UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
2100 RENAISSANCE BOULEVARD, SUITE 100
KING OF PRUSSIA, PENNSYLVANIA 19406-2713

May 10, 2018

Mr. Anthony J. Vitale
Site Vice President
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
450 Broadway, General Services Building
P.O. Box 249
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING – INTEGRATED INSPECTION REPORT 05000247/2018001 AND 05000286/2018001

Dear Mr. Vitale:

On March 31, 2018, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Indian Point Nuclear Generating, Units 2 and 3. On April 25, 2018, the NRC inspectors discussed the results of this inspection with you and other members of your staff. The results of this inspection are documented in the enclosed report.

The NRC inspectors documented two findings of very low safety significance (Green) in this report. These findings involved violations of NRC requirements The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest the violations or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement; and the NRC Resident Inspector at Indian Point. In addition, if you disagree with a cross-cutting aspect assignment or a finding not associated with a regulatory requirement 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 U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC, 20555-0001; with copies to the Regional Administrator, Region I, and the NRC Resident Inspector at Indian Point.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and the NRC Public Document Room in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Part 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

/RA/

Daniel L. Schroeder, Chief
Reactor Projects Branch 2
Division of Reactor Projects

Docket Numbers: 50-247 and 50-286
License Numbers: DPR-26 and DPR-64

Enclosure:
Inspection Report 05000247/2018001 and
05000286/2018001

cc w/encl: Distribution via ListServ

SUBJECT: INDIAN POINT NUCLEAR GENERATING – INTEGRATED INSPECTION REPORT 05000247/2018001 AND 05000286/2018001 dated May 10, 2018

DISTRIBUTION:
DLew, RA
DCollins, DRA
DPelton, DRP
MGray, DRP
JYerokun, DRS
BWelling, DRS
DSchroeder, DRP
TSetzer, DRP
MRossi, DRP
BHaagensen, DRP, SRI
ASiwy, DRP, RI
JVazquez, DRP, RI
DHochmuth, DRP, AA
JBowen, RI OEDO
RidsNrrPMIndianPoint Resource
RidsNrrDorlLpl1 Resource
ROPReports Resources

DOCUMENT NAME: G:\DRP\BRANCH2\a - Indian Point\IP2&3 Inspection Reports\2018\2018-001\IP2&3 2018.001.final.docx
ADAMS Accession Number: **ML18131A024**

☑ SUNSI Review		☑ Non-Sensitive ☐ Sensitive		☑ Publicly Available ☐ Non-Publicly Available	
OFFICE	RI/DRP	RI/DRP	RI/DRP		
NAME	BHaagensen	TSetzer/DSchroeder for	DSchroeder		
DATE	5/10/18	5/10/18	5/10/18		

OFFICIAL RECORD COPY

U.S. NUCLEAR REGULATORY COMMISSION
Inspection Report

Docket Numbers:	50-247 and 50-286
License Numbers:	DPR-26 and DPR-64
Report Numbers:	05000247/2018001 and 05000286/2018001
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	Indian Point Nuclear Generating, Units 2 and 3
Location:	450 Broadway, General Services Building Buchanan, NY 10511-0249
Inspection Dates:	January 1, 2018, to March 31, 2018
Inspectors:	B. Haagensen, Senior Resident Inspector A. Siwy, Resident Inspector R. Barkley, Acting Resident Inspector S. Elkhiamy, Acting Resident Inspector M. Hardgrove, Acting Resident Inspector C. Safouri, Acting Resident Inspector R. Vadella, Acting Resident Inspector E. Burket, Reactor Inspector R. Clagg, Senior Resident Inspector, Calvert Cliffs T. Dunn, Operations Engineer N. Floyd, Reactor Engineer J. Grieves, State Liaison Officer J. Lilliendahl, Emergency Response Coordinator J. Schoppy, Senior Reactor Inspector S. Wilson, Health Physicist
Approved By:	Daniel L. Schroeder, Chief Reactor Projects Branch 2 Division of Reactor Projects

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring Entergy's performance at Indian Point Nuclear Generating, Units 2 and 3, by conducting the baseline inspections described in this report in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information. NRC-identified and self-revealing findings, violations, and additional items are summarized in the table below.

List of Findings and Violations

Failure to Incorporate Adequate Test Controls for Quarterly Stroke Close Testing of the Steam Supply Valves to Turbine-Driven Auxiliary Feedwater Pump			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green Non-Cited Violation (NCV) 05000247/2018001-01 Closed	H.11 – Human Performance – Challenge the Unknown	71111.22
The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," when Entergy did not assure that surveillance tests required to demonstrate that structures, systems, and components will perform satisfactorily in service are identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Specifically, during quarterly stroke testing of the steam isolation valves to the 22 turbine-driven auxiliary feedwater pump, PCV-1310A and PCV-1310B, Entergy did not ensure that these valves traveled to the closed position as required to verify that the safety function was met.			

Inadequate Procedure for Placing Chemical and Volume Control System Demineralizer In Service			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000286/2018001-02 Closed	P.3 – Problem Identification and Resolution – Resolution	71153
A self-revealing Green NCV of Technical Specification 5.4.1, "Procedures," was identified because Entergy failed to provide adequate guidance in 3-SOP-CVCS-004, "Placing the CVCS Demineralizers In or Out of Service." Specifically, Entergy did not provide adequate procedural direction to prevent exceeding the reactor coolant filter differential pressure while placing the demineralizers in service. As a result, the pressurizer water level technical specification limit was exceeded and the CVCS piping upstream of the filter was over-pressurized resulting in diaphram ruptures on valves CH-305 and CH-352 thereby spreading contamination throughout the Primary Auxiliary Building.			

Additional Tracking Items

Type	Issue Number	Title	Report Section	Status
URI	05000286/2017004-01	Adequacy of Entergy's Modification to Meet Seismic Category I Design Requirements	71111.18	Closed
URI	05000247/2017004-02	Unidentified Reactor Pressure Vessel O-Ring Leakage Caused Degraded Studs	71152	Closed
LER	05000247/2017-001-00	Manual Reactor Trip Due to Decreasing Steam Generator Levels Caused by Main Boiler Feedwater Pump Turbine Low Pressure Governor Valves Failed Closed	71153	Closed
LER	05000286/2017-002-00	Manual Isolation of Chemical and Volume Control System Normal Letdown to Stop a Valve Leak Resulted in an Exceedance of Technical Specification 3.4.9 Condition A Limit for Pressurizer Level	71153	Closed

PLANT STATUS

Unit 2 began the inspection period at 100 percent power. On March 15, 2018, the operators reduced power to 97 percent for equipment testing. On March 18, 2018, the operators commenced a shutdown of Unit 2 for a planned refueling and maintenance outage 2RFO23. The unit reached Mode 5 (cold shutdown) on March 19, 2018. Unit 2 remained in the refueling outage at the end of the inspection period.

Unit 3 began the inspection period at 100 percent power. On February 16, 2018, the reactor tripped due to a turbine trip in response to loss of main generator exciter field. The operators returned Unit 3 to 100 percent power on March 1, 2018. Unit 3 remained at or near 100 percent power for the remainder of the inspection period.

INSPECTION SCOPES

Inspections were conducted using the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection unless otherwise noted. Currently approved IPs with their attached revision histories are located on the public website at http://www.nrc.gov/reading-rm/doc-collections/insp-manual/inspection-procedure/index.html. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter (IMC) 2515, “Light-Water Reactor Inspection Program - Operations Phase.” The inspectors performed plant status activities described in IMC 2515, Appendix D, “Plant Status” and conducted routine reviews using IP 71152, “Problem Identification and Resolution.” The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel to assess Entergy’s performance and compliance with Commission rules and regulations, license conditions, site procedures, and standards.

REACTOR SAFETY

71111.01 - Adverse Weather Protection

Impending Severe Weather (3 Samples)

(1) The inspectors evaluated readiness for impending adverse weather conditions for winter storm Grayson at Units 2 and 3 on January 4, 2018.
(2) The inspectors evaluated readiness for impending adverse weather conditions for winter storm Riley at Units 2 and 3 on March 2, 2018.
(3) The inspectors evaluated readiness for impending adverse weather conditions for winter storm Quinn at Units 2 and 3 on March 7, 2018.

71111.04 - Equipment Alignment

Partial Walkdown (4 Samples)

The inspectors evaluated system configurations during partial walkdowns of the following systems/trains at Unit 2:

(1) Reverse osmosis skid (installed under a licensing amendment) for refueling water storage tank purification on February 8, 2018

(2) 24 service water pump following return from maintenance on February 12, 2018
(3) Fire header for the Turbine Building on February 23, 2018
(4) 22 residual heat removal heat exchanger on March 23, 2018

71111.05A/Q - Fire Protection Annual/Quarterly

Quarterly Inspection (6 Samples)

The inspectors evaluated fire protection program implementation in the following selected areas:

Unit 2

(1) Auxiliary feedwater building pre-fire plans (PFP-259, PFP-260, PFP-261, and PFP-262) on January 20, 2018
(2) Superheater building and electrical tunnel (PFP-156 and PFP-213) on February 6, 2018
(3) Superheater building (PFP-151) on February 23, 2018
(4) Containment building (PFP-201 and PFP-202) on March 19, 2018
(5) Containment building (PFP-203) on March 19, 2018

Unit 3

(6) Turbine building, chemical laboratory, H2 seal oil, and 6.9kV switchgear (PFP-362A, PFP-362B, and PFP-362C) on February 6, 2018

Annual Inspection (1 Sample)

(1) The inspectors evaluated the fire brigade performance at Units 2 and 3 on January 16, 2018.

71111.06 - Flood Protection Measures

Internal Flooding (1 Sample)

The inspectors evaluated internal flooding mitigation protections in:

(1) Emergency safeguards switchgear room at Unit 2 on March 16, 2018

Cables (1 Sample)

The inspectors evaluated cable submergence protection in:

(1) 31 electrical manhole cable at Unit 3 on March 6, 2018

71111.07A - Heat Sink Performance

Heat Sink (1 Sample)

(1) The inspectors reviewed the levels of silting in the Unit 2 service water bay during sonic mapping of the silt levels on March 10, 2018.

71111.11 - Licensed Operator Requalification Program and Licensed Operator Performance

Operator Requalification (2 Samples)

Unit 2

(1) The inspectors observed and evaluated just-in-time training for the reactor shutdown in the simulator prior to Unit 2 Refueling Outage No. 23 (2RFO23) on March 13 and 14, 2018.

Unit 3

(2) The inspectors observed and evaluated licensed operator/management crew training in the simulator as part of the IP 92709 strike contingency plan on January 8, 2018.

Operator Performance (3 Samples)

Unit 2

(1) The inspectors observed and evaluated operator performance in the control room as part of IP 92709 strike contingency preparations on January 18, 2018.
(2) The inspectors observed and evaluated operator performance during the reactor shutdown for entry into 2RFO23 on March 18 and 19, 2018.

Unit 3

(3) The inspectors observed and evaluated operator performance in the control room as part of IP 92709 strike contingency preparations on January 18, 2018.

71111.12 - Maintenance Effectiveness

Routine Maintenance Effectiveness (3 Samples)

The inspectors evaluated the effectiveness of routine maintenance activities associated with the following equipment and/or safety significant functions:

Unit 2

(1) Service water system maintenance on March 9, 2018
(2) Spent fuel pool cooling on March 30, 2018

Unit 3

(3) Chemical and volume control valve maintenance on February 13, 2018

71111.13 - Maintenance Risk Assessments and Emergent Work Control (8 Samples)

The inspectors evaluated the risk assessments for the following planned and emergent work activities:

Unit 2

(1) Yellow risk for planned 22 auxiliary feedwater inservice test and surveillance testing in parallel with reactor coolant system (RCS) Tavg and Delta T instrument channel calibrations on February 1, 2018
(2) Elevated fire risk for the 24 service water pump during maintenance on February 2, 2018
(3) Yellow risk for planned repairs to the 23 charging pump on February 23, 2018
(4) Risk assessment for the 2RFO23 outage on March 18, 2018

Unit 3

(5) Yellow risk for unplanned maintenance on the relief valve for the 32 charging pump on February 8, 2018
(6) Yellow risk for planned maintenance on the 34 control building exhaust fan on February 9, 2018
(7) Yellow risk for calibration of the 33 steam generator atmospheric dump valve pressure controller on February 14, 2018
(8) Yellow risk for 480 volt under voltage and degraded grid protection testing on February 15, 2018

71111.15 - Operability Determinations and Functionality Assessments (8 Samples)

The inspectors evaluated the following operability determinations and functionality assessments:

Unit 2

(1) Pressurizer level loops, L-459 and L-461, deviation greater than 8 percent, on January 8, 2018
(2) 23 emergency diesel generator cylinder temperature deviation on January 11, 2018
(3) Service water to 21 component cooling water heat exchanger cross connect, SWN-33, damaged gearbox on January 20, 2018
(4) 21, 22, 23, and 24 station battery for unsatisfactory overall test acceptance criteria on January 29, 2018
(5) Fire header rupture compensatory actions on February 23, 2018
(6) 23 safety injection pump failed to start during loss of power testing during 2RFO23 on March 29, 2018
(7) Source range nuclear instruments N31 and N32 while in Mode 5 prior to starting refueling operations during 2RFO23 on March 31, 2018

Unit 3

(8) Main feedwater line pipe whip restraint pin on March 21, 2018

71111.18 - Plant Modifications (2 Samples)

The inspectors evaluated the following temporary or permanent modifications:

(1) Engineering Change (EC)-32179 – Reverse osmosis skid installation for refueling water storage tank cleanup (temporary modification) at Unit 2 on March 31, 2018
(2) EC-61654 – Repair degraded weld on service water line 405 (permanent modification) at Unit 3 on January 30, 2018

71111.19 - Post Maintenance Testing (4 Samples)

The inspectors evaluated post maintenance testing for the following maintenance/repair activities:

Unit 2

(1) 24 fan cooler unit service water supply line leak repair on February 15, 2018
(2) 23 charging pump following repairs on February 28, 2018
(3) 22 residual heat removal heat exchanger discharge line vent valve A-98 weld repair on March 30, 2018

Unit 3

(4) Main generator exciter following repairs on February 28, 2018

71111.20 - Refueling and Other Outage Activities (1 Sample, 1 Partial Sample)

(1) The inspectors evaluated the Unit 3 forced outage 3FO18A activities to repair main generator exciter from February 16 to 28, 2018.
(2) The inspectors evaluated the Unit 2 refueling outage, 2RFO23, from March 18 to 31, 2018. This was a partial sample because the outage did not end by the close of the quarter.

71111.22 - Surveillance Testing

The inspectors evaluated the following surveillance tests:

Routine (8 Samples)

(1) 3-PT-M079B, 32 Emergency Diesel Generator Functional Test, at Unit 3 on January 5, 2018
(2) 2-PC-Q109-1, Over-Temperature Delta-T and Overpower Delta-T Recalibration Test, at Unit 2 on February 5, 2018
(3) 3-PT-Q87A, RCS Temperature Calibration Channel 411 Test, at Unit 3 on March 1, 2018
(4) 52690015-01, 23 Fan Cooler Unit Clean and Inspect Air Side Coils Test, at Unit 2 on March 10, 2018
(5) 2-PT-R006, Main Steam Safety Valve Setpoint Determination Test, at Unit 2 on March 16, 2018

(6) 3-PT-Q98A, Steamline Pressure Functional Test – Channel I, at Unit 3 on March 16, 2018
(7) 3-PT-M110, Appendix R Diesel Generator Functional Test, at Unit 3 on March 16, 2018
(8) 2-PT-R014, Safety Injection System Electrical Load and Blackout Test, at Unit 2 during 2RFO23 on March 21, 2018

Inservice (1 Sample)

(1) Inservice testing of steam supply valves, PCV-1310A and PCV-1310B, to turbine-driven auxiliary boiler feedwater pump (ABFP) at Unit 2 on January 11, 2018

71114.06 - Drill Evaluation

Drill/Training Evolution (1 Sample)

(1) The inspectors evaluated a Unit 3 emergency planning drill at the Alternate Emergency Operations Facility on January 31, 2018.

RADIATION SAFETY

71124.01 - Radiological Hazard Assessment and Exposure Controls

Radiological Hazard Assessment (1 Sample)

The inspectors conducted independent radiation measurements during walkdowns of the facility and reviewed:

- the radiological survey program
- any changes to plant operations since the last inspection
- recent plant radiation surveys for radiological work activities
- air sampling and analysis
- continuous air monitor use

Instructions to Workers (1 Sample)

The inspectors reviewed high radiation area work permit controls and use, reviewed electronic alarming dosimeter alarms and setpoints, observed worker briefings on radiological conditions, and observed containers of radioactive materials and assessed whether the containers were labeled and controlled in accordance with requirements.

Contamination and Radioactive Material Control (1 Sample)

The inspectors observed the monitoring of potentially contaminated material leaving the radiological controlled area and inspected the methods and radiation monitoring instrumentation used for control, survey, and release of that material. The inspectors selected several sealed sources from inventory records and assessed whether the sources were accounted for and were tested for loose surface contamination. The inspectors evaluated whether any recent transactions involving nationally tracked sources were reported in accordance with requirements.

Radiological Hazards Control and Work Coverage (1 Sample)

The inspectors evaluated in-plant radiological conditions and performed independent radiation measurements during facility walkdowns and observation of radiological work activities. The inspectors assessed whether posted surveys; radiation work permits; worker radiological briefings and radiation protection job coverage; the use of continuous air monitoring, air sampling and engineering controls; and dosimetry monitoring were consistent with the present conditions. The inspectors examined the control of highly activated or contaminated materials stored within the spent fuel pool and the posting and physical controls for selected high radiation areas, locked high radiation areas, and very high radiation areas.

High Radiation Area and Very High Radiation Area Controls (1 Sample)

The inspectors reviewed the procedures and controls for high radiation areas, very high radiation areas, and radiological transient areas in the plant.

Radiation Worker Performance and Radiation Protection Technician Proficiency (1 Sample)

The inspectors evaluated radiation worker performance with respect to radiation protection work permit requirements. The inspectors evaluated radiation protection technicians in performance of radiation surveys and in providing radiological job coverage.

71124.02 - Occupational As Low As Reasonably Achievable (ALARA) Planning and Controls

Radiological Work Planning (1 Sample)

The inspectors evaluated radiological work planning by reviewing significant work activities to verify that ALARA planning was integrated into work procedures and radiation work permit documents.

Verification of Dose Estimates and Exposure Tracking Systems (1 Sample)

The inspectors reviewed the current annual collective dose estimate, basis methodology, and measures to track, trend, and reduce occupational doses for ongoing work activities. The inspectors evaluated the adjustment of exposure estimates or replanning of work. The inspectors reviewed post-job ALARA evaluations.

Implementation of ALARA and Radiological Controls (1 Sample)

The inspectors reviewed radiological work controls and ALARA practices during the observation of in-plant work activities. The inspectors verified use of shielding, contamination controls, airborne controls, radiation work permit controls, and other work controls were consistent with ALARA plans. The inspectors ensured that work-in-progress reviews were performed in a timely manner and adjustments made to the ALARA estimates when appropriate. The inspectors reviewed the results achieved against the intended ALARA estimates to confirm adequate implementation and oversight of radiological work controls. The inspectors also verified that the ALARA staff were involved with emergent work activities and were revising both dose estimates and ALARA controls in the associated radiation work permits and ALARA plans, as appropriate.

Radiation Worker Performance (1 Sample)

The inspectors observed radiation worker and radiation protection technician performance during radiological work to evaluate worker ALARA performance according to specified work controls and procedures.

71124.03 - In-Plant Airborne Radioactivity Control and Mitigation

Engineering Controls (Partial Sample)

The inspectors evaluated the airborne controls and monitoring. The inspectors observed temporary ventilation system setups and portable airborne radioactivity monitoring systems and verified that Entergy established alarm setpoints for evaluating levels of airborne for both beta and alpha emitting radionuclides.

Use of Respiratory Protection Devices (Partial Sample)

The inspectors evaluated the respiratory protection program. The inspectors reviewed Entergy's ALARA reviews and the storage, selection, and use of respiratory protection devices and verified that air used in supplied air devices meets or exceeds "Grade D" quality. The inspectors also reviewed the qualifications of several individuals to ensure the individuals were qualified to use respiratory protections devices.

Self-Contained Breathing Apparatus for Emergency Use (Partial Sample)

The inspectors evaluated the self-contained breathing apparatus program. The inspectors verified that personnel who are required to use self-contained breathing apparatus's were trained and qualified and that the control rooms were stocked with an adequate variety of respirator face pieces.

OTHER ACTIVITIES – BASELINE

71151 - Performance Indicator Verification

The inspectors verified Entergy's performance indicators submittals listed below for the period from January 1, 2017 through December 31, 2017 (4 Samples).

Unit 2

(1) RCS Specific Activity (BI01)
(2) Reactor Coolant Leak Rate (BI02)

Unit 3

(3) RCS Specific Activity (BI01)
(4) Reactor Coolant Leak Rate (BI02)

71152 - Problem Identification and Resolution

Annual Followup of Selected Issues (1 Sample)

The inspectors reviewed Entergy's implementation of its corrective action program related to the following issue:

(1) Condition report (CR)-IP2-2018-01775, reactor vessel O-ring leak corrective actions, past operability, and generic issue screening at Unit 2.

71153 - Followup of Events and Notices of Enforcement Discretion

Events (1 Sample)

The inspectors evaluated response to the following event:

(1) On February 16, 2018, Unit 3 experienced a reactor trip from 100 percent power when the main generator exciter failed causing a main generator trip and lockout. The reactor tripped as designed and the plant entered Mode 3 with heat removal on the steam dump valves to the condenser. The trip was uncomplicated and the systems worked as designed. The resident staff responded to the control room and verified that the reactor plant was stable. Repairs were completed and Unit 3 was returned to 100 percent power on March 1, 2018.

Licensee Event Reports (LERs) (2 Samples)

The inspectors evaluated the following LERs which can be accessed at https://lersearch.inl.gov/LERSearchCriteria.aspx:

(1) LER 05000286/2017-002-00, Manual Isolation of Chemical and Volume Control System Normal Letdown to Stop a Valve Leak Resulted in an Exceedance of Technical Specification 3.4.9 Condition A Limit for Pressurizer Level, on March 9, 2018. The review for this event is documented in the Inspection Results Section.
(2) LER 05000247/2017-001-00, Manual Reactor Trip Due to Decreasing Steam Generator Levels Caused by Main Boiler Feedwater Pump Turbine Low Pressure Governor Valves Failed Closed, on March 31, 2018. The review for this event is documented in the Inspection Results Section.

OTHER ACTIVITIES – TEMPORARY INSTRUCTIONS, INFREQUENT, AND ABNORMAL

92709 – Strike Contingency

On January 17, 2018, at midnight, the contract between the Utilitiy Workers Union of America and Entergy expired. In preparation, the NRC inspectors verified that Entergy had established a qualification program using the systems approach to training process to ensure that management personnel would be properly qualified to assume union positions should a strike have actually occurred. NRC verified that the management watchstanders had completed the qualification requirements and verified that licensed operators had active licenses. Two crews consisting of non-union personnel were observed in the simulator to verify that the qualification program was reasonably effective and that non-union personnel could safely operate the reactor should a strike have occurred.

After contract negotiations had not been completed by 6:00 p.m. on January 17, 2018, the NRC stationed inspectors in the control rooms prior to the contract deadline to verify a proper relief and turnover if the contract deadline passed at midnight and a stike occurred. A strike did not occur and negotiations continued. NRC continued round the clock strike coverage until both sides came to a verbal agreement on January 24, 2018.

INSPECTION RESULTS

Failure to Incorporate Adequate Test Controls for Quarterly Stroke Close Testing of the Steam Supply Valves to Turbine-Driven Auxiliary Feedwater Pump			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000247/2018001-01 Closed	H.11 – Human Performance – Challenge the Unknown	71111.22

The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," when Entergy did not assure that surveillance testing required to demonstrate that structures, systems, and components will perform satisfactorily in service are identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Specifically, during quarterly stroke testing of the steam isolation valves to the 22 turbine-driven auxiliary feedwater pump (referred to as the auxiliary boiler feedwater pump (ABFP) at Indian Point), PCV-1310A and PCV-1310B, Entergy did not ensure that these valves traveled to the closed position as required to verify that the safety function was met.

Description: On October 30, 2017, the inspectors observed surveillance test 2-PT-Q034A, "22 Auxiliary Feed Pump Steam Supply Valves," where PCV-1310A and PCV-1310B were cycled in the closed and open directions. The inspectors noted that the valve stroke distance in the closed direction appeared to be different between these identical valves. The inspectors also noted that the surveillance test did not ensure that PCV-1310A/B were fully closed. Upon further review, the inspectors noted that the valve baseline opening stroke time for PCV-1310B had increased from approximately eight seconds in November 2016 to a condition where the valve took 35 minutes to open. The valve had to be cycled several times in the open and closed direction before the valve would travel to the fully open position, which was indicative of stem binding from internal degradation.

On December 8, 2017, Unit 2 was shutdown for a forced outage to repair the 21 reactor coolant pump seal. During this two-week shutdown, PCV-1310B was fully disassembled and repaired. The workers noted that there were two loose parts inside the valve body; a loose stem pin and a double disc gate spring that had come loose from their retention points. These loose parts fell out of the valve when it was disassembled. They also identified extensive galling on the valve stem that may have caused the long delay time for opening the valve. These conditions could have potentially caused the valve not to fully seat, especially if one of the loose parts had jammed in the seating surfaces. After repairs, both valves cycled open and closed with the same apparent stem travel distance and within the baseline times for opening and closing.

On January 8, 2018, the inspectors conducted an in-depth review of the repairs to steam supply valves, PCV-1310A and PCV-1310B, to the 22 ABFP. This review was performed due to extensive degradation previously noted during PCV-1310B valve disassembly performed on December 13, 2017, and reaffirmed the cause(s) of the valve's inability to complete a full-stroke in the open direction within a baseline valve stroke time since November 6, 2016.

The auxiliary boiler feedwater system is a safety-related system that consists of two motor-driven ABFPs and a turbine-driven ABFP located in a common room. The two motor-driven ABFPs are not environmentally qualified to be operable if a postulated break of the steam supply line to the turbine-driven ABFP occurred. Therefore, in order to prevent a loss of all ABFPs due to a postulated steam line break in the ABFP room, PCV-1310A and PCV-1310B are in series and have a safety function to isolate the steam supply in order to support the operability of the two safety-related motor-driven (21 and 23) ABFPs. The stroke testing requirements are proceduralized in 2-PT-Q034A, "22 Auxiliary Feed Pump Steam Supply Valves," and 2-PT-Q034B, "PCV-1310A and PCV-1310B Nitrogen Supply," which are performed on a quarterly frequency.

Entergy procedure 2-PT-Q034A directs operators to "VERIFY PCV-1310A [and 1310B] stroked fully closed (local)" and 2-PT-Q034B directs operators to "VERIFY PCV-1310A [and 1310B] stem position agrees with indication" when closed. During the quarterly stroke testing of the valves, the inspectors observed that Entergy had no positive means to ensure that the valve actually reached its full-closed position. Instead, operators performing the test would only observe the valve stem had stopped traveling in the closed direction to determine that the valve has fully closed. Simply observing termination of stem travel without confirmatory information for reaching the closed position is insufficient to validate that the valve has accomplished its safety function to fully close. Foreign material or valve degradation could obstruct or prevent the valve from accomplishing its safety function to fully close, and may not be identified by Entergy's testing practices.

When the adverse condition was assessed, Entergy incorrectly concluded that no degraded or nonconforming condition existed since the surveillance testing procedure indicated that there is no required opening stroke time and PCV-1310B was able to close within the stroke time acceptance criteria. During each performance of the quarterly surveillance test, Entergy did not establish the proper testing criteria for demonstrating that the valve was accomplishing its safety function to fully close, nor did Entergy consider the effects of the steam leak on the stem and retardation in the opening stroke time relevant to the state of internal degradation.

Corrective Actions: On January 10, 2018, Entergy captured the issue in a condition report and initiated a temporary modification package (EC-75845) to install a stroke travel indication mechanism on PCV-1310A and PCV-1310B prior to the next surveillance.

Corrective Action Reference: CR-IP2-2018-00175

<u>Performance Assessment</u>:

Performance Deficiency: Entergy did not assure that surveillance tests that are required to demonstrate that structures, systems, and components will perform satisfactorily in service are identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Specifically, during quarterly stroke testing of the steam supply valves to the 22 ABFP, PCV-1310A and PCV-1310B, Entergy did not verify that the valve actually reached the fully closed position and, therefore, did not adequately test the valves safety function to fully close.

Screening: This finding is more than minor because it was associated with the procedure quality attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the reliability and capability of systems that respond to initiating events to prevent undesirable consequences.

Significance: The inspectors assessed the significance of the finding using IMC 0609.04, “Initial Characterization of Findings,” and IMC 0609, Appendix A, Exhibit 2, “Mitigating Systems Screening Questions.” The inspectors determined that this finding was a deficiency affecting the design or qualification of a mitigating structure, system, or component, where the structure, system, or component maintained its operability or functionality. Specifically, although the valve was internally degraded, Entergy determined that it still was able to satisfy its safety function. Therefore, the inspectors determined the finding to be of very low safety significance (Green).

Cross-Cutting Aspect: This finding has a cross-cutting aspect of Human Performance – Challenge the Unknown, because Entergy did not stop and question whether the valve was accomplishing its safety function to close when there was no direct means of verifying that the valve had actually reached a fully closed position and there was obvious internal degradation in the valve actuator mechanism. In addition, Entergy did not challenge unanticipated test results.

Enforcement:

Violation: 10 CFR Part 50, Appendix B, Criterion XI, “Test Control," requires that a test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. The test program shall include, as appropriate, proof tests prior to installation, preoperational tests, and operational tests during nuclear power plant or fuel reprocessing plant operation, of structures, systems, and components. Test procedures shall include provisions for assuring that all prerequisites for the given test have been met, that adequate test instrumentation is available and used, and that the test is performed under suitable environmental conditions. Test results shall be documented and evaluated to assure that test requirements have been satisfied.

Contrary to the above, Entergy did not ensure that the test program demonstrated that structures, systems, and components will perform satisfactorily in service and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Measures were not established to incorporate adequate testing requirements and associated acceptance limits to demonstrate that the steam supply valves to the turbine-driven ABFP were capable of performing its safety function to fully close in accordance with test procedures and design basis documents.

Disposition: This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy.

Inadequate Procedure for Placing Chemical and Volume Control System Demineralizer In Service			
Cornerstone	Significance/Severity	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000286/2018001-02 Closed	P.3 – Problem Identification and Resolution – Resolution	71153
A self-revealing Green NCV of Technical Specification 5.4.1, “Procedures,” was identified because Entergy failed to provide adequate guidance in 3-SOP-CVCS-004, “Placing the CVCS Demineralizers In or Out of Service.” Specifically, Entergy did not provide adequate procedural direction to prevent exceeding the reactor coolant filter design pressure while placing the demineralizers in service. As a result, the pressurizer water level technical specification limit was exceeded and the CVCS piping upstream of the filter was over-pressurized resulting in diaphragm ruptures of valves CH-305 and CH-352 thereby spreading contamination throughout the Primary Auxiliary Building.			

Description: On June 11, 2017, operators on Unit 3 were performing a swap from the 32 mixed bed demineralizer to the 31 mixed bed demineralizer. A leak developed on the 32 mixed bed demineralizer inlet isolation valve CH-352. Then operators diverted flow from the demineralizer beds to the volume control tank. A leak of approximately 18 gallons per minute developed on the reactor coolant filter inlet isolation valve CH-305. Then operators manually isolated normal letdown. The reactor coolant filter likely exceeded its rated differential pressure of 75 psid since it was found damaged and deformed. Pressurizer water level rose above Technical Specification 3.4.9 limit of 54.3 percent within 8 minutes of normal letdown isolation. The maximum pressurizer water level attained during the event was approximately 61 percent.

Entergy performed emergency maintenance on valves CH-352 and CH-305 to re-torque the body to bonnet fastening bolts. Operators bypassed the reactor coolant filter, removed the 31 mixed bed demineralizer from service, and placed normal letdown in service with no observable valve leakage. The pressurizer water level was restored to less than the technical specification limit of 54.3 percent within 145 minutes after it was exceeded.

Entergy performed an apparent cause analysis and determined the direct cause of the event to be the release of materials from the 31 mixed bed demineralizer which clogged the reactor coolant filter. A similar event occurred at Unit 2 in 2012 when suspended material from a previously used demineralizer and the residual heat removal system were released and clogged the reactor coolant filter. The corrective actions listed in the operating experience form for CR-IP2-2012-00197 included increased monitoring of the reactor coolant filter differential pressure during plant evolutions that can result in rapid change in the filter’s differential pressure. However, the increased monitoring was not documented formally and records of increased monitoring could not be obtained. Entergy determined during the apparent cause analysis that corrective actions from the 2012 event were not adequately tracked or implemented.

Corrective Actions: Entergy staff re-torqued the bolts to the CH-305 and CH-352 valves and revised procedure 3-SOP-CVCS-0004 to provide guidance to operators to increase monitoring of the reactor coolant filter differential pressure while placing the demineralizers in service.

Corrective Action Reference: CR-IP3-2017-03208

Performance Assessment: Performance Deficiency: The inspectors determined that Entergy's failure to provide adequate guidance in procedure 3-SOP-CVCS-004 was a performance deficiency which was within Entergy's ability to foresee and prevent and should have been corrected. Specifically, Entergy did not provide adequate procedural direction to prevent exceeding the reactor coolant filter design pressure while placing the demineralizers in service, which subsequently led to the pressurizer water level technical specification limit being exceeded. Screening: In accordance with IMC 0612, Appendix B, "Issue Screening" this finding is more than minor because it adversely affected the procedure quality attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to provide adequate procedural direction to prevent exceeding the reactor coolant filter design pressure while placing the demineralizers in service led to operators placing the system in a configuration that challenged the availability, reliability, and capability of the pressurizer to respond to RCS pressure transients. Significance: The inspectors assessed the significance of this finding using IMC 0609.04, "Initial Characterization of Findings," and IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power." This finding was determined to be of very low safety significance (Green) because the loss of function was less than the technical specification allowed outage time. Cross-Cutting Aspect: This finding has a cross-cutting aspect in the area of Problem Identification and Resolution – Resolution because Entergy staff did not take effective corrective actions after the Unit 2 2012 event to address issues in a timely manner commensurate with their safety significance.
Enforcement: Violation: Unit 3 Technical Specification 5.4.1 requires that written procedures shall be established, implemented, and maintained as recommended by Appendix A of Regulatory Guide 1.33, Revision 2. Appendix A requires operating procedures for control of the CVCS including letdown and purification. Specifically, procedure 3-SOP-CVCS-0004 did not provide adequate procedural direction to prevent exceeding the reactor coolant filter design pressure while placing the demineralizers in service which subsequently led to the pressurizer water level technical specification limit being exceeded. Contrary to the above, Entergy did not adequately maintain operating procedure 3-SOP-CVCS-004, "Placing the CVCS Demineralizers In or Out of Service," by failing to include specific steps or precaution detail to prevent exceeding the reactor coolant filter design pressure while placing the demineralizers in service. Disposition: This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. The disposition of this finding and associated violation closes LER 05000286/2017-002-00.

Observations Annual Followup of Selected Issues	71152

The inspectors performed an in-depth review of Entergy's cause evaluation and corrective actions associated with CR-IP2-2017-05095 for the unidentified leakage past the reactor pressure vessel outer O-ring at Unit 2. The inspectors interviewed engineering staff and reviewed Entergy's maintenance activities to further assess the cause.

Entergy's causal analysis concluded that the RCS leak rate was small enough that it could not be accurately detected by the plant leakage monitoring systems. Also, Entergy concluded that the outer reactor pressure vessel flange leak-off drain line was clogged and would not support reliable detection of the leak via the temperature indicator installed on the line.

Unit 3 recently demonstrated the capability to reliably detect RCS leakage through both reactor vessel flange O-ring seals on two occasions, in 2015 and 2017. The inspectors reviewed the design of the leak-off system at both Units 2 and 3 and determined that the systems were identical in design and operation. The inspectors noted that the leak rate trends during past O-ring leaks varied significantly between the two units. Specifically, the Unit 3 unidentified leak rate steadily increased from 0.01 gpm and 0.13 gpm over a period of 19 days. The Unit 2 leak rate did not show much day-to-day variation. The Unit 2 weighted average leak rate was 0.044 gpm from September 20 to December 8, 2017, and showed no statistically significant trend. [Note: The September (20-30) average was 0.040 gpm, October (1-31) average was 0.045 gpm, November (1-30) average was 0.043 gpm, and December (1-8) average was 0.054 gpm with an overall standard deviation of 0.01 gpm)].

Based on discussions with Entergy staff, the design of the outer leak-off line is such that a small leak past the outer O-ring may not reach the leak detection drain line because it flows around the head and down the drain line by the force of gravity whereas a leak between the O-rings is forced through the leakoff drain line under RCS pressure into the equipment drain tank inside containment. The inspectors further noted that the final safety analysis report states that leakage from the reactor through its head flange will leak-off between the double O-ring seal and actuate an alarm in the control. There is no similar discussion regarding leak detection for leakage past the outer O-ring.

After discovery of the outer O-ring leak on Unit 2, Entergy staff revised the operational decision making input document for Unit 3, which had also been operating on its outer O-ring, to include additional actions for visually inspecting the vessel flange area for evidence of boric acid deposits. Entergy staff also added measures to ensure the leak-off lines were clear. The inspectors reviewed the revised operational decision making input and determined that the corrective actions were adequate to detect leakage past the reactor pressure vessel flange outer O-ring. These actions included bi-weekly containment entries and visual inspections of the reactor pressure vessel flange area. The inspectors also determined the vessel O-ring leakage detection system worked as designed within the limitations of the system.

In addition, the inspectors reviewed the past operability of the six degraded vessel studs to ensure the reactor pressure vessel bolting system maintained its function to safely secure the vessel head to the reactor vessel. As part of the operability evaluation, the inspectors reviewed Entergy's non-destructive examinations of the studs and the calculation that analyzed the extent of material degradation. Four of the six degraded studs had sufficient metal to pass the stud replacement criteria and were reused. One stud had experienced greater wastage than was allowable and was replaced. Entergy concluded the as-found condition of the five effected studs that were removed was bounded by a design basis stress report, which demonstrated the reactor pressure vessel could operate with this level of degradation. The remaining four degraded studs were replaced during the refueling outage in March. The sixth degraded stud, stud 20, could not be physically removed from the reactor vessel. For restart, stud 20 was conservatively assumed to be completely failed, even though the vessel head tensioning data from December 2017 showed that stud 20 was fully loaded during tensioning and corresponded to a stud with no loss of material. The inspectors reviewed the stress report, including the stud tensioning data, and determined that the as-found reactor pressure vessel bolting system maintained past operability.

A review of the trend of O-ring leaks and associated corrective actions at Units 2 and 3 was documented in Indian Point Integrated Inspection Report 05000247/2017003 and 05000286/2017003 (Agencywide Documents Access and Management System Accession No. ML17303A977) and Indian Point Integrated Inspection Report 05000247/2017004 and 05000286/2017004 (Agencywide Documents Access and Management System Accession No. ML18045A497). The inspectors determined Entergy's overall response to the issue was commensurate with the safety significance, was timely, and included appropriate corrective actions.

However, during the review of maintenance associated with the vessel flange leak-off lines, the inspectors identified that Entergy was not performing the required pressure testing of the leak-off lines, which are designated as code class 2, in accordance with IWC-5220 of the American Society of Mechanical Engineers, Section XI Code. The inspectors also identified that Entergy was not performing the system leakage test alternative to IWC-5220, which was approved by the NRC in American Society of Mechanical Engineers, Code Case N-805. While not formally tested by Entergy staff, the inspectors noted that portions of the leak-off line system at Units 2 and 3 were indirectly pressurized during previous O-ring leaks and demonstrated integrity of the pipe. The issue was determined to be minor because there would be no safety impact on any adjacent equipment as a result of a leak-off line failure. In addition, the inner leakoff drain line had been pressurized to RCS normal operating pressure during the period of time when Unit 2 was operating on the outer O-ring and no evidence of leakage (boric acid deposits) was observed. In accordance with IMC 0612, "Power Reactor Inspection Reports," the above issue constituted a violation of minor significance that is not subject to enforcement action in accordance with the NRC's Enforcement Policy. Entergy subsequently completed all corrective actions to implement Code Case N-805 and entered the inspectors' observations into the corrective action program as CR-IP2-2018-02065.

Licensee Event Report (Closed)	**Manual Reactor Trip Due to Decreasing Steam Generator Levels Caused by Main Boiler Feedwater Pump Turbine Low Pressure Governor Valves Failed Closed LER 05000247/2017-001-00**	**71153**

Description: On June 26, 2017, operations commenced a downpower from 100 percent to 93 percent reactor power to support performance of the main turbine stop and control valve test. With reactor power at 94 percent, the 22 main boiler feedwater pump (MBFP) turbine speed control trouble alarm annunciated coincident with pump speed swings of 800 rpm. Control room operators ceased the downpower and attempted to dampen the speed oscillations using manual control. Entergy determined that the 22 MBFP turbine low pressure governor valves were cycling from full-closed to full-open. Operators were dispatched to take local pneumatic control of the 22 MBFP bypassing the Lovejoy speed control system to stabilize pump speed. Two minutes after establishing local control, the low pressure governor valves went fully closed, resulting in a rapid reduction in 22 MBFP speed and feedwater

delivery to the steam generators. Steam generator levels decreased to 15 percent and were trending downward. At 1531 hours, a manual reactor trip was initiated by the operators prior to receiving an automatic trip on low steam generator level at 9 percent power. Closure Basis: Entergy conducted a causal evaluation and determined that the governor linkage alignment had failed. Entergy revised the applicable maintenance procedure to include additional directions to stake the governor linkage set screws to prevent failure of the linkage. The vendor requirement to stake the governor linkage screws was contained in a proprietary drawing held by the vendor that Entergy could not have accessed and could not be foreseen and prevented. The inspectors did not identify any new issues during the review of the LER. This LER is closed.

Licensee Event Report (Closed)	**Manual Isolation of Chemical and Volume Control System Normal Letdown to Stop a Valve Leak Resulted in an Exceedance of Technical Specification 3.4.9 Condition A Limit for Pressurizer Level** **LER 05000286/2017-002-00**	**71153**
<u>Description</u>: On June 11, 2017, operators on Unit 3 were performing a swap from the 32 mixed bed demineralizer to the 31 mixed bed demineralizer. Leaks developed on two valves on the letdown line system due to an over pressurization event caused by a clogged reactor coolant filter. Ultimately, the technical specification pressurizer water level was exceeded for 145 minutes due to isolating the letdown line system. Entergy established excess letdown, tightened the bolts on the leaking valves to stop the leak, and unisolated the letdown line system allowing the pressurizer water level to return to nominal. Entergy generated CR-IP3-2017-03208 to address this issue. Closure Basis: Entergy revised the operating procedure for placing the mixed bed demineralizer in service to ensure the operators increase monitoring of the reactor coolant filter. A finding for this event is documented in the Inspection Results Section, NCV 05000286/2018001-02, Inadequate Procedure for Placing Chemical and Volume Control System Demineralizer In Service. The inspectors did not identify any new issues during the review of the LER. This LER is closed.		

Unresolved Item (Closed)	**Adequacy of Entergy's Modification to Meet Seismic Category I Design Requirements** **URI 05000286/2017004-01**	**71111.18**
<u>Description</u>: This item was opened to review the adequacy of the service water modification to determine if the component cooling water common discharge header could be declassified from American Society of Mechanical Engineers Class III to seismic Category I, whether the system remained qualified as seismic Category I with the carbon fiber reinforced polymer (CFRP) installed, and whether there was a performance deficiency or violation of regulatory requirements associated with the modification. The inspectors had internal discussions with members of the Office of Nuclear Reactor Regulation staff regarding the design and installation of a CFRP wrap to evaluate whether the modification was adequately designed and installed to ensure seismic Category I design requirements were met. NRC staff independently evaluated the design of the CFRP wrap by performing an allowable stress		

calculation that considered the CFRP wrap alone as providing structural integrity in the vicinity of the degraded weld. The inspectors also reviewed the tensile test procedure of the CFRP material and test results to determine whether the strength test was conducted in accordance with industry standards. The inspectors reviewed the completed work order to evaluate if the CFRP wrap was installed as specified in EC-61654. Corrective Action Reference: CR-IP3-2015-00266 Closure Basis: The inspectors determined that Entergy's modification to service water line 405 was adequately designed to ensure seismic Category I criteria was maintained. The inspectors determined there was not a performance deficiency associated with the modification. This URI is closed.

Unresolved Item (Closed)	**Unidentified Reactor Pressure Vessel O-Ring Leakage Caused Degraded Studs URI 05000247/2017004-02**	**71152**
<u>Description</u>: On December 9, 2017, during a planned shutdown to replace the 21 reactor coolant pump seal, Entergy staff identified an RCS leak from the inner and outer reactor pressure vessel flange O-rings on Unit 2. Boric acid deposits on the reactor pressure vessel head insulation blocks indicated the outer O-ring was leaking. The NRC opened a URI to determine whether there was a performance deficiency associated with the function of the reactor pressure vessel O-ring leakage detection system and whether the as-found Unit 2 degraded reactor pressure vessel bolting system maintained past operability. The NRC conducted an inspection in response to this URI as documented in Section 71152, Problem Identification and Resolution. Corrective Action References: CR-IP2-2017-05095 and CR-IP2-2018-00756 Closure Basis: The inspectors did not identify any findings during the follow-up inspection for this URI. The inspectors determined that the leakage detection system functioned as designed within the limitations of the system and that past operability of the degraded vessel studs was supported by Entergy's analysis. This URI is closed.		

EXIT MEETINGS AND DEBRIEFS

The inspectors confirmed that proprietary information was controlled to protect from public disclosure.

- On April 25, 2018, the inspectors presented the quarterly resident inspector inspection results to Mr. John Ferrick, General Manager of Plant Operations, and other members of the Entergy staff.

DOCUMENTS REVIEWED

Common Documents Used
Indian Point Units 2 and 3, Upated Final Safety Anaysis Report
Indian Point Units 2 and 3, Individual Plant Examination
Indian Point Units 2 and 3, Individual Plant Examination of External Events
Indian Point Units 2 and 3, Technical Specifications and Bases
Indian Point Units 2 and 3, Technical Requirements Manual
Indian Point Units 2 and 3, Control Room Narrative Logs
Indian Point Units 2 and 3, Plan of the Day

71111.01

Procedures
OAP-048, Seasonal Weather Preparation, Revision 19
OAP-008, Severe Weather Preparations, Revision 24

71111.04

Procedures
2-COL-24.1.1, Service Water System, Revision 51
2-COL-24.1.2, Service Water Essential Header Verification, Revision 16

Engineering Evaluations
EC-32179, Reverse Osmosis Skid Installation for Refueling Water Storage Tank Cleanup, Revision 0

Drawings
9321-2722, Service Water System Nuclear Steam Supply Plant – UFSAR Figure No. 9.6-1 (Sheet 1), Revision 130

Miscellaneous
Clearance 2C23-1-FP-048

71111.05A/Q

Procedures
EN-DC-161, Control of Combustibles, Revision 17
EN-TQ-125, Fire Brigade Drills, Revision 5

Condition Reports (CR-IP2-)

2018-00432	2018-00526	2018-00879

Drawings
227552, Fire Protection System Diagram Details, Sheet No. 2, Revision 47
227553, Fire Protection System Diagram Details, Sheet No. 3, Revision 51
9321-4006, Yard Fire Protection Piping, Revision 78

Miscellaneous
PFP-151, Superheater Building, General Floor Plan, 15-Foot Elevation, Revision 12
PFP-156, Superheater Building, General Floor Plan, 72-Foot Elevation, Revision 14

PFP-213, Electrical Tunnel, 33-Foot and 68-Foot Elevations, Revision 0
PFP-255, Turbine Building, General Area, 15-Foot Elevation, Revision 14
PFP-259, Auxiliary Feedwater Building, Auxiliary Feedwater Pump Room, Revision 0
PFP-260, Auxiliary Feedwater Building, Chemical Additive Room, Revision 13
PFP-261, Auxiliary Feedwater Building, 43-Foot and 53-Foot Elevations, Revision 12
PFP-262, Auxiliary Feedwater Building, 64-Foot and 77-Foot Elevations, Revision 0
PFP-362A, Turbine Building, 6.9KV Switchgear Area, 15-Foot Elevation, Revision 12
PFP-362B, Turbine Building, H2 Seal Oil Unit, 15-Foot Elevation, Revision 12
PFP-362C, Turbine Building, Chemical Laboratory Area, 15-Foot Elevation, Revision 15

71111.06

Procedures
0-ELC-418-GEN, Unit 3 Cable Manhole Inspections, Revision 7

Maintenance Orders/Work Orders
WO 51456228 WO 52690015 WO 52692553 WO 52810053

71111.07

Condition Reports (CR-IP2-)
2017-05332

Maintenance Orders/Work Orders
WO 52804609

71111.11

Procedures
2-AOI-4.2.2, Loss of Coolant Accident When RCS Temperature at Least 200 F and Less Than 350 F, Revision 15
2-AOP-480V-1, Loss of Normal Power to Any 480 Bus, Revision 8
2-AOP-RHR-1, Loss of Residual Heat Removal, Revision 9
2-POP-2.1, Operation at Greater Than 45 Percent Power, Revision 63
2-POP-3.1, Plant Shutdown from 45 Percent Power, Revision 60

71111.12

Condition Reports (CR-IP2-)
2017-00726 2017-03768

Maintenance Orders/Work Orders
WO 00477983 WO 00478023

Miscellaneous
Action Request 141087
Action Request 271090
Action Request 288003
Maintenance Rule Basis Document CVCS, Revision 1
Maintenance Rule Basis Document Service Water System, Revision 1

71111.13

Procedures
2-PT-Q034A, 22 Auxiliary Feed Pump Steam Supply Valves, Revision 11
2-PT-Q033C, 23 Charging Pump, Revision 18
3-PT-M62A, 480 Volt Undervoltage/Degraded Grid Protection System Bus 2A and 3A Functional, Revision 13
3-PT-M62B, 480 Volt Undervoltage/Degraded Grid Protection System Bus 5A Functional, Revision 11
3-PT-M62C, 480 Volt Undervoltage/Degraded Grid Protection System Bus 6A Functional, Revision 13
EN-OP-102, Protective and Caution Tagging, Revision 21
EN-OP-119, Protected Equipment Postings, Revision 9
IP-SMM-WM-101, IPEC Site Management Manual, Revision 6

Condition Reports (CR-IP2-)

2018-00566	2018-00902	2018-00955	2018-00968
2018-00979			

Maintenance Orders/Work Orders

WO 00475655	WO 380274-02	WO 495178-02	WO 52675013
WO 52675151	WO 52803336	WO 52803337	WO 52803338

Miscellaneous
2RFO23 Outage Risk Assessment Plan
Clearance 3C20-1, Tagout CVCS-042-32 CHG PMP
Clearance 3C20-1, Tagout HVCB-004-34 CBEXH FN
Clearance 3C20-1, Tagout MS-023-PCV-1136 IC
Daily Plant Status Report, January 30, 2017
Equipment Out of Service on Line Risk Assessment for January 30, 2017
Equipment Out of Service Risk Assessment Software Tool
Unit 2 Equipment Out of Service Monitor

71111.15

Procedures
2-PC-2Y72A, Source Range Neutron Flux N-31 Channel Calibration, Revision 9
2-PC-2Y72B, Source Range Neutron Flux N-31 Channel Calibration, Revision 10
2-PC-R4, Pressurizer Level-CCR, Revision 21
2-PT-A035C, 23 Station Battery Intercell Resistance Check, Revision 8
2-PT-M021A, Emergency Diesel Generator 21 Load Test, Revision 32
2-PT-M021C, Emergency Diesel Generator 23 Load Test, Revision 28
2-PT-Q013-DS230, Valve SWN-33 IST Data Sheet, Revision 22
2-PT-Q013-DS231, Valve SWN-31-1 IST Data Sheet, Revision 22
2-PT-R076A, Station Battery 21 Load Test, Revision 14
2-PT-R076B, Station Battery 22 Load Test, Revision 18
2-PT-R076C, Station Battery 23 Load Test, Revision 14
2-PT-R076-D, Station Battery 24 Load Test, Revision 14
2-PT-R14, Automatic Safety Injection System Electrical Load and Blackout Test, Revision 27
2-PT-W010, Weekly Battery Surveillance Requirement, Revision 9
2-SOP-27.1.6, Instrument Bus, DC Distribution System and PA System Invertor, Revision 43

EN-OP-104, Operability Determination Process, Revision 14
PFM-82, BCT-2000, Battery Test Computer Calibration, Revision 5
SAO-703, Fire Protection Impairment Criteria and Surveillance Administrative, Revision 35

Condition Reports (CR-IP2-) (*initiated in response to inspection)

2012-03773	2016-01991	2017-03377	2017-04488
2017-05117	2017-05401	2018-00008	2018-00013
2018-00022	2018-00131	2018-00149	2018-00323
2018-00330	2018-00409	2018-00677	2018-01470
2018-01478	2018-01519	2018-01521	2018-01522
2018-01524	2018-01593	2018-01935*	2018-01945*

Maintenance Orders/Work Orders

WO 00491904	WO 00492628	WO 52572092	WO 52572560
WO 52579649	WO 52620261	WO 52679115	WO 52706868-01

Calculations
FEX-00204, Station Battery 22 System Calculation, Revision 1

Drawings
209762, Flow Diagram Service Water System Nuclear Steam Supply Plant, Sheet 2 of 2, UFSAR Figure No. 9.6-1 (Sheet 2), Revision 77
227552, Fire Protection System Diagram Details, Sheet No. 2, Revision 47
227553, Fire Protection System Diagram Details, Sheet No. 3, Revision 51
253808, Diesel Generator No. 23 Jacket Water Temperature Loop No. 6600, Revision 0
9321-2028, Flow Diagram Jacket Water to Diesel Generators, Revision 37
9321-4006, Yard Fire Protection Piping, Revision 78
9321-F-2722, Flow Diagram Service Water System Nuclear Steam Supply Plant, Sheet 1 of 2, UFSAR Figure No. 9.6-1 (Sheet 1), Revision 130
D260420, Loop Diagram RCS Pressurizer Level Control Loop No. 459, Revision 3
D260431, Loop Diagram RCS Pressurizer Level Control Loop Nos. 461 and 462, Revision 4
D260433, Loop Diagram RCS Pressurizer Level Control Loop Nos. 459, 460, and 461, Revision 2
IP3V-171-0070, Pressurizer Level Instrument Block Diagram, Revision 6

Miscellaneous
IP2-DBD-226, Design Basis Document for 125V Direct Current Batteries and Distribution System, Revision 1
IP3-RPT-EL-03577, High Energy Line Break Evaluation on Electrical Cables and Penetrations Inside Containment, Revision 1
Operations Fire Watch Request, 2-SAO-18-0816, dated Feburary 19, 2018
Security Fire Watch Checklist, 2-SAO-18-0816, documentation for February 19, 2018, and February 20, 2018
Service Water Basis Document, dated May 4, 2016, Revision 1
Static Pressure Affect (Due to Draining of Tubing) for PT-457 and LT-461 Calculation
TMOD 65236, Temporary Optimize LT-461 Loop

71111.18

Procedures
2-PT-Q033C, 23 Charging Pump, Revision 18

Condition Reports (CR-IP2-)

2018-00902 2018-00955 2018-00968 2018-00979

Maintenance Orders/Work Orders

WO 00404774 WO 380274-02 WO 495178-02

Miscellaneous

EC-32179 – Reverse Osmosis Skid Installation for Refueling Water Storage Tank Cleanup, Revision 0

EC-61654, Repair Degraded Weld on Service Water Line 405, Revision 0

71111.19

Condition Reports (CR-IP2-)

2017-05238 2018-00801

Maintenance Orders/Work Orders

WO 00223237 WO 00491522 WO 00495380-01

Engineering Evaluations

MPR Letter 2288-0057-LTR-001 Subject: Independent Review of IPEC U3 Generator Trip Failure Mode Analysis, Revision 0, dated November 20, 2017

ONLT18_A23, Voltage Regulator Forced Outage Report, GEN SP No 75P0475

ONLT17001A23, Voltage Regulator

Drawings

9321-2561, Containment Building Residual Heat Exchanger Piping Plans and Sections (A.C.S.), Revision 23

9321-2720, Flow Diagram Auxiliary Coolant System Upated Final Safety Analysis Report, Figure No. 9.3-1 (Sheet 2), Revision 93

Miscellaneous

EC-75309, Install Temporary Clamp to Facilitate Weld Repair at Valve A-98, Revision 2

EC-75312, Engineering to Evaluate the Repair of Weld Per Code Case N-666, Revision 0

EC-75993, Engineering to Provide Guidance on How to Make Weld Repair on Weld Upstream on SWN-43-4, Revision 0

NYPA 439-100000359, Type WMA Voltage Regulator and Excitation Switchgear for Brushless Exciter System, dated September 1969

71111.20

Procedures

3-GNR-042-MTG, Main Generator Exciter Disassembly, Inspection, Repair, and Reassembly without Removal, Revision 0

3-POP-1.3, Plant Startup from Zero to 45 Percent Power, Revision 91

3-SOP-TG-4, Turbine Generator Operation, Revision 58

Condition Reports (CR-IP3-)

2018-00539 2018-00540 2018-00541 2018-00542

2018-00543	2018-00544	2018-00545	2018-00546
2018-00549	2018-00551		

71111.22

Procedures
0-MTR-401-ELC, Inspection/Repair and Overhaul of Motors – 480 Volt and Lower, Revision 34
0-SOP-CB-001, Vapor Containment Entry in Modes 1, 2, 3, and 4, Revision 0
2-PC-Q109-1, Recalibration of NIS and OT/OP Delta T Parameters – Channel I, Revision 56
2-PT-M110, Appendix R Diesel Generator Functional Test, Revision 9
2-PT-Q034A, 22 Auxiliary Feed Pump Steam Supply Valves, Revision 11
2-PT-Q034B, PCV-1310A and PCV-1310B Nitrogen Supply, Revision 8
2-PT-R006, Main Steam Safety Valve Setpoint Determinaton, Revision 31
2-PT-R14, Automatic Safety Injection System Electrical Load and Blackout Test, Revision 27
2-SOP-27.6, Unit 2 Appendix R Diesel Generator Operation, Revision 17
3-PT-M079B, 32 EDG Functional Test, Revision 56
3-PT-Q101, Main Steam Valves PCV-1310A, PCV-1310B, PCV-1139, PCV-1310A, and PCV-1310B Nitrogen Supply Stroke Test, Revision 18
3-PT-Q98A, Steam Line Pressure Functional Test – Channel I, Revision 7
3-PT-Q98B, Steam Line Pressure Functional Test – Channel II, Revision 8

Condition Reports (CR-IP2-) (*initiated in response to inspection)

2016-01204	2016-01905	2016-06651	2017-05183
2017-05470	2018-00175*	2018-01593	

Condition Reports (CR-IP3-)

2015-00898	2015-02128	2018-00759

Maintenance Orders/Work Orders

WO 51456228	WO 52683711	WO 52690015	WO 52692553
WO 52706868-01	WO 52708297	WO 52798297-01	WO 52800135
WO 52810053			

Drawings
9321-F-2017, Flow Diagram Main Steam, Revision 84
D251232, Auxiliary Feedwater Steam Supply, Auxiliary Feedwater Pump No. 22, Loop No's. 1118, 1126, 1176, 1310, and 1311, Revision 11

Miscellaneous
EC-39376, Modification to IP2 Main Steam Valves to Install Sacrificial Bronze Wear-Sleeves in the Inner Diameter of the Spring-Washers (Upper and Lower) and Adjusting Bolts (Base EC), Revision 0
EC-75845, Installation of Stroke Indication Mechanism for Valves MS-PCV-1310A and MS-PCV-1310B
IP2-AFW DBD, Auxiliary Feedwater System Design Basis Document, Revision 2
SEP-IP2-IST-2, Indian Point 2 Fifth Ten-Year Interval Inservice Testing Program Plan, Revision 2
SEP-IP2-IST-2, Indian Point 2 Fifth Ten-Year Interval Inservice Testing Program Plan, Revision 3

71114.06

Condition Reports (CR-IP3-)
2018-00393

Miscellaneous
Indian Point Energy Center Emergency Planning Training Drill Controller/Evaluator Manual, dated January 31, 2018
IPEC ERO Team "B" Site Controller Report, dated January 31, 2018

71124.01

Procedures
0-RP-RWP-407, Refueling Support, Revision 8
EN-RP-101, Access Control for Radiologically Controlled Areas, Revision 14
EN-RP-105, Radiation Work Permits, Revision 18
EN-RP-106, Radiological Survey Documentation, Revision 7
EN-RP-106-01, Radiological Survey Guidelines, Revision 4
EN-RP-108, Radiation Protection Posting, Revision 20
EN-RP-122, Alpha Monitoring, Revision 9
EN-RP-123, Radiological Controls for Highly Radioactive Objects, Revision 1
EN-RP-202, Personnel Monitoring, Revision 13
EN-RP-203, Dose Assessment, Revision 10
EN-RW-104, 10 CFR Part 61, Waste Stream Sample Screening and Evaluation, Revision 13

Radiation Work Permits

20182501	20182502	20182504	20182518	20182520	20182521
20182531	20182533	20182543			

Engineering Evaluations
Engineering Report No. IP-RPT-16-00047, 2016 Groundwater Project Report, Units 2 and 3 Floor Drains Flow Verification and Current Condition, Hennigan Engineering Company, Inc., updated January 9, 2018

Drawings (Floor Drains)
9321-F-2649, Holdup Tank Pit and Trench Floor and Hub Drains Plans, Revision 10
9321-F-2696, Primary Auxiliary Building Service Air, City Water, and Drains Plans, Revision 35
9321-F-26493, Holdup Tank Pit and Pipe Trench Floor and Hub Drains Plans and Sections, Revision 6
9321-F-26973, Primary Auxiliary Building Floor and Hub Drains at Elevation 55 Foot - 0 Inches and Elevation 73 Foot – 0 Inches Plans and Sections, Revision 9

Dose Rate and Contamination Surveys
Survey Numbers:

17-2-2377	17-2-2379	17-2-2380	17-2-2381	17-2-2383	18-2-0625
18-2-0659	18-2-0828	18-2-0864	18-2-0874	18-2-0910	18-2-1022

Air Sample Surveys (EN-RP-131):

18-2-0213	18-2-0289	18-2-0320	18-2-0344	18-2-0349	18-2-0814
18-2-0900					

71124.02

Procedures
0-RP-ALA-203, Outage Cumulative Radiation Exposure Estimates and Goals, Revision 0
EN-RP-110, ALARA Program, Revision 14
EN-RP-110-02, Elemental Cobalt Sampling, Revision 0
EN-RP-110-03, Collective Radiation Exposure Reduction Guidelines, Revision 4
EN-RP-110-04, Radiation Protection Risk Assessment Process, Revision 7
EN-RP-110-06, Outage Dose Estimating and Tracking, Revision 1
IP-SMM, RP-101, Temporary Shielding Program, Revision 2
RP-STD-17, Satellite Radiation Protection Key Issuance and High Radiation Area, Locked High Radiation Area, and Very High Radiation Area Boundary Verifications, Revision 13

ALARA Reviews and Reports
Indian Point Unit 3, 3RFO19 Outage Summary Report
EN-RP-105 "Radiological Work Permits," Revision 18, ALARA Plans for 2RFO23:
- 20182501, Radiation Protection Support
- 20182504, Routine Waste Management
- 20182515, In-Vessel Internal Inspection
- 20182520, Valve Work Inside Radiological Controlled Area
- 20182521, Reactor Disassembly/Reassembly
- 20182533, Reactor Coolant Pump Work
- 20182543, Steam Generator Primary Side Work

71124.03

Procedures
EN-RP-503, Selection, Issue and Use of Respiratory Protection Equipment, Revision 7
EN-RP-502, Inspection and Maintenance of Respiratory Protection Equipment, Revision 10
EN-RP-404, Operation and Maintenance of High-Efficiency Particulate Air Vacuum Cleaners and High-Efficiency Particulate Air Ventilation Units, Revision 8
EN-RP-504, Breathing Air, Revision 4
EN-RP-505, PortaCount Respirator Fit Testing, Revision 7
0-RPRSP-102, Decontamination, Sanitizing, and Cleaning of Respiratory Protection and Other Related Equipment, Revision 0

Evaluations and Tests
Airgas Certificate of Analysis, Grade of Product: Breathing Air, dated March 7, 2018
AAA Emergency Supply Co. Inc., Air/Gas Quality Report and Certification, Supply Air Analysis for Filling of SCBA Units, dated March 16, 2018
TEDE-ALARA Evaluation, Radiation Work Permit No. 20183022, Unit 3 Non-Outage Filter Change-Outs, Revision 0
TEDE-ALARA Evaluation, Radiation Work Permit No. 20182522, Seal Table Work, Revision 0

71151

Procedures
0-CY-2765, Coolant Activity Limits – Dose Equivalent Iodine/Xenon, Revision 5
0-SOP-LEAKRATE, RCS Leakrate Surveillance, Evaluation and Leak Identification, Revision 6

71152

Condition Reports (CR-IP2-)

2016-06882	2017-05277	2017-05356	2018-01775

Miscellaneous

IP-CALC-17-00056, Reactor Vessel Stud Area Reduction Due to O-Ring Leakge, Revision 0

R-4147-00-1, Reactor Vessel Tensioning Optimization Stress Report Indian Point Units 2 and 3, Revision 0, dated August 2002