next charging pump that underwent maintenance. This happened to be the 22 charging pump. Entergy chose a six-month monitoring interval, even though only one of the three charging pumps is in service at any given time, and the 22 charging pump only ran for four out of the six months it was monitored. Additionally, the action plan did not justify why a single successful fill-and-vent demonstrated adequate corrective actions. On November 19, 2014, during the six month monitoring interval, the 21 charging pump underwent maintenance requiring a fill-and-vent, and experienced check valve failure two weeks later on December 4. Entergy documented this as a maintenance rule functional failure, and discussed the possibility that it could be due to an inadequate fill-and-vent, but did not change the (a)(1) action plan.
- The (a)(1) action plan for FCV-110A specified a monitoring interval of six months to include the winter because the previous valve failures had all occurred during the winter months. However, the actual monitoring interval documented in the corrective action was from April to October 2015, and therefore did not cover the winter months as intended. In January 2016, Entergy performed maintenance on valve CH-297 on Unit 3, which is a heat-traced boric acid valve, and did not properly restore the insulation. The valve function was not impacted because it does not often contain high concentrations of boric acid.

The (a)(1) action plans described above were all reviewed and approved by the maintenance rule expert panel.

Further information regarding the performance of these SSCs is required to determine whether these issues of concern represent performance deficiencies and whether they are more than minor. **(URI 05000247/2016002-01, CVCS Goal Monitoring Under the Maintenance Rule)**

.2 Quality Control

a. Inspection Scope

The inspectors reviewed the weld repair performed on the 21 CCW heat exchanger service water inlet nozzle for Unit 2 to verify Entergy was properly applying quality controls specified in their quality assurance program. The inspectors reviewed CAP documents, maintenance WOs, ECs, and engineering procedures associated with the weld repair. The inspectors verified Entergy specified quality control hold points in accordance with their procedures, properly controlled the quality of materials used during the repair, and adequately justified deviations from the existing design. Additionally, the inspectors reviewed the welding procedure specification qualification by the vendor to ensure it was in accordance with American Society of Mechanical Engineers code.

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 – 7 samples)

a. Inspection Scope

The inspectors reviewed station evaluation and management of plant risk for the maintenance and emergent work activities listed below to verify that Entergy performed the appropriate risk assessments prior to removing equipment for work. The inspectors selected these activities based on potential risk significance relative to the reactor safety cornerstones. As applicable for each activity, the inspectors verified that Entergy performed risk assessments as required by 10 CFR 50.65(a)(4) and that the assessments were accurate and complete. When Entergy performed emergent work, the inspectors verified that operations personnel promptly assessed and managed plant risk. The inspectors reviewed the scope of maintenance work and discussed the results of the assessment with the station's probabilistic risk analyst to verify plant conditions were consistent with the risk assessment. The inspectors also reviewed the TS requirements and inspected portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met.

Unit 2

- Temporary loss of spent fuel pool cooling due to 345-kilovolt disturbance on April 3, 2016
- Equipment hatch closure plug seal demonstration for outage risk on April 5, 2016
- Reduced inventory operations during vessel reassembly on June 7, 2016
- 21 CCW heat exchanger OOS during mode 4 on June 25, 2016

Unit 3

- 32 EDG OOS while Bus Tie BT 4-5 was OOS on May 4, 2016 (this sample was part of an in-depth review of the EDG system)
- 33 EDG OOS while Bus Tie BT 4-5 was OOS on June 2, 2016
- 31 EDG OOS while Bus Tie BT 4-5 was OOS on June 21, 2016

b. Findings

No findings were identified.

1R15 Operability Determinations and Functionality Assessments (71111.15 – 7 samples)

a. Inspection Scope

The inspectors reviewed operability determinations for the following degraded or non-conforming conditions:

Unit 2

- 23 EDG failure to run on March 7, 2016, and subsequent failure to pass the surveillance test on March 10, 2016, as identified in CR-IP2-2016-01260
- Operability determination for N33 gamma metrics wide range nuclear instrument channel in CR-IP2-2016-03660 on June 13, 2016
- Pressurizer level transmitter LT-461 reads high in CR-IP-2016-3806 on June 14, 2016
- Through-wall leak in line 411, service water inlet to the 21 CCW heat exchanger, on June 15, 2016

Unit 3

- Immediate operability determination of the degraded condition of the baffle-former bolts identified from Unit 2 operating experience in CR-IP3-2016-01035 on April 1, 2016
- Anomalies noted during digital metal impact monitoring system self-test in CR-IP3-2015-03468 on April 1, 2016
- Prompt operability determination of the degraded condition of the baffle-former bolts identified from Unit 2 operating experience in CR-IP2-2016-03660 on June 30, 2016

The inspectors selected these issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the operability determinations to assess whether TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TSs and UFSAR to Entergy's evaluations to determine whether the components or systems were operable.

The inspectors confirmed, where appropriate, compliance with bounding limitations associated with the evaluations. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled by Entergy. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations.

b. Findings

Introduction. The inspectors identified a Green NCV of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because Entergy did not adequately accomplish the actions prescribed by procedure EN-OP-104 for a degraded condition associated with the Unit 3 baffle-former bolts. Specifically, Entergy incorrectly concluded that no degraded or non-conforming condition existed related to the Unit 3 baffle-former bolts and exited the operability determination procedure. Entergy subsequently performed the remaining steps in the procedure and provided appropriate justification for their plans to examine the baffle-former bolts at the next Unit 3 RFO.

Description. On March 29, 2016, Entergy identified baffle-former (“baffle”) bolt degradation at Indian Point Unit 2 that was determined to be unanalyzed because it did not meet the minimum acceptable bolt pattern analysis developed to support plant startup. Entergy staff identified a total of 227 baffle bolts out of a population of 832 that were potentially degraded (182 bolts had UT indications; 31 had visual indications of failure; and 14 were inaccessible for testing and conservatively assumed to be degraded). Entergy staff entered this problem into the CAP as CR-IP2-2016-02081, performed a root cause evaluation, and replaced the degraded bolts on Unit 2. Due to the number of baffle bolt indications discovered on Unit 2, Entergy staff initiated CR-IP3-2016-01035 on April 21, 2016, and performed an immediate operability determination (IOD) in accordance with Entergy procedure EN-OP-104 Section 5.3, to evaluate the baffle bolts and baffle-former assembly structure on Unit 3. Entergy staff planned further corrective actions to move up the planned Unit 3 baffle bolt ultrasonic examinations to the next RFO in spring 2017.

The inspectors reviewed the design basis and current licensing basis documents for Indian Point Unit 3 to identify the specific safety functions of the baffle bolts. The baffle bolts are part of the baffle former assembly structure located in the reactor pressure vessel. The bolts secure a series of vertical metal plates called baffle plates, which help direct water up through the nuclear fuel assemblies to ensure proper cooling of the fuel. A sufficient number of baffle bolts are required to secure the plates to ensure proper core flow during normal and postulated accident conditions, and also to ensure that control rods can be inserted to shut down the reactor.

The inspectors reviewed Entergy’s IOD issued on April 21, 2016, and concluded the immediate determination was completed in accordance with Section 5.3 of procedure EN-OP-104. The IOD provided sufficient technical detail to support the initial conclusion, based on limited information, that the Unit 3 baffle bolts would retain sufficient capability to perform their intended functions. Specifically, the IOD stated that Unit 2 baffle bolt failures were likely due to irradiation-assisted stress corrosion cracking (IASCC) and that the Unit 3 baffle bolts were also susceptible because they both utilize a baffle bolt design with similar geometry and material to other plants with bolt failures. The IOD concluded that Unit 3 baffle bolt degradation would likely not be as significant as Unit 2, and that the Unit 3 baffle former assembly was currently operable pending further evaluation because of the following differences with Unit 2: (1) less effective full power years of operation; (2) less neutron fluence levels (i.e., irradiation); (3) less pressure differential across the baffle plates; and (4) less fatigue-induced loading cycles on the bolts over the operating life of the plant. The inspectors concluded that there was no immediate safety concern.

On May 5, 2016, Entergy staff revised the operability input for CR-IP3-2016-01035 under corrective action #2. The inspectors noted that Entergy staff concluded an operability evaluation was not needed, in part, because “the baffle-former bolts are not required by TS and are not described in the UFSAR.” The inspectors noted that while the baffle bolts are not described in these documents, their failure in sufficient numbers could have consequential effects on the TS-controlled ECCS if the baffle plates were to become detached or deformed. This was described in Entergy’s bolt pattern analysis report documenting an acceptable bolt pattern prior to the spring 2016 RFO. The inspectors reviewed Unit 3 TS 3.5.2, "ECCS – Operating," which requires multiple trains of ECCS to be operable. The inspectors concluded that since the baffle bolts support the ECCS, which is subject to TS, Entergy's decision to not perform further evaluation of the operability determination was inconsistent with EN-OP-104. Specifically, Section 5.1(7) of Entergy's procedure EN-OP-104 requires that an operability determination be performed whenever a condition exists in the supporting SCC that may affect the ability of the TS-controlled SSC to perform its specified safety function.

Further, the inspectors noted that Entergy staff concluded a degraded condition did not exist in Unit 3, and therefore, an operability evaluation was not required as a follow-up to the immediate determination. The documented basis provided was the differences between the two units, plant operating data, and fuel performance. The inspectors noted that plant operating data and fuel performance from Unit 2 did not result in identification of the bolt degradation; therefore, the absence of indications for these problems on Unit 3 was technically insufficient to support Entergy's conclusion that there was no degraded condition on Unit 3.

The inspectors' review of procedure EN-OP-104, Section 3.0, identified that examples of the effects of equipment aging and operating experience can be sources of information considered to enter the operability or functionality process. The inspectors acknowledged that licensees apply judgment in these decisions. In this particular instance, the inspectors considered that operating experience was available that showed the Unit 3 baffle bolts were subject to IASCC and that plants of similar design (4-loop Westinghouse pressurized water reactors with a down-flow configuration and baffle bolts of 347 material and similar dimensions) were subject to greater amounts of bolt degradation compared to other reactor designs. Furthermore, the inspectors noted the baffle bolts had experienced levels of neutron radiation exposure above the threshold for IASCC initiation as referenced in NUREG/CR-7027, "Degradation of LWR Core Internal Materials due to Neutron Irradiation."

Based on the above information available to Entergy staff, the inspectors concluded that Entergy's basis for determining that a degraded condition did not exist on Unit 3 was not technically supported. The inspectors noted that in completing an IOD in EN-OP-104, Step 5.3.2 states "determine if there is an ongoing degradation mechanism that may impact future operability based on changing conditions, specifically consider the SSCs specified safety function and mission time." On May 5, 2016, Entergy's basis for concluding an operability evaluation was not required and exiting the operability determination procedure at Step 5.3.3 was inconsistent with this procedural requirement because their IOD concluded Unit 3 was susceptible to baffle bolt degradation, which is time based and subject to changing conditions including fatigue inducing loading cycles and neutron fluence. As a result, the inspectors concluded Entergy staff did not complete the additional actions prescribed by EN-OP-104 to perform an operability evaluation. Specifically, Step 5.3.9 states in part "if an Operability Evaluation is required then perform the following: Proceed to Subsection 5.5, Operability Evaluation."

On July 11, 2016, Entergy staff subsequently completed the steps in EN-OP-104 and performed an operability evaluation, which assumed an estimated number of baffle-former bolt failures based on the degradation found in Unit 2, and adjusted to take credit for the small number of inaccessible bolts and a sample of bolts extracted with high removal torque that indicated residual structural capacity. The inspectors determined this estimated number of bolt failures was conservative because the evaluation did not credit the baffle-edge bolts or the differences in operational history between the two units such as neutron fluence levels or fatigue from thermal cycles. The operability evaluation concluded that the Unit 3 baffle bolts would perform as intended to secure the baffle plates from being dislodged. The inspectors concluded that Entergy's operability evaluation provided appropriate basis to conclude that the Unit 3 baffle assembly would support ECCS operability until the planned Unit 3 RFO in spring 2017.

Analysis. The inspectors determined that Entergy's failure to adequately accomplish the actions prescribed in EN-OP-104 for a degraded condition and perform an operability evaluation associated with the Unit 3 baffle-former bolts was a performance deficiency. Specifically, Entergy incorrectly concluded that no degraded or non-conforming condition existed related to the Unit 3 baffle-former bolts and exited the operability determination procedure. As a result, Entergy's initial documentation did not provide sufficient basis for operability and continued operation until questioned by NRC inspectors.

This finding is more than minor because it is associated with the equipment performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). This issue was also similar to example 3.j of IMC 0612, Appendix E, "Examples of Minor Issues," because the condition resulted in reasonable doubt of operability of the ECCS and additional analysis was necessary to verify operability. In accordance with IMC 0609.04, "Initial Characterization of Findings," and Exhibit 2 of IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power," issued June 19, 2012, the inspectors screened the finding for safety significance and determined it to be of very low safety significance (Green), since the finding did not represent an actual loss of system or function. After inspector questioning, Entergy performed an operability evaluation, which provided sufficient bases to conclude the Unit 3 baffle assembly would support ECCS operability. This finding is related to the cross-cutting aspect of Problem Identification and Resolution, Operating Experience, because Entergy did not effectively evaluate relevant internal and external operating experience. Specifically, Entergy did not adequately evaluate the impact of degraded baffle bolts at Unit 3 when relevant operating experience was identified at Unit 2. [P.5]

Enforcement. 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," states, in part, that activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with those procedures. The introduction to Appendix B states that 'quality assurance' comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component (SSC) will perform satisfactorily in service. Procedure EN-OP-104, Step 5.3[2], related to immediate operability, states "Determine if there is an ongoing degradation mechanism that may impact future operability based on changing conditions, specifically consider the SSCs specified safety function and mission time." Step 5.3(3) follows with, in part "If no Degraded or Non-conforming Condition exists, then perform the following as the Immediate Determination:" "Declare the SSC Operable" and "Exit this procedure."

Contrary to the above, from May 5, 2016 until July 11, 2016, Entergy did not adequately accomplish actions as prescribed by EN-OP-104 for a degraded condition associated with the Unit 3 baffle-former bolts. Specifically, Entergy incorrectly concluded that no degraded or non-conforming condition existed related to the Unit 3 baffle-former bolts and exited the operability determination procedure. The NRC determined this is contrary to EN-OP-104 because a comparison of Unit 2 and 3 operational factors resulted in Entergy concluding that the Unit 3 baffle bolts would likely be affected due to the same degradation mechanism. Entergy's corrective actions included entering the issue into the CAP and documenting an operability evaluation to support the basis for operability of the baffle bolts and ECCS. Because this issue is of very low safety significance (Green) and Entergy entered this into their CAP as CR-IP3-2016-01961, this finding is being treated as an NCV consistent with Section 2.3.2.a of the Enforcement Policy. **(NCV 05000286/2016002-02, Failure to Follow Operability Determination Procedure for Unit 3 Baffle-Former Bolts)**

<u>Update to URI 05000247/2016001-06, 23 Emergency Diesel Generator Automatic Voltage Regulator Failure</u>

<u>Introduction</u>. The NRC opened a URI in Inspection Report 05000247/2016001 related to two failures of the 23 EDG to run and maintain bus voltage on March 7, 2016, and to provide adequate control of bus voltage on March 10, 2016. This report provides an update of the status of this URI.

<u>Description</u>. On March 7, 2016, approximately one hour after the trip of the 3A normal feed breaker, the 23 EDG tripped on overcurrent while powering the 6A 480V safety bus. The 6A bus remained de-energized for approximately one hour until the crew restored the 6A bus via off-site power. The 23 EDG was declared inoperable. All four 480V safety buses were restored to off-site power. Entergy replaced the overcurrent relays and retested the 23 EDG satisfactorily on March 8, 2016. However, bench testing of the overcurrent relays demonstrated that they were accurately calibrated.

Subsequently, on March 10, 2016, during performance of PT-R14, "Automatic Safety Injection System Electrical Load and Blackout Test," the 23 EDG exhibited anomalous behavior during the train 'B' load sequencing. During this test, the voltage on safety bus 6A dropped to approximately 200V when the 23 auxiliary feedwater pump was sequenced onto the bus (CR-IP2-2016-01430) and the sequencer failed to complete the first two sequences. The 23 EDG was again declared inoperable and the period of inoperability was backdated to March 7, 2016, when it originally tripped. Further troubleshooting and additional failure modes analysis by Entergy initially determined that the cause of both events may have been a degraded resistor (R25) on the 23 EDG automatic voltage regulator (AVR) card.

The 23 EDG AVR card was replaced, and the 23 EDG was again tested satisfactorily. The voltage anomaly issues exhibited during the March 10, 2016, test were documented in CR-IP2-2016-01430 which was closed in CR-IP2-2016-01260 to be included in the causal assessment associated with the tripping of 23 EDG breaker on March 7, 2016. Entergy assigned a vendor to perform laboratory bench testing and failure analysis of the 23 EDG AVR card. The vendor report attributed the cause of the March 10, 2016, loss of voltage control to a degraded solder joint on the AVR card. However, the vendor report explicitly did not attribute the event on March 7, 2016, to the same cause. Entergy assigned a corrective action in CR-IP2-2016-01260 to review the cause of the

23 EDG overcurrent trip on March 7, 2016, in light of the vendor report. The inspectors determined that the issue of concern remains open as a URI until this causal assessment has been completed by Entergy and assessed by NRC. **(URI 05000247/2016001-06, 23 Emergency Diesel Generator Automatic Voltage Regulator Failure)**

1R18 Plant Modifications (71111.18 – 2 samples)

Permanent Modifications

.1 Control Rod Guide Tube Repairs in Location E-9

a. Inspection Scope

The inspectors evaluated a modification to the reactor vessel upper internals to swap damaged control rod guide tube in location E-9 with abandoned guide tube in location D-10. The inspectors verified that the design bases, licensing bases, and performance capability of the affected systems were not degraded by the modification. In addition, the inspectors reviewed modification documents associated with the design change, including evaluation of equivalency and core flow changes, and post-modification testing. The inspectors also reviewed revisions to the affected drawings and interviewed refueling and engineering personnel.

b. Findings

No findings were identified.

.2 Core Baffle-Former Bolt EC 64038

a. Inspection Scope

The inspectors reviewed EC 64038, "IP2 Reactor Vessel Equivalent Replacement Baffle-to-Former Bolt." This modification was completed during RFO 2R22 and involved the replacement of 278 baffle-former bolts out of a total of 832 located in the Unit 2 reactor vessel. Entergy replaced all of the bolts that were potentially degraded as observed by visual indications of a protruding bolt head or lock bar problem, bolts that did not pass UT, and bolts inaccessible for UT. Entergy staff also replaced 51 additional bolts that passed ultrasonic and visual examinations to increase the structural margin of the baffle-former assembly for future operating cycles.

The inspectors reviewed the equivalency evaluation completed by Entergy staff to install baffle-former bolts of a different material and configuration than the original bolts. The inspectors reviewed the associated EC package to determine whether the replacement bolts' form, fit, and function were maintained compared to the original bolts and whether the change conformed to the design and licensing bases of the baffle-former assembly. Specifically, this change involved replacing the original baffle-former bolts made of type 347 stainless steel with bolts made of type 316 stainless steel. The baffle-former bolt head configuration was also changed from an original internal hex and slot design (secured with a welded lock bar) to an external hex configuration with an integral locking cup design. The design change document further evaluated a more gradual fillet geometry between the bolt head and shank intended to reduce the stress concentration at that transition and provide for improved fatigue resistance.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19 – 8 samples)

a. Inspection Scope

The inspectors reviewed the post-maintenance tests for the maintenance activities listed below to verify that procedures and test activities ensured system operability and functional capability. The inspectors reviewed the test procedure to verify that the procedure adequately tested the safety functions that may have been affected by the maintenance activity, that the acceptance criteria in the procedure was consistent with the information in the applicable licensing basis and/or design basis documents, and that the test results were properly reviewed and accepted and problems were appropriately documented. The inspectors also walked down the affected job site, observed the pre-job brief and post-job critique where possible, confirmed work site cleanliness was maintained, witnessed the test or reviewed test data to verify quality control hold points were performed and checked, and that results adequately demonstrated restoration of the affected safety functions.

Unit 2

- 21 EDG fuel oil transfer pump after planned maintenance on May 5, 2016
- Replacement of pressurizer level transmitters LT-459 and LT-460 on May 25, 2016
- 21 CCW heat exchanger service water outlet weld repair on June 26, 2016
- Flux mapping system drive repairs following motor failures on June 28, 2016

Unit 3

- Maintenance on service water components associated with the 32 EDG on May 5, 2016 (this sample was part of an in-depth review of the EDG system)
- Modification of the 32 EDG space heaters on May 5, 2016 (this sample was part of an in-depth review of the EDG system)
- Maintenance on the 32 EDG air start system on May 6, 2016 (this sample was part of an in-depth review of the EDG system)
- Replacement of failed bistable LC-427K, steam generator 32 low level mismatch trip interlock, on May 18, 2016

b. Findings

No findings were identified.

1R20 Refueling and Other Outage Activities (71111.20 – 2 samples)

.1 Unit 2 RFO 2R22

a. Inspection Scope

The inspectors reviewed the station's work schedule and outage risk plan for the Unit 2 maintenance during RFO 2R22, which was conducted from March 7, 2016, to June 16, 2016. The inspectors reviewed Entergy's development and implementation of outage plans and schedules to verify that risk, industry experience, previous site-specific problems, and defense-in-depth were considered. During the outage, the inspectors observed portions of the shutdown and cooldown processes and monitored controls associated with the following outage activities:

- Configuration management, including maintenance of defense-in-depth, commensurate with the outage plan for the key safety functions and compliance with the applicable TSs when taking equipment OOS
- Implementation of clearance activities and confirmation that tags were properly hung and that equipment was appropriately configured to safely support the associated work or testing
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication and instrument error accounting
- Status and configuration of electrical systems and switchyard activities to ensure that TSs were met
- Monitoring of decay heat removal operations
- Impact of outage work on the ability of the operators to operate the spent fuel pool cooling system
- Reactor water inventory controls, including flow paths, configurations, alternative means for inventory additions, and controls to prevent inventory loss
- Activities that could affect reactivity
- Maintenance of secondary containment as required by TSs
- Refueling activities, including fuel handling and fuel receipt inspections
- Fatigue management
- Tracking of startup prerequisites, walkdown of the primary containment to verify that debris had not been left which could block the ECCS suction strainers, and startup and ascension to full power operation
- Foreign Object Search and Retrieval for missing baffle bolts and locking tabs
- Identification and resolution of problems related to RFO activities

During this outage, Entergy replaced 278 degraded baffle bolts in the Unit 2 reactor vessel baffle assembly. This emergent project resulted in the extension of the outage schedule from 30 days to 102 days.

b. Findings

Introduction. The inspectors identified a Green NCV of TS 5.4.1 for Entergy's failure to implement procedure OAP-007, "Containment Entry and Egress." Specifically, workers transiting the inner and outer crane wall sections of containment on June 11, 2016, failed to maintain at least one (of two) flow channeling gate closed to ensure availability of the containment sumps to provide suction for the ECCS.

Description. On June 11, 2016, in mode 4, in preparation for reactor startup, Entergy was performing maintenance in containment required prior to mode 3, such as reactor coolant pump motor balancing and steam flow transmitter troubleshooting. These activities required scaffolds to be temporarily erected for workers to safely perform maintenance. While transiting from the inner to outer section of containment, the inspectors noted that both flow channeling gates were maintained open simultaneously as workers carried scaffold poles and hardware out of the area.

In the event of a postulated LOCA, Unit 2 relies on two sumps to provide a suction source for the internal recirculation pumps and residual heat removal pumps, respectively, after the injection phase of the accident. The sumps have cylindrical screens with large surface area and small holes to filter small debris and maintain adequate net positive suction head for the associated pumps. The reactor cavity sump and large intervening barriers prevent large debris generated from the accident, such as insulation, from reaching and blocking the recirculation and containment sump screens.

Entergy procedure OAP-007, "Containment Entry and Egress," precaution and limitation step 2.30.2, states, "In mode 1, 2, 3, or 4, entry inside the crane wall shall use the double gate entry point via gates 17 and 23. One gate shall remain shut and secured at all times to maintain flow channeling and sump operability. Securing gates requires a padlock or nut and bolt closure from the outside. This will require posting a gate monitor to allow exit." The inspectors noted, while a gate monitor was posted, both gates were maintained open during passage and not secured with a padlock or nut and bolt closure. Upon questioning by the inspectors, Entergy immediately coached the gate monitor and restored the gates to an acceptable position. Entergy generated CR-IP2-2016-04036 to address this issue.

Analysis. The inspectors determined that Energy's failure to maintain either gate 17 or gate 23 closed during passage in accordance with OAP-007 was a performance deficiency. The performance deficiency was more than minor because it is associated with the configuration control (shutdown equipment lineup) attribute and adversely affected the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The inspectors evaluated the finding in accordance with IMC 0609, Appendix G, Attachment 1, Exhibit 3, and determined that a detailed risk evaluation was necessary because the finding represented a loss of system safety function. A detailed risk assessment was conducted conservatively assuming complete failure of the recirculation and containment sumps due to the performance deficiency. Given that Unit 2 was in mode 4, in plant operating state 1, with a late time window, the at-power simplified plant analysis risk model for large-break LOCAs was determined to best model the degrade condition and plant response. An exposure time of one day was assumed. No credit was assumed for the decrease in energy that would be anticipated in a release during a LOCA in mode 4, nor the corresponding reduction in debris generation. This was also considered conservative. Utilizing Systems Analysis Program for Hands-On Integrated Reliability Evaluation, version 8.13, with Indian Point Unit 2 Simplified Plant Analysis Risk Model, version 8.19, for the assumed conditions, the change in core damage frequency was determined to be 7E-9. Therefore, this issue represents a Green finding.

This finding had a cross-cutting aspect in the area of Human Performance, Avoid Complacency, because Entergy did not consider potential undesired consequences of actions before performing work and implement appropriate error-reduction tools. Specifically, the work crew did not understand the requirements and potential consequences prior to commencing work and the gate monitor did not enforce these requirements to maintain at least one gate locked or pinned closed as required by OAP-007. [H.12]

Enforcement. Unit 2 TS 5.4.1.a requires that the procedures listed in Attachment A to Regulatory Guide 1.33, “Quality Assurance Program Requirements,” Revision 2, be established and implemented. Attachment A states that instructions should be prepared, as appropriate, for access to containment and changing modes of operation of the ECCS. Entergy procedure OAP-007, “Containment Entry and Egress,” Step 2.30.2, states, “In mode 1, 2, 3, or 4, entry inside the crane wall shall use the double gate entry point via gates 17 and 23. One gate shall remain shut and secured at all times to maintain flow channeling and sump operability. Securing gates requires a padlock or nut and bolt closure from the outside.” Contrary to the above, on June 11, 2016, Entergy did not maintain one gate secured at all times with a padlock or nut and bolt closure. Entergy entered this issue into the CAP as CR-IP2-2016-04036. Because this violation was of very low safety significance (Green), and Entergy entered this performance deficiency into the CAP, the NRC is treating this as a NCV in accordance with Section 2.3.2.a of the NRC Enforcement Policy. **(NCV 05000247/2016002-03, Failure to Maintain Flow Channeling Gates Closed in Accordance with the Containment Procedure)**

.2 Unit 2 Forced Outage

a. Inspection Scope

Unit 2 conducted a forced outage from June 24 to 27, 2016, in order to complete weld repairs on a through-wall leak on the service water inlet line to the 21 CCW heat exchanger. These repairs required shutting down to mode 4 in order to meet the TS 3.7.7, “Component Cooling Water (CCW) System,” limiting condition for operations for CCW operability. While these repairs were being completed, the grid operator completed repairs to breaker 9 in the offsite switchyard. During the outage, the inspectors observed portions of the shutdown and cooldown processes and monitored controls associated with the following outage activities:

- Configuration management, including maintenance of defense-in-depth, commensurate with the outage plan for the key safety functions and compliance with the applicable TSs when taking equipment OOS
- Implementation of clearance activities and confirmation that tags were properly hung and that equipment was appropriately configured to safely support the associated work or testing
- Status and configuration of electrical systems and switchyard activities to ensure that TSs were met
- Monitoring of decay heat removal operations
- Reactor water inventory controls, including flow paths, configurations, alternative means for inventory additions, and controls to prevent inventory loss
- Activities that could affect reactivity

- Tracking of startup prerequisites
- Identification and resolution of problems related to RFO activities

When all repairs had been completed, Entergy restarted Unit 2 on June 27, 2016.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22 – 6 samples)

a. Inspection Scope

The inspectors observed performance of surveillance tests and/or reviewed test data of selected risk-significant SSCs to assess whether test results satisfied TSs, the UFSAR, and Entergy's procedure requirements. The inspectors verified that test acceptance criteria were clear, tests demonstrated operational readiness and were consistent with design documentation, test instrumentation had current calibrations and the range and accuracy for the application, tests were performed as written, and applicable test prerequisites were satisfied. Upon test completion, the inspectors considered whether the test results supported that equipment was capable of performing the required safety functions. The inspectors reviewed the following surveillance tests:

Unit 2

- WO 446385, 21 EDG AVR card inspection, on May 24, 2016
- 2-PT-Q013 for containment isolation valve 851B (22 safety injection (SI) pump tie to 23 SI pump discharge) on June 6, 2016
- 2-PT-Q029B quarterly in-service surveillance test for the 22 SI pump on June 6, 2016

Unit 3

- 3-PT-M079B 32 EDG monthly surveillance on May 30, 2016 (this sample was part of an in-depth review of the EDG system)
- 34 steam generator pressure instrument channel check on June 21, 2016
- 0-SOP-LEAKRATE-001, RCS Leakrate Surveillance, Evaluation and Leak Identification, beginning on June 28, 2016

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06 – 1 sample)

Training Observations

a. Inspection Scope

The inspectors evaluated the conduct of Entergy's ingestion pathway emergency preparedness drill on April 19, 2016, to identify any weaknesses and deficiencies in the classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the emergency operations facility to determine whether the event classification, notifications, and protective action recommendations were performed in accordance with procedures. The inspectors also attended the facility drill critique to compare inspector observations with those identified by Entergy staff in order to evaluate Entergy's critique and to verify whether the staff was properly identifying weaknesses and entering them into the CAP.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety and Occupational Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

a. Inspection Scope

During May 10–12 and June 13–17, 2016, the inspectors reviewed Entergy's performance in assessing the radiological hazards and exposure control in the workplace. The inspectors used the requirements in 10 CFR 20, TSs, applicable industry standards, and procedures required by TSs as criteria for determining compliance.

Radiological Hazards Control and Work Coverage

The inspectors reviewed:

- Ambient radiological conditions during tours of the radiological controlled area, posted surveys, radiation work permits, adequacy of radiological controls, radiation protection job coverage, and contamination controls
- Controls for highly activated or contaminated materials stored within spent fuel pools
- Posting and physical controls for high radiation areas and very high radiation areas

b. Findings

No findings were identified.

2RS2 Occupational As Low As Is Reasonably Achievable (ALARA) Planning and Controls (71124.02)

a. Inspection Scope

During May 10–12 and June 13–17, 2016, the inspectors assessed performance with respect to maintaining occupational individual and collective radiation exposures ALARA. The inspectors used the requirements in 10 CFR 20, TSs, applicable industry standards, and procedures required by TSs as criteria for determining compliance.

Radiological Work Planning

The inspectors reviewed:

- ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements
- ALARA work planning, use of dose mitigation features and dose goals
- Work planning and the integration of ALARA requirements

b. Findings

No findings were identified.

2RS7 Radiological Environmental Monitoring Program (REMP) (71124.07 – 3 samples)

a. Inspection Scope

The inspectors reviewed the REMP to validate the effectiveness of the radioactive gaseous and liquid effluent release program and implementation of the groundwater protection initiative (GPI). The inspectors used the requirements in 10 CFR 20, 40 CFR 190, 10 CFR 50, Appendix I, TSs, offsite dose calculation manual (ODCM), Nuclear Energy Institute 07-07, and procedures required by TSs as criteria for determining compliance.

Inspection Planning

The inspectors reviewed Entergy's 2014 and 2015 annual radiological environmental and effluent monitoring reports, REMP program audits, ODCM changes, land use census, the UFSAR, and inter-laboratory comparison program results.

Site Inspection

The inspectors walked down various thermoluminescent dosimeter and air and water sampling locations and reviewed associated calibration and maintenance records. The inspectors observed the sampling of various environmental media as specified in the ODCM and reviewed any anomalous environmental sampling events including assessment of any positive radioactivity results. The inspectors reviewed any changes to the ODCM. The inspectors verified the operability and calibration of the meteorological tower instruments and meteorological data readouts. The inspectors reviewed environmental sample laboratory analysis results, laboratory instrument measurement detection sensitivities, laboratory quality control program audit results, and the inter- and intra-laboratory comparison program results. The inspectors reviewed the groundwater monitoring program as it applies to selected potential leaking SSCs.

GPI Implementation

The inspectors reviewed groundwater monitoring results, changes to the GPI program since the last inspection, anomalous results or missed groundwater samples, leakage or spill events including entries made into the decommissioning files (10 CFR 50.75(g)), evaluations of surface water discharges, and Entergy's evaluation of any positive groundwater sample results including appropriate stakeholder notifications and effluent reporting requirements.

Identification and Resolution of Problems

The inspectors evaluated whether problems associated with the REMP were identified at an appropriate threshold and properly addressed in Entergy's CAP.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151 – 6 samples)

Initiating Events Performance Indicators

a. Inspection Scope

The inspectors reviewed Entergy's submittals for the following Initiating Events cornerstone performance indicators for the period April 1, 2015, to March 31, 2016:

Unit 2

- Unplanned scrams per 7000 critical hours (IE01)
- Unplanned power changes per 7000 critical hours (IE03)
- Unplanned scrams with complications (IE04)

Unit 3

- Unplanned scrams (IE01)
- Unplanned power changes (IE03)
- Unplanned scrams with complications (IE04)

To determine the accuracy of the performance indicator data reported during those periods, inspectors used definitions and guidance contained in Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7. The inspectors reviewed Entergy's operator narrative logs, maintenance planning schedules, CRs, event reports, and NRC integrated inspection reports to validate the accuracy of the submittals. There were no unplanned power changes or scrams with complications during the review period.

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152 – 4 samples)

.1 Routine Review of Problem Identification and Resolution Activities

a. Inspection Scope

As required by Inspection Procedure 71152, “Problem Identification and Resolution,” the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that Entergy entered issues into the CAP at an appropriate threshold, gave adequate attention to timely corrective actions, and identified and addressed adverse trends. In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow up, the inspectors performed a daily screening of items entered into the CAP and periodically attended CR screening meetings. The inspectors also confirmed, on a sampling basis, that, as applicable, for identified defects and non-conformances, Entergy performed an evaluation in accordance with 10 CFR 21.

b. Findings

No findings were identified.

.2 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a semi-annual review of site issues, as required by Inspection Procedure 71152, “Problem Identification and Resolution,” to identify trends that might indicate the existence of more significant safety issues. In this review, the inspectors included repetitive or closely-related issues that may have been documented by Entergy outside of the CAP, such as trend reports, performance indicators, major equipment problem lists, system health reports, maintenance rule assessments, and maintenance or CAP backlogs. The inspectors also reviewed Entergy’s CAP database for the first and second quarters of 2016 to assess CRs written in various subject areas (equipment problems, human performance issues, etc.), as well as individual issues identified during the NRCs daily CR review (Section 4OA2.1). The inspectors reviewed the Entergy quarterly trend report for the first quarter of 2016 to verify that Entergy was appropriately evaluating and trending adverse conditions in accordance with applicable procedures.

b. Findings and Observations

No findings were identified.

The inspectors identified a trend in work being performed that was contrary to written work instructions and procedures, and work packages had been closed out without documenting the deviation from the work order. While reviewing completed work order WO 447966 Task 10, Internal Coating Repair for 21 CCW HX, the inspectors found a note in the work order stating that the internal coating repair to the pipe had not been done in accordance with the engineering change. The engineering change had been written when the coating repair was expected to be small, but the actual area that was recoated was much larger. A larger area of coating increases the impact on the heat exchanger if the coating were to flake off and block the flow of service water. The work package was closed and no condition report was written. This performance deficiency is minor because the coating was applied with procedurally directed quality controls and the likelihood that it would flake off is very small; and is the same as the original smaller area specified in the work package. However, the work package was closed without documenting the deviation and no CR was written.

In another example, the inspectors noted that WO 412920 Task 15 to perform a surge test on 11 centrifugal air compressor (CENTAC) after overhaul was completed on December 22, 2015. However, the completion notes and documentation for the task showed that the test was unable to be performed due to a test equipment problem. The work package was closed and no CR was written. Subsequently, after being returned to service, the compressor failed in service due to multiple surging events on January 7, 2016. Troubleshooting under WO 433939 revealed that the motor high load limit had not been adjusted to account for the increased load due to reduced compressor clearances introduced by the overhaul. This performance deficiency is screened to minor because the 11 CENTAC is not a safety-related piece of equipment and would not affect the MC 0609 cornerstone thresholds or other generic criteria. Unit 2 and Unit 3 have dedicated instrument air compressors that are credited in the FSAR to respond to a loss of instrument air event. If the 11 CENTAC (located in Unit 1) were to fail, the unit-specific IACs would automatically start to prevent a loss of instrument air at both Units 2 and 3.

A third recent example of work being performed contrary to written instructions occurred during 2RFO22 when the inspectors identified that the workers deviated from the surveillance procedure by demonstrating the installation of the emergency containment hatch plug without properly inflating the plug seals as directed by the procedure. This performance deficiency was previously documented in a prior inspection report as non-cited violation 05000247/05000286/2016001-02, “Failure to Adequately Implement Risk Management Actions for the Containment Key Safety Function.”

In all cases, the deviations from written work instructions were directed by Entergy supervision. In addition, the inspectors noted that Entergy had self-identified similar observations where work packages or condition reports had been closed without fully completing the specified actions including CR-IP2-2015-05833, CR-IP2-2016-00103, CR-IP3-2015-04729, CR-IP3-2016-00072, CR-IP3-2016-00075, and CR-IP3-2015-04019. These CRs are further examples of work orders that were closed with deviations that were not documented or resolved. Nuclear Oversight had identified several of these condition reports. Entergy has taking immediate corrective action in response to these performance deficiencies.

.3 Annual Sample: Maintenance Rule Self-Assessment of Corrective Actions

a. Inspection Scope

The inspectors performed an in-depth review of Entergy's corrective actions associated with self-assessment LO-IP3LO-2015-72, "Maintenance Rule (a)(3) Assessment." The self-assessment was performed to satisfy both the self-assessment criteria in EN-LI-104, "Self-Assessment and Benchmark Process," and the maintenance rule periodic assessment criteria in EN-DC-207.

The inspectors assessed Entergy's problem identification threshold, extent of condition reviews, and the prioritization and timeliness of Entergy corrective actions to determine whether Entergy was appropriately identifying, characterizing, and correcting problems associated with this issue and whether the planned or completed corrective actions were appropriate. The inspectors compared the actions taken to the requirements of Entergy's CAP and 10 CFR 50, Appendix B. In addition, the inspectors interviewed engineering personnel to assess the effectiveness of the implemented corrective actions.

b. Findings and Observations

No findings were identified.

Entergy identified three standard deficiencies during their self-assessment and wrote CRs to document each one. One of the standard deficiencies was that the maintenance rule basis documents were not being reviewed at least once every two years as required by procedure EN-DC-204, "Maintenance Rule Scope and Basis." The purpose of this review was to ensure that the documents were updated if the configuration of the system changed or if the performance criteria needed to be adjusted. Entergy wrote CR-IP3-2015-03628 and assigned a corrective action to create work trackers to perform the basis document reviews. They chose to use work trackers instead of corrective actions under the CAP because the work had historically been assigned using work trackers. However, because work trackers do not receive the same priority as corrective actions, some of the maintenance rule basis documents had still not been reviewed at the time of this inspection, over a year after the completion of the self-assessment. The inspectors determined that this was not a more than minor issue because the systems in question did not show signs of inadequate maintenance.

.4 Annual Sample: Unit 2 Reactor Trip on December 5, 2015

a. Inspection Scope

The inspectors performed an in-depth review of Entergy's evaluations and corrective actions associated with CR-IP2-2015-05484 and the related apparent cause evaluation for the December 5, 2015, manual reactor trip in response to indications of multiple dropped control rods caused by the loss of control rod power due to a power supply failure. Entergy performed an apparent cause evaluation and determined the direct cause of the event was the loss of motor control center (MCC)-24 due to an internal fault at the line side leads at cubicle 2H where they connect to the bucket stab assemblies. The apparent cause was an unanticipated loss of power to the control rod system due to the degradation of the primary control rod power supply (PS1) which failed to function for more than 10 minutes when the operating alternate power supply (PS2) was deenergized.

The inspectors assessed Entergy's problem identification threshold, problem analysis, extent of condition reviews, compensatory actions, and the prioritization and timeliness of Entergy's corrective actions to determine whether Entergy was appropriately identifying, characterizing, and correcting problems associated with this issue and whether the planned or completed corrective actions were appropriate. The inspectors compared the actions taken to the requirements of Entergy's CAP and 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action."

b. Findings and Observations

No findings were identified.

The inspectors found that Entergy took appropriate actions to identify the direct and apparent cause of the issue. The direct cause of the event was the loss of MCC-24 due to an internal fault at the line side leads at cubicle 2H where they connect to the bucket stab assemblies. The apparent cause was an unanticipated loss of power to the control rod system due to the degradation of the primary control rod PS1, which failed to function when PS2 was lost. Entergy replaced the degraded rod control PS1; and the MCC-24 compartments were removed to facilitate inspection and testing of the MCC bus, control wires, and MCC internal. PS2 was also restored to operation after the fault was cleared.

The inspector determined that the internal electrical fault that deenergized PS2 and the prior degradation in PS1 was not within Entergy's ability to foresee and prevent. Therefore, there was no performance deficiency identified. Entergy's overall response to the issue was commensurate with the safety significance, was timely, and the actions taken and planned were reasonable to resolve the failure of the primary control rod PS1.

.5 Annual Sample: Unexpected Number of Degraded Baffle-Former Bolts Discovered in the Unit 2 Reactor Pressure Vessel

a. Inspection Scope

The inspectors performed an in-depth review of Entergy's root cause evaluation and corrective actions associated with CR-IP2-2016-02348 for baffle-former ("baffle") bolts found with indications of degradation during the Indian Point Unit 2 RFO 2R22. Entergy performed ultrasonic examinations of the baffle bolts in accordance with their procedures as part of a planned activity. After an unexpected number of degraded baffle bolts were discovered, Entergy staff reported the issue to the NRC as Event Notification No. 51829 on March 29, 2016, because the as-found number and location of degraded bolts represented an unanalyzed condition. Entergy staff completed corrective actions to replace all of the potentially degraded baffle bolts on Unit 2. Entergy staff further replaced a population of additional bolts that exhibited no indications of degradation and performed an evaluation to determine the potential for baffle bolt failures at Unit 3.

The baffle-former bolts help secure vertical plates (also referred to as baffle plates) inside the reactor vessel, which then forms a structure surrounding the reactor fuel assemblies to orient the fuel and to direct coolant flow through the core. A sufficient number of baffle bolts are required to remain intact to secure the baffle plates in place so as to not affect control rod insertion or impede emergency core cooling flow during postulated accident conditions. Bolt heads that separate and are no longer held in place by bolt lock-tabs can also become a loose parts concern.

The inspectors determined whether Entergy's acceptable baffle bolt pattern analysis for Unit 2 was completed in accordance with the NRC-approved methodology and provided appropriate structural margin for the next cycle of operation to ensure the Unit 2 baffle plates will remain in place during both normal operation and limiting postulated accident conditions. The inspectors further determined whether Entergy's evaluations of the baffle bolts installed in Indian Point Unit 3 were technically sufficient to conclude the Unit 3 baffle assembly will perform as intended until the next planned RFO, at which time Entergy plans to examine the bolts. The inspectors reviewed Entergy's procedures for determining the functionality and operability of degraded SSC as they relate to Unit 3. The inspectors further interviewed Entergy engineering personnel and contractor staff to discuss the results of Entergy's technical evaluations and to assess the effectiveness of the implemented and planned corrective actions.

The inspectors assessed Entergy's problem identification threshold, cause analyses, extent of condition, compensatory actions, and the prioritization and timeliness of Entergy's corrective actions to determine whether Entergy staff were properly identifying, characterizing, and correcting problems associated with this issue and whether the planned or completed corrective actions were appropriate. The inspectors compared the actions taken to Entergy's CAP, operability determination process, and the requirements of 10 CFR 50, Appendix B. The inspectors observed portions of baffle bolt replacement activities at Unit 2 and reviewed the final visual examination of the baffle bolts and plates once the work was completed.

b. Findings and Observations

One Green NCV was identified and documented in Section 1R15 of this report.
The NRC responded to the initial discovery of an unexpected number of baffle bolts found degraded at Indian Point Unit 2 by implementing a comprehensive inspection plan consisting of various baseline inspection samples to assess the extent of the issue and to determine the necessary NRC actions. A follow-up inservice inspection sample (Refer to Section 1R08) was conducted to review the capability of the non-destructive examination techniques, evaluate the UT results, and observe a portion of bolt replacement activities on-site. A permanent modification sample (Refer to Section 1R18) was conducted to review the design change package and evaluations associated with the new, replacement baffle bolts. The NRC resident inspectors reviewed Entergy's foreign material controls and loose parts analysis (Refer to Section 1R20) to address the potential for missing bolt heads and concluded it would not impact safe operation of the plant.

NRC Region I based inspectors accompanied by an expert from the NRC Office of Nuclear Reactor Regulation completed an annual problem identification and resolution inspection, documented in this section of the report, to verify that Entergy's evaluations and corrective actions to replace Unit 2 baffle bolts were completed in accordance with an NRC approved methodology to support a conclusion that the Unit 2 baffle assembly meets the plant design basis. The inspectors also determined the adequacy of Entergy's evaluations completed to determine there is a reasonable expectation that the

Unit 3 baffle assembly will perform as intended during the current operating cycle. The results of this review are discussed herein and in Section 1R15 of this report.

Entergy staff determined the cause of the degraded baffle bolts was primarily due to IASCC in combination with increased fatigue loading on the baffle plates. This cause determination was based on industry operating experience related to baffle-former bolt failure in both foreign and domestic plants. IASCC is a cracking mechanism that occurs over a long period of time when susceptible metals are exposed to neutron radiation from the reactor core and stresses as part of normal design and operation. Entergy staff concluded that failure of a critical number of bolts in a localized area subsequently imposed increased loading on adjacent bolts, which propagated failures and generated the moderate clustered pattern observed in the examination results. No other contributing causes were identified.

The inspectors reviewed Entergy's root cause evaluation and the supporting operating experience related to baffle bolt failures at other plants. The inspectors determined that there is documented evidence in the existing technical literature (including materials testing of bolts from other plants) and operating experience to conclude that the likely cause is IASCC; however, the inspectors found that Entergy staff did not define the cause of the fatigue failure mechanism. The inspectors noted that Entergy staff sent a sample of baffle bolts removed from the reactor pressure vessel to a metallurgical laboratory for detailed failure analysis and materials property testing. Entergy indicated their plans to use the results of the laboratory testing to confirm the likely root cause. The inspectors concluded that Entergy staff conducted an appropriate review to identify the likely causes of the degraded baffle bolts and noted that further test results will be used to confirm these causes.

Following identification of the degraded baffle bolts on Unit 2, Entergy's immediate corrective action was to analyze the as-found condition and begin replacing bolts that either had visual indications of bolt failure (protruding bolt head for example), did not pass UT examination, or were not accessible for UT examination. The as-found number and pattern of these bolts exceeded the acceptance criteria in the plant's analysis that was prepared in advance of the baffle bolt examinations; therefore, Entergy reported this discovery to the NRC as an unanalyzed condition. Entergy staff completed corrective actions to replace all of the 227 potentially degraded baffle bolts, plus an additional 51 bolts for increased structural integrity, for a total of 278 bolts. The inspectors noted the 51 additional bolts were installed in strategic locations to prevent clustering of potential bolt failures during the next operating cycle.

The inspectors determined that Entergy staff performed an acceptable bolt pattern analysis that evaluated the replacement bolt pattern for Unit 2 and modeled the potential for future bolt failures. The inspectors found the results of the analysis accounted for a conservative failure rate of bolts and provided appropriate margin for one cycle of operation. The inspectors verified that Entergy's methodology for its acceptable bolt pattern analyses, including its determination of margin, was consistent with the NRC-approved methodology in topical report WCAP-15029-NP-A (ML15222A882). The inspectors determined that Entergy staff tracked corrective actions to re-examine the Unit 2 baffle bolts during the next planned RFO. The inspectors noted the new baffle bolts were made of a material with improved resistance to IASCC and included an improved design to reduce the stresses at the head to shank transition, both of which are enhancements compared to the original bolts.

As part of an extent of condition assessment, Entergy entered CR-IP3-2016-01035 in its CAP to evaluate the potential for degraded baffle bolts on Unit 3. Entergy operators performed an IOD and concluded that the baffle assembly was operable. Entergy staff performed a subsequent "extent of condition review" that concluded Unit 3 would experience less baffle bolt degradation than Unit 2 based on several plant factors. Entergy also conducted sensitivity analyses to show acceptable bounding conditions in the event of bolt failures. The inspectors reviewed Entergy's evaluations and noted that Entergy staff concluded there was not a degraded condition at Unit 3. In consideration of the guidance in their operability procedure and operating experience from Unit 2 and other plants, the NRC issued an NCV in this report because Entergy did not perform an operability evaluation for Unit 3 as a follow-up to the immediate determination for the potential impact on supported systems controlled by the TS (Refer to Section 1R15).

As a corrective action, Entergy staff performed an operability evaluation and demonstrated that the Unit 3 baffle former assembly remained operable. The inspectors concluded that this supplemental evaluation provided appropriate technical justification for the continued operation of Unit 3 until the next RFO in spring 2017, at which time Entergy plans to examine the baffle bolts. Entergy also implemented a corrective action as part of an enhancement to plant operations to monitor the RCS for any signs of fuel leakage, which could be an indicator of baffle bolt failures.

The inspectors reviewed Westinghouse Nuclear Safety Advisory Letter NSAL-16-1, which discussed the results of recent baffle-former bolt inspections and provided Westinghouse's recommendations on this issue. The letter described the plants as most susceptible (i.e. Tier 1a) to this degradation as Westinghouse 4-loop reactors limited to those with a down-flow configuration and using Type 347 stainless steel bolts. The inspectors noted the recommendation was to complete UT volumetric examination of the baffle bolts at the next scheduled RFO, and that Entergy had already planned this action for Unit 3. Entergy also planned a long-term corrective action to convert Units 2 and 3 from a "down-flow" baffle configuration to an "up-flow" configuration, which would significantly reduce the load on baffle-former bolts and provide for increased structural margin of the baffle-former assembly. The inspectors determined Entergy's overall response to the issue was commensurate with the safety significance, was timely, and included appropriate compensatory actions. The inspectors concluded that the actions completed and planned were reasonable to address the ongoing aging management of baffle bolts.

4OA3 <u>Follow Up of Events and Notices of Enforcement Discretion</u> (71153 – 5 samples)

.1 <u>Plant Events</u>

a. <u>Inspection Scope</u>

For the plant events listed below, the inspectors reviewed and/or observed plant parameters, reviewed personnel performance, and evaluated performance of mitigating systems. The inspectors communicated the plant events to appropriate regional personnel, and compared the event details with criteria contained in IMC 0309, "Reactive Inspection Decision Basis for Reactors," for consideration of potential reactive inspection activities. As applicable, the inspectors verified that Entergy made appropriate emergency classification assessments and properly reported the event in accordance with 10 CFR 50.72 and 50.73. The inspectors reviewed Entergy's follow-up actions related to the events to assure that Entergy implemented appropriate corrective actions commensurate with their safety significance.

Unit 2

- Turbine trip occurred while synchronizing Unit 2 to the grid on June 15, 2016
- Shutdown required by TS for repairs to a leak on the 21 CCW heat exchanger service water inlet on June 23, 2016

Unit 3

- Rapid power reduction from 100 percent to 45 percent power in response to a loss of both heater drain pumps on May 26, 2016

b. Findings

No findings were identified.

.2 (Closed) Licensee Event Report (LER) 05000247/2015-003-00: Manual Reactor Trip Due to Indications of Multiple Dropped Control Rods Caused by Loss of Control Rod Power Due to a Power Supply Failure

The inspector's reviewed Entergy's actions and reportability criteria associated with LER 05000247/2015-003-00, which was submitted to the NRC on February 3, 2016. On December 5, 2015, control room operators initiated a manual reactor trip after observing indications consistent with multiple dropped control rods following an alarm for the trip of MCC-24/24A. No control rod indication was available due to MCC-24 being faulted and de-energized. The direct cause of the event was the loss of MCC-24 due to an internal fault at the line sides leads at cubicle 2H where they connect to the bucket stab assemblies. The apparent cause was an unanticipated loss of power to the control rod system due to the degradation of the primary control rod PS1 which failed to function when the operating PS2 was lost. The inspectors determined that both the unexpected failure of PS2 and the internal fault in PS1 was not within Entergy's ability to foresee and prevent and was not a performance deficiency. The inspectors reviewed the LER, the associated apparent cause evaluation analysis, and interviewed Entergy staff. This LER is closed.

.3 (Closed) LER 05000247/2016-003-00: TS Prohibited Condition Due to an Inoperable 21 MBFP Discharge Valve for Greater Than the TS Allowed Outage Time

The inspector's reviewed Entergy's actions and reportability criteria associated with LER 05000247/2016-003-00, which was submitted to the NRC on May 6, 2016. On March 7, 2016, during the shutdown to enter 2RFO22, the control switch for the 21 MBFP was tripped from the control room but the MBFP discharge valve BFD-2-21 failed to fully close as designed. The MBFP discharge valve was declared inoperable and TS 3.7.3 Condition C was entered. The MFD-2-21 isolation valve was then manually closed. The direct cause of the failure to close the MBFP discharge valve BFD-2-21 was the motor operated valve's (MOV's) close torque switch contact finger out of position. The apparent cause was that the MOV preventative maintenance procedure lacked the level of detail and direction due to an unrecognized susceptibility associated with the orientation of the close torque switch contact finger bracket opening and spreading of the "U" shape bracket. The downward arrangement made it easier for the torque switch contact finger to move out with spreading of the "U" shaped contact holder. The inspectors reviewed the LER, the associated apparent cause evaluation analysis, and interviewed Entergy staff. This LER is closed.

Introduction. The inspectors identified a Green NCV of 10 CFR 50.65(b)(1) for Entergy's failure to include a function of a safety-related system within the scope of the maintenance rule program. Specifically, Entergy failed to include the feedwater isolation function performed by the MBFP discharge valves, MBFPs, and feedwater regulating valves and feedwater isolation valves which are required to remain functional during and following a design basis event to mitigate the consequences of an accident, within the scope of the maintenance rule monitoring program.

Description. On March 7, 2016, during an RFO, the control switch for the 21 MBFP was positioned to trip and the 21 MBFP tripped as designed, but the MBFP discharge valve BFD-2-21 failed to fully close. Entergy declared MBFP discharge valve BFD-2-21 inoperable and entered TS 3.7.3 Condition C. After troubleshooting, Entergy determined the MOV close torque switch contact finger was out of position within the contact holder. The misalignment allowed the contact finger to move out of the proper position causing the MOV BFD-2-21 to fail to close. This is the same failure mechanism which caused MOV BFD-2-21 to fail to close in 2010 which is referenced in CR-IP2-2010-07013. On December 5, 2015, the 21 MBFP failed to trip and required closure of the steam admission valves to secure it. This failure occurred because of contaminated control oil that prevented the solenoid valves from operating.

The inspectors reviewed Entergy's maintenance rule basis documents and identified the feedwater isolation function was not properly included in the maintenance rule monitoring program as required by 10 CFR 50.65(b)(1). The basis document for the feedwater system did identify the need to monitor the feedwater isolation function under the maintenance rule and stated that it would be monitored as a part of the vapor containment supersystem. However, the basis document for the vapor containment supersystem does not include the feedwater isolation components within the system boundaries. As a result, when component failures occurred which affected the feedwater isolation function, they were not reviewed to determine if they were maintenance rule functional failures; and Entergy was unable to identify that the performance of the main feedwater isolation equipment was not effectively controlled through preventative maintenance. Entergy entered this issue into the CAP as CR-IP2-2016-03963 and initiated actions to include the MBFP discharge valves into the maintenance rule program.

Analysis. The failure to appropriately scope the safety-related feedwater isolation function within the maintenance rule program was a performance deficiency. This finding is more than minor because it is associated with the SSC and barrier performance attribute of the Barrier Integrity cornerstone and affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to properly scope the feedwater isolation function prevented Entergy from identifying that equipment reliability was no longer effectively controlled through preventative maintenance. Additionally, this issue is similar to example 7.d described in IMC 0612, Appendix E, "Examples of Minor Issues," dated August 11, 2009. In accordance with IMC 0609.04, "Initial Characterization of Findings," and Exhibit 2 of IMC 0609, Appendix

A, "The Significance Determination Process for Findings At-Power," issued June 19, 2012, the inspectors determined that the finding was of very low safety significance (Green) because the finding did not represent an actual open pathway in the physical integrity of reactor containment, containment isolation system, and heat removal components. There are redundant methods of feedwater isolation. They include tripping the MBFPs and closing the MBFP discharge valves, closing the main feedwater regulating valves and low flow bypass valves, and closing the main feedwater isolation valves. On both December 5, 2015, and March 7, 2016, the main feedwater regulating valves and isolation valves were functional; so there was no loss of the ability to isolate feedwater to mitigate accident and transient conditions.

This finding does not have a cross-cutting aspect, since the failure to scope this equipment into the maintenance rule program was not recognized when Entergy combined the maintenance rule basis documents for Units 2 and 3 in 2012 and as a result, is not indicative of current licensee performance.

Enforcement. 10 CFR 50.65(b)(1) requires, in part, that the holders of an operating license shall include within the scope of the monitoring program, specified in 10 CFR 50.65(a)(1), SSCs that are relied upon to remain functional during and following design basis events. Contrary to the above, since the combined maintenance rule scoping for Units 2 and 3 in 2012, Entergy failed to include within the scope of the monitoring program specified in 10 CFR 50.65(a)(1), the safety-related MBFP discharge valves. These SSCs are relied upon during and after design basis events to mitigate the consequences of a feedwater line break accident inside containment. Entergy's corrective action included entering this issue into the corrective action program. Because the violation was of very low safety significance (Green) and Entergy entered this issue into their CAP as CR-IP2-2016-03963, this violation is being treated as an NCV, consistent with Section 2.3.2.a of the NRC Enforcement Policy.
(NCV 05000247/2016002-04, Failure to Scope Safety-Related Main Boiler Feedwater Pump Discharge Valves into the Maintenance Rule Program)

4OA5 Other Activities

.1 Groundwater Contamination

a. Inspection Scope

On February 5, 2016, Entergy notified the NRC of a significant increase in groundwater tritium levels measured at three monitoring wells (MWs) (MW-30, MW-31, and MW-32) located near the Unit 2 fuel storage building. These samples were drawn on January 26 to 27, 2016, and analyzed and confirmed on February 2 to 4, 2016. The highest concentration was detected at MW-32, which increased from 12,000 pCi/l on January 11, 2016, to 8,100,000 pCi/l on January 26, 2016, and subsequently up to 14,800,000 pCi/l on February 4, 2016. This increased tritium concentration event was documented by Entergy in CR-IP2-2016-00564 which documents its investigation of this event including a root cause evaluation. The inspectors reviewed Entergy's root cause evaluation for this event during this inspection period as well as recent groundwater monitoring results.

b. Findings and Observations

No findings were identified.

Update to URI 05000247/2016001-07, January 2016 Groundwater Contamination

Entergy continues to conduct weekly, biweekly, and monthly groundwater sampling of MWs at the initial site of groundwater contamination and at downstream wells towards the Hudson River. For the initial three MWs (MW-30, MW-31, and MW-32), the general trend in tritium activity has been downward, with periodic increases seen in some weekly samples. The downstream MWs located in the Unit 2 switchyard (especially MW-55) showed an initial increase in activity up to 117,000 pCi/l, but the activity at that location has plateaued at the end of the reporting period.

Entergy documented its investigation of this event as root cause evaluation for CR-IP2-2016-00564. The inspectors reviewed Entergy's root cause evaluation for this event. Entergy concluded that the source of the groundwater contamination was from the reject water of a temporary reverse osmosis unit used to process water from the refueling water storage tank at Unit 2 in preparation for RFO 2R22. Although this analysis documents a number of issues identified during the operation of the contractor reverse osmosis unit, which is believed to be the source of the groundwater contamination, one of two leakage paths to groundwater have still not been established. The established pathway involves leakage from two cut drain lines located above the floor on the 35-foot elevation of the PAB. Further investigation by Entergy following the conclusion of the Unit 2 RFO 2R22 must be conducted to verify the second pathway to groundwater via the floor of the fuel storage building truck bay.

Entergy's long-term corrective action for reducing tritium levels in the groundwater is the same as previously identified for the March 2014 tritium spike (CR-IP2-2015-03806), the start-up and operation of recovery well (RW)-1. Following installation of equipment and system testing, full operation of the RW system is expected later this year. This system will allow for the collection of tritiated groundwater in the vicinity of Unit 2 to be returned inside the Unit 2 PAB for processing. The NRC will be conducting an inspection in August 2016 to review the testing plan and results of the RW-1 tests. This inspection will include a specialist region-based inspector, and a staff hydrogeologist.

The NRC's continuing assessment of the safety significance of this event focused on validating the safety impact of dose to the public from the release of tritium to the site groundwater, and ultimately to the Hudson River. The NRC verified that Entergy's bounding public dose calculations on the groundwater contamination leak was sufficiently conservative and a maximum worst case scenario would result in a dose of 0.000112 millirem per year, which represents a very small fraction of the allowable dose (liquid effluent dose objective of 3 millirem per year). This low value is due to groundwater at Indian Point not being a source of any drinking water. There are no drinking water wells on the Indian Point site, groundwater flow from the site is to the Hudson River and not to any near site drinking water wells, and the Hudson River has no downstream drinking water intakes as it is brackish water. Pathways to the public are therefore limited to the consumption of fish and river invertebrates. The inspection determined that there is no safety impact to the public as a result of this groundwater contamination event. **(URI 05000247/2016001-07, January 2016 Groundwater Contamination)**

.2 Institute of Nuclear Power Operations (INPO) Report Review

a. Inspection Scope

The inspectors also reviewed the final report for the INPO equipment reliability scram review visit that was conducted to review the scrams that occurred over the past two years, conducted in June 2016. The inspectors reviewed the report to ensure that any issues identified were consistent with NRC perspectives of Entergy performance and to determine if INPO identified any significant safety issues that required further NRC follow-up.

b. Findings

No findings were identified.

4OA6 Meetings, Including Exit

On August 4, 2016, the inspectors presented the inspection results to Mr. Larry Coyle, Site Vice President, and other members of Entergy. Based on additional information provided, the inspectors conducted an updated exit meeting on August 30, 2016 with John Kirkpatrick, Plant Operations General Manager and other members of Entergy. The inspectors verified that no proprietary information was retained by the inspectors or documented in this report.

ATTACHMENT: SUPPLEMENTARY INFORMATION

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

<u>Entergy Personnel</u>
A. Vitale, Site Vice President
J. Kirkpatrick, Plant Operations General Manager
R. Alexander, Unit 2 Shift Manager
R. Andersen, Maintenance Instrumentation and Controls Superintendent
N. Azevedo, Engineering Supervisor
J. Baker, Shift Manager
S. Bianco, Operations Fire Marshal
K. Brooks, Assistant Operations Manager
R. Burroni, Engineering Director
T. Chan, Engineering Supervisor
C. Chapin, Training Superintendent
D. Dewey, Assistant Operations Manager
J. Dignam, Unit 3 Control Room Supervisor
R. Dolansky, Inservice Inspection Program Manager
W. Durr, Outage Control Center Manager
R. Drake, Engineering Supervisor
K. Elliott, Fire Protection Engineer
J. Ferrick, Regulatory and Performance Improvement Director
L. Frink, Radiation Protection Supervisor
D. Gagnon, Security Manager
L. Glander, Emergency Preparedness Manager
D. Gray, Radiological Environmental Manager
J. Johnson, Unit 2 Control Room Supervisor
M. Johnson, Unit 3 Shift Manager
M. Khadabux, Instrumentation and Controls Supervisor
F. Kich, Performance Improvement Manager
M. Lewis, Unit 3 Assistant Operations Manager
N. Lizzo, Training Manager
S. McAllister, Baffle Bolt Replacement Project Manager
M. McCarthy, Unit 3 Control Room Supervisor
B. McCarthy, Operations Manager
F. Mitchell, Radiation Protection Manager
E. Mullek, Maintenance Manager
S. Stevens, Radiation Protection Operations Superintendent
B. Sullivan, Training Superintendent
J. Taylor, Unit 3 Shift Manager
M. Tesoriero, Outage Control Center Manager
M. Troy, Nuclear Oversight Manager
R. Walpole, Regulatory Assurance Manager
A. Zastrow, Assistant Operations Manager

LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

Opened

05000247/2016002-01	URI	CVCS Goal Monitoring Under the Maintenance Rule (Section 1R12)

Opened/Closed

05000286/2016002-02	NCV	Failure to Follow Operability Determination Procedure for Unit 3 Baffle-Former Bolts (Section 1R15)
05000247/2016002-03	NCV	Failure to Maintain Flow Channeling Gates Closed in Accordance with the Containment Procedure (Section 1R20)
05000247/2016002-04	NCV	Failure to Scope Safety-Related Main Boiler Feedwater Pump Discharge Valves into the Maintenance Rule Program (Section 4OA3)

Closed

05000247/2015-003-00	LER	Manual Reactor Trip due to Indications of Multiple Dropped Control Rods Caused by Loss of Control Rod Power Due to a Power Supply Failure (Section 4OA3)
05000247/2016-003-00	LER	Technical Specification Prohibited Condition Due to an Inoperable 21 Main Boiler Feedwater Pump Discharge Valve for Greater Than the TS Allowed Outage Time (Section 4OA3)

Discussed

05000247/2016001-01	URI	Baffle-Former Bolts with Identified Anomalies (Section 1R08)
05000247/2016001-06	URI	Emergency Diesel Generator Automatic Voltage Regulator Failure (Section 1R15)
05000247/2016001-07	URI	January 2016 Groundwater Contamination Section (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

Common Documents Used
Indian Point Unit 2 and Unit 3, UFSARs
Indian Point Unit 2 and Unit 3, Individual Plant Examinations
Indian Point Unit 2 and Unit 3, Individual Plant Examination of External Events
Indian Point Unit 2 and Unit 3, TSs and Bases
Indian Point Unit 2 and Unit 3, Technical Requirements Manuals
Indian Point Unit 2 and Unit 3, Control Room Narrative Logs
Indian Point Unit 2 and Unit 3, Plans of the Day

Section 1R04: Equipment Alignment

Procedures
2-COL-4.2.1, Residual Heat Removal System, Revision 30
2-COL-4.3.1, Spent Fuel Pit Cooling, Revision 10
2-COL-24.1.1, Service Water System, Revision 50
3-COL-EL-005, Diesel Generators, Revision 37
OAP-019, Component Verification and System Status Control, Revision 7
OAP-044, Plant Labeling Program, Revision 3

Condition Reports (CR-IP2)
2016-01311 2016-01505 2016-01761 2016-02330 2016-02428 2016-02470

Condition Reports (CR-IP3)
2016-01382 2016-01810

Drawings
209762, Flow Diagram Service Water System Nuclear Steam Supply Plant, Revision 75
227781, Flow Diagram Auxiliary Coolant System, Revision 22
9321-2720, Auxiliary Coolant System, Sheet 2, Revision 22

Miscellaneous
IP3-DBD-308, CCW System, Revision 3

Section 1R05: Fire Protection

Procedures
EN-MA-133, Control of Scaffolding, Revision 12

Condition Reports (CR-IP2)
2016-04148

Condition Reports (CR-IP3)
2016-01272

Miscellaneous
PFP-203, Containment Building, 95-Foot Elevation (Fire Zone 86A), Revision 15
PFP-204, General Floor Plan, PAB, 15-Foot Elevation, Revision 0
PFP-209, Component Cooling Pump Room, PAB, 68-Foot Elevation, Revision 0
PFP-211, General Floor Plan, PAB, 80-Foot Elevation, Revision 14
PFP-351, 480V Switchgear Room, Revision 15

<u>Section 1R07: Heat Sink Performance</u>

<u>Procedures</u>
0-HTX-405-EDG, EDG Lube Oil and Jacket Water Heat Exchanger Maintenance, Revision 4

<u>Condition Reports (CR-IP3)</u>

2010-02900	2011-03594	2011-03596	2011-03961	2012-02071	2012-03912
2013-02338	2013-02695	2013-03009	2014-00957	2014-01239	2014-03158
2014-03175	2015-00031	2015-00599	2015-02848	2015-05209	2015-05526
2016-00886	2016-00895	2016-00899			

<u>Maintenance Orders/Work Orders</u>
WO 52489888 WO 52626563

<u>Miscellaneous</u>
SEP-SW-IPC-001, Indian Point Energy Center NRC Generic Letter 89-13 Service Water Program, Revision 0

<u>Section 1R08: Inservice Inspection Activities</u>

<u>Procedures</u>
GBRA-104-659, Collection of Protocols for Baffle Bolt Replacement, Revision C
GBRA-175-115, Field Service Procedure for Baffle Bolt Replacement, Revision 3
WDI-STD-088, Underwater Remote Visual Examination of Reactor Vessel Internals, Revision 13
WDI-STD-1073, Ultrasonic Test Procedure for the Inspection of Internal Hex Head Baffle-Former Bolts with Welded Lock Bars, Revision 4

<u>Condition Reports</u> (CR-IP2)
2016-02081

<u>Maintenance Orders/Work Orders</u>
442412-13

<u>Miscellaneous</u>
Indian Point Unit 2 Baffle Bolt Ultrasonic Examination Expanded Analysis Report, dated April 28, 2016
IP2 Reactor Vessel Visual Examination Report, dated May 2006
Loose Parts Inventory Log for Baffle Bolt Replacements, dated May 24, 2016
MRP-227-A, Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (ML120170453)
MRP-228, Materials Reliability Program: Inspection Standard for PWR Internals – 2012 Update, Revision 1
SEP-ISI-IP2-001, IP2 Fourth Ten-Year Interval Inservice Inspection (ISI)/Containment Inservice Inspection (CISI) Program Plan, Revision 2
WDI-PJF-1315504-EPP-001, Indian Point Nuclear Power Plant MRP-227-A Reactor Vessel Internals Examination Program Plan, Revision 0
WDI-PJF-1315505-FSR-001, Indian Point Unit 2 2R22 MRP-227-A Baffle-Former Bolt Ultrasonic Inspections Field Service Report, dated March 29, 2016
WDI-TJ-1100, Technical Justification for the Ultrasonic Inspection of Baffle-Former Bolts for Indian Point Units 2 and 3, Revision 1

Section 1R11: Licensed Operator Requalification Program

Procedures
2-AOP-480V-1, Loss of Normal Power to Any 480V Vital Bus, Revision 8
2-AOP-RCP-1, Reactor Coolant Pump Malfunction, Revision 14
2-AOP-TURB-1, Main Turbine Trip without a Reactor Trip, Revision 5
2-E-0, Reactor Trip or Safety Injection, Revision 7
2-FR-H.1, Response to Loss of Secondary Heat Sink, Revision 11
2-POP-1.2, Reactor Startup, Revision 59
2-SOP-26.4, Turbine Generator Startup, Synchronizing, Voltage Control and Shutdown, Revision 62
3-AOP-480V-1, Loss of Normal Power to Any Safeguards Bus, Revision 7
3-AOP-CVCS-1, Chemical and Volume Control System Malfunction, Revision 8
3-AOP-FW-1, Loss of Feedwater, Revision 8
3-AOP-INST-1, Instrument/Controller Failures, Revision 11
3-E-0, Reactor Trip or Safety Injection, Revision 6
3-E-1, Loss of Reactor or Secondary Coolant, Revision 4
3-FR-C.2, Response to Degraded Core Cooling, Revision 3

Condition Reports (CR-IP2)
2016-03946 2016-04162 2016-04164 2016-04165 2016-04169 2016-04178

Condition Reports (CR-IP3)
2016-01087 2016-01092 2016-01098 2016-01336

Miscellaneous
13SX-LOR-SES026, Licensed Operator Requalification Program Scenario
Emergency Action Level Table, Revision 15.2
LRQ-SES-04, IPEC Simulator Evaluated Scenario, Revision 6

Section 1R12: Maintenance Effectiveness

Procedures
CEP-NDE-0640, Non-Section XI Liquid Penetrant Examination, Revision 9
CEP-WP-WIIR-1, Attachment 5.1, Inprocess Inspections for Installation and Replacement Welds Located Inside the ASME Section XI Boundary, Revision 3
EN-DC-206, Maintenance Rule (a)(1) Process, Revision 3

Condition Reports (CR-IP2)
2010-00864 2013-03130 2014-00162 2014-00185 2014-01144 2014-02184
2015-00278 2016-01260 2016-01430 2016-01500

Condition Reports (CR-IP3)
2012-03836 2013-04758 2015-01396 2015-03404 2015-03653 2015-04053
2015-04162 2015-04184 2015-04539 2015-05316 2015-05384 2015-05729

2016-00098	2016-00653	2016-00723	2016-01189	2016-01227	2016-01274
2016-01313	2016-01531	2016-01536	2016-01543	2016-02432	

Maintenance Orders/Work Orders

WO 00397793	WO 00408019	WO 00414886	WO 00416091
WO 00421841	WO 00429532	WO 00429532	WO 00431497
WO 00446165	WO 00447042	WO 00447966	WO 52602429
WO 52621178			

Miscellaneous
EC 65389, 21 CCW HX Leak Repair: Service Water Inlet Nozzle – Elbow Weld Configuration Change
IPEC Maintenance Rule Basis Document – Chemical and Volume Control System, Revision 0
PQR 913, 134 F42 MN-GTAW ASME IX Welding Procedure Qualification Record, Revision 0
System Health Report, Unit 3, EDG, Q1-2016
Weld Map Number 447966-20-01, Revision 0
WPS 134 F42, MN-GTAW, ASME Section IX Welding Procedure Specification, Revision 0

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures
EN-OP-119, Protected Equipment, Revision 8
IP-SMM-OU-104, Attachment 9.1, Shiftly Outage Shutdown Safety Assessments, Revision 15
IP-SMM-OU-104, Attachment 9.2, Shiftly Outage Shutdown Safety Assessment Guidelines, Revision 15

Condition Reports (CR-IP2)
2016-04141

Condition Reports (CR-IP3)
2016-01545

Miscellaneous
EOOS Risk Assessment Software Tool

Section 1R15: Operability Determinations and Functionality Assessments

Procedures
2-PC-R3-1, Pressurizer Level Transmitters, Revision 10
3-ARP-010, Metal Impact Monitoring System, Page 10, Revision 32
3-SOP-RCS-016, Operation of the Metal Impact Monitoring System, Revision 8
EN-OP-104, Operability Determination Process, Revision 10

Condition Reports (CR-IP2)

2016-2221	2016-2356	2016-2961	2016-3345	2016-3418	2016-3660
2016-3636	2016-3784	2016-3806	2016-3818	2016-4085	

Condition Reports (CR-IP3)
2014-01670 2015-03468

Maintenance Orders/Work Orders

WO 00327574	WO 00425980	WO 52571030

Miscellaneous

EN-LI-100, Attachment 9.1, Change Channel Check Comparison Criteria/2-PT-M100, 2-PT-D001, Revision 0

Section 1R18: Plant Modifications

Drawings

10073E87-001, Indian Point Unit 2 Baffle Bolt Replacement - Core Barrel and Baffle Assembly Elevation, Revision 0

10111D06, Indian Point Unit 2 Baffle Bolt Replacement - Replacement Baffle Bolt .625 and .750, Revision 0

Miscellaneous

EC 64308, IP2 Reactor Vessel Equivalent Replacement Baffle-to-Former Bolt, Revision 0

Process Applicability Determination Form for EC 64308, dated April 21, 2016

WCAP-18136-P, Replacement Type 316 Cold-Worked Baffle-Former Bolt Qualification for Indian Point Unit 2, Revision 0

Section 1R19: Post-Maintenance Testing

Procedures

3-PT-M079B, 32 EDG Functional Test, Revision 52

2-PC-Q109-4, Recalibration of NIS and OT/OP Delta-T Parameters – Channel IV, Revision 44

Condition Reports (CR-IP2)

2016-03961 2016-04266

Condition Reports (CR-IP3)

2016-01189 2016-01199 2016-01218

Maintenance Orders/Work Orders

WO 00414886	WO 00420649	WO 00446094	WO 00447966
WO 52545181	WO 52626563	WO 52626564	WO 52630619
WO 52630620	WO 52658943	WO 00236158	WO 00277374
WO 52571030			

Drawings

5651D72, Logic Diagrams Steam Generator Trip Signals, Revision 7

Miscellaneous

EC 64545, Emergency Temporary Modification to Disconnect 32 EDG Generator Space Heater Adjacent to End Plate on Outboard End of Generator

FIX00091, Pressurizer Level Uncertainty – Indication, Trip Setpoints, and Annunciation Setpoints, Revision 1

E-mail from J. Michetti to G. Newman, dated July 19, 2016, Subject: Westinghouse Report on E9

Section 1R20: Refueling and Other Outage Activities

Procedures
2-POP-1.1, Plant Heatup from Cold Shutdown, Revision 90
2-POP-1.2, Reactor Startup, Revision 59
2-POP-1.3, Plant Startup from Zero to 45 Percent Power, Revision 89
2-POP-3.1, Plant Shutdown from 45 Percent Power, Revision 58
2-POP-3.3, Plant Cooldown, Hot to Cold Shutdown, Revision 81
2-POP-3.4, Secondary Plant Shutdown, Revision 10
2-POP-4.1, Operation at Cold Shutdown, Revision 5
2-POP-4.2, Operation Below 20 Percent Pressurizer Level with Fuel in the Reactor, Revision 8
2-POP-4.3, Operation without Fuel in the Reactor, Revision 1

Condition Reports (CR-IP2-)

2016-04118	2016-04119	2016-04123	2016-03124	2016-04126	2016-04129
2016-04130	2016-04131	2016-04132	2016-04139	2016-04141*	2016-04142*
2016-04144	2016-04145	2016-04146	2016-04148*	2016-04151	2016-04152
2016-04155	2016-04161	2016-04162	2016-04165	2016-04169	

*NRC identified

Maintenance Orders/Work Orders
52681465

Miscellaneous
2R22 Performance Indicator Curves Daily from March 7 to June 14, 2016
Outage Schedules and Plans of the Day from March 7 to June 14, 2016
Westinghouse LTR-PL-16-16, Operability Assessment for Primary Side Loose Parts at Indian Point Unit 2, Revision 0, dated March 27, 2016

Section 1R22: Surveillance Testing

Procedures
0-SOP-LEAKRATE-001, RCS Leakrate Surveillance Evaluation and Leak Identification, Revision 6
2-PT-D001, Control Room Operations Surveillance Requirements, Revision 16
2-PT-M029B, 22 Safety Injection Pump, Revision 20
2-PT-Q013, Inservice Valve Tests, Revision 51
2-PT-Q013-DS040, Valve 887B Inservice Test Data Sheet, Revision 22
3-PT-M079B, 32 EDG Functional Test, Revision 52

Condition Reports (CR-IP2)
2016-03360 2016-03363

Condition Reports (CR-IP3)
2016-01716 2016-01752

Maintenance Orders/Work Orders

WO 00443040	WO 00446385	WO 00446867	WO 52681652-01
WO 52681646-01			

Miscellaneous

EC 64855, EC Reply to Provide Input Regarding MPR Maintenance Bulletin MB-2007-01 for Auto Voltage Regulator Solder Joints

MB-2007-01, Potential for Solder Joint Cracks on Basler SBSR Auto Voltage Regulator Cards and Technical Manual Addendum TM-2007-01, November 5, 2007

Unit 3 RCS Routine Activity Sample, 28-June-16-10006

Section 1EP6: Drill Evaluation

Procedures

IP-EP-120, Emergency Classification, Revision 10

IP-EP-410, Protective Action Recommendations, Revision 11

Section 2RS7: Radiological Environmental Monitoring Program

Procedures

0-CY-1920, REMP Land Use Census, Revision 1

0-CY-1980, Preparation, Placement and Collection of Site Environmental Thermoluminescent Dosimeters, Revision 2

Condition Reports (CR-IP2)

2014-05319	2015-00948	2015-01300	2015-02687	2015-02800	2015-02987
2015-03271	2015-03396	2016-02313			

Condition Reports (CR-IP3)

2016-00514

Miscellaneous

2014 Annual Radiological Environmental Operating Report, Indian Point Nos. 1, 2, and 3

2015 Annual Radiological Environmental Operating Report, Indian Point Nos. 1, 2, and 3

Environmental Dosimetry Company, Annual Quality Assurance Status Report, January to December 2015

Indian Point Energy Center ODCM, Revision 4

June 2015 to May 2016 Meteorological Data Recovery

Met One Instruments, Inc. Certificates of Calibration for Temperature, Wind Direction, and Wind Speed

Teledyne Brown Engineering Environmental Services Annual 2015 Quality Assurance Report

Exelon PowerLabs Certificates of Calibration for Gas Meters

3471875	3482909	3471871	3471867	3482920	3471873
3482910	3482916	3471877	3482914	3482918	3482921
3471881	3471879	3471872	3471869	3471880	3482908

Quality Assurance

Quality Assurance Audit Report QA-2-6-2015-IP-1, Chemistry, Effluents, and Environmental Monitoring Snapshot Self-Assessment, LO-IP3LO-2015-00126, Chemistry-REMP

Section 4OA2: Problem Identification and Resolution

Procedures

EN-DC-204, Maintenance Rule Scope and Basis, Revision 3

EN-DC-204, Maintenance Rule Scope and Basis, Revision 3

EN-DC-207, Maintenance Rule Periodic Assessment, Revision 3

EN-LI-102, Corrective Action Program, Revision 26
EN-LI-104, Self-Assessment and Benchmark Process, Revision 11
EN-LI-110-01, Equipment Failure Evaluation, Revision 0
EN-LI-119, Apparent Cause Evaluation Process, Revision 11
EN-OP-104, Operability Determination Process, Revision 10

Condition Reports (CR-IP2)

2010-07013	2015-04574	2015-05458	2015-05460	2015-05461	2015-05464
2015-05466	2015-05467	2016-01374	2016-02348		

Condition Reports (CR-IP3)

2015-3628	2016-01035	2016-01961

Maintenance Orders/Work Orders
WO 00442412

Apparent Cause Evaluations
IP2-2015-05458

Drawings
504405, Sheet 1 of 2, Replacement Baffle Bolt Location Matrix, Revision 0
504405, Sheet 2 of 2, Replacement Baffle Bolt Location Matrix, Revision 0

Miscellaneous
61265, Temporary Modification Control Form-Install Additional Temporary Spare Power Supply Inside Rod Control Cabinet 2BD to Restore Margin, Revision 0
Appendix A to Facility Operating License DPR-64, Technical Specifications and Basis For The Indian Point 3 Nuclear Generating Station Unit No.3, Through Amendment 260
CN-RIDA-15-43, Indian Point Units 2 and 3 Acceptable Baffle-Former Bolting LOCA and Seismic Analysis, Revision 2
Engineering Change 63938, As-left condition of the baffle-former plate assembly following the replacement of degraded bolts, Revision 0
EPRI TR-112209, Analysis of Baffle Former Bolt Cracking in EDF CPO Plants (PWRMRP-03), dated June 1999
Indian Point Entergy Center (IPEC) Unit 3, Updated Final Safety Analysis Report, dated May 2013
IP-RPT-16-00025, Evaluation of Indian Point Unit 3 Reactor Core Baffle Bolting Following MRP-227-A Inspection Findings at Indian Point Unit 2 during 2R22, Revision 0
LO-IP3LO-2015-72
LTR-PL-16-21, Transmittal of Indian Point Unit 3 Final Engineering Evaluations Supporting Extent of Condition Review, Revision 0
LTR-RIDA-16-103, Indian Point Unit 2 Baffle Bolting Anti-Clustering Pattern and Margin Assessment, Revision 0
LTR-RIDA-16-152, Indian Point Unit 3 Baffle Bolt Leak Before Break Operability Assessment, Revision 0
LTR-RIDA-16-60, Indian Point Unit 2 Baffle Bolting One Cycle Replacement Pattern Summary Letter, Revision 0
MRP-227-A, Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (ML120170453)
Operation Decision Making Issue Action Plan for IP3 Baffle Bolt Monitoring, dated May 19, 2016
WCAP-15029-NP-A, Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions, Revision 0 (ML15222A882)

WCAP-17949-P, Background and Technical Basis Supporting Engineering Flaw Acceptance Criteria for Indian Point Unit 2 Reactor Vessel Internals MRP-227-A Primary and Expansion Components, Revision 1
WCAP-18048-P, Determination of Acceptable Baffle-Former Bolting for Indian Point Units 2 and 3, Revision 0

Section 4OA5: Other Activities

Miscellaneous
INPO Letter, INPO Equipment Reliability Scram Review Visit, May 31, 2016
Root Cause Evaluation for CR-IP2-2016-00564

LIST OF ACRONYMS

10 CFR	Title 10 of the *Code of Federal Regulations*
ADAMS	Agencywide Document Access and Management System
ALARA	as low as is reasonably achievable
AVR	automatic voltage regulator
CAP	corrective action program
CCW	component cooling water
CR	condition report
CVCS	chemical and volume control system
EC	engineering change
ECCS	emergency core cooling system
EDG	emergency diesel generator
GPI	groundwater protection initiative
IASCC	irradiation-assisted stress-corrosion cracking
IMC	Inspection Manual Chapter
INPO	Institute of Nuclear Power Operations
LER	licensee event report
LOCA	loss-of-coolant accident
MBFP	main boiler feedwater pump
MCC	motor control center
MOV	motor operated valve
MRP	materials reliability program
MW	monitoring well
NCV	non-cited violation
NRC	Nuclear Regulatory Commission, U.S.
ODCM	offsite dose calculation manual
OOS	out of service
PAB	primary auxiliary building
PFP	pre-fire plan
RCS	reactor coolant system
REMP	radiological environmental monitoring program
RFO	refueling outage
RW	recovery well
SI	safety injection
SSC	structure, system, and component
TS	technical specification
UFSAR	updated final safety evaluation report
URI	unresolved item
UT	ultrasonic testing
WO	work order