UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE RD. SUITE 210
LISLE, ILLINOIS 60532-4352

January 22, 2018

Mr. Dean Curtland
Director of Site Operations
Duane Arnold Energy Center
3277 DAEC Road
Palo, IA 52324-9785

SUBJECT: DUANE ARNOLD ENERGY CENTER—NRC INTEGRATED INSPECTION REPORT 05000331/2017004; 07200032/2017001; AND EMERGENCY PREPAREDNESS ANNUAL INSPECTION REPORT 05000331/2017501

Dear Mr. Curtland:

On December 31, 2017, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Duane Arnold Energy Center. On January 4, 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 also completed its annual inspection of the Emergency Preparedness Program. This inspection began on January 1, 2017, and issuance of this letter closes Inspection Report 05000331/2017501.

Based on the results of this inspection, the NRC identified one issue that was evaluated under the risk significance determination process as having very low safety significance (Green). The inspectors also evaluated one item under the traditional enforcement process. The NRC determined that a violation is associated with each of these issues. Because the licensee initiated condition reports to address these issues, these violations are being treated as Non-Cited Violations (NCVs), consistent with Section 2.3.2 of the Enforcement Policy. These NCVs are described in the subject inspection report.

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 III; the Director, Office of Enforcement, and the NRC Resident Inspector at the Duane Arnold Energy Center.

If you disagree with the 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 III; and the NRC Resident Inspector at the Duane Arnold Energy Center.

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 at the NRC Public Document Room in accordance with 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

/RA/

Karla Stoedter, Chief
Branch 1
Division of Reactor Projects

Docket No. 50–331; 72–032
License No. DPR–49

Enclosure:
IR 05000331/2017004; 07200032/2017001;
05000331/2017501

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Letter to Dean Curtland from Karla Stoedter dated January 22, 2018

SUBJECT: DUANE ARNOLD ENERGY CENTER—NRC INTEGRATED INSPECTION REPORT 05000331/2017004; 07200032/2017001; AND EMERGENCY PREPAREDNESS ANNUAL INSPECTION REPORT 05000331/2017501

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50–331; 72–032
License No:	DPR–49
Report No:	05000331/2017004; 07200032/2017001; 05000331/2017501
Licensee:	NextEra Energy Duane Arnold, LLC
Facility:	Duane Arnold Energy Center
Location:	Palo, IA
Dates:	October 1 through December 31, 2017
Inspectors:	C. Norton, Senior Resident Inspector J. Steffes, Resident Inspector Dr. N. Fields, Health Physicist/ISFSI Inspector G. Hansen, Senior Emergency Preparedness Inspector J. Cassidy, Senior Health Physicist V. Myers, Senior Health Physicist R. Baker, Senior Operations Engineer D. Reeser, Operations Engineer
Approved by:	K. Stoedter, Chief Branch 1 Division of Reactor Projects

Enclosure

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SUMMARY

Inspection Report 05000331/2017004, 07200032/2017001; 10/01/2017 – 12/31/2017; 05000331/2017501; 01/01/2017 – 12/31/2017; Duane Arnold Energy Center; Annual Follow-up of Selected Issues and Operation of an Independent Spent Fuel Storage Installation at Operating Plants.

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. This report documents a U.S. Nuclear Regulatory Commission (NRC) identified Severity Level IV Non-Cited Violation (NCV) of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 72.212(b)(6). In addition, one Green, self-revealing finding and associated NCV of the NRC requirements of 10 CFR 50, Appendix B, Criterion V was documented by the resident inspectors. The significance of inspection 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 NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated November 1, 2016. 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: Initiating Events

Green. The inspectors documented a self-revealed finding of very low safety significance and an associated NCV of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," due to operations personnel failing to control reactor vessel water level in accordance with Integrated Plant Operating Procedure 2, Startup, Revision 160. Specifically, during a reactor startup, while at 55 percent reactor power with only one reactor feed pump running, the operating crew failed to maintain reactor water level within the procedurally required level band which resulted in a recirculation pump runback to 45 percent speed and an unplanned reactor power decrease from 55 to 43 percent. The licensee responded to the transient and verified that reactor power stabilized at 43 percent without complications, conducted a human performance review, and entered this issue into their corrective action program (CAP) as condition report (CR) 02233094.

The performance deficiency was determined to be more than minor because it was associated with the human performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during power operations. Specifically, the failure to control reactor water level within the procedurally specified water level band resulted in an unplanned recirculation pump runback and a decrease in reactor power from 55 to 43 percent. The finding was determined to be of very low safety significance because the finding did not cause a reactor trip. The inspectors determined this finding affected the cross-cutting area of human performance in the aspect of teamwork, where individuals and work groups communicate and coordinate their activities within and across organizational boundaries to ensure nuclear safety is maintained. Specifically, a reactor operator dialed down the reactor water level control set point without notifying the control room supervisor, briefing the evolution, or obtaining a peer check. [H.4] (Section 4OA2.4)

Miscellaneous

Severity Level IV: The inspectors identified a Severity Level IV NCV of 10 CFR 72.212(b)(6), "Conditions of General License Issued under 72.210," for the failure of the licensee as of June 9, 2003, to determine whether or not reactor site parameters were enveloped by the cask design bases as considered in the Updated Final Safety Analysis Report (UFSAR). Specifically, the licensee failed to evaluate site-specific fire and explosion hazards that were allowed to be near the dry cask storage systems under its Administrative Control Procedure (ACP) 1412.2, "Control of Combustibles," Revision 48. The licensee documented this issue in its CAP as CR 02228514 and CR 02228558 and took timely corrective actions.

The inspectors determined that the violation was of more than minor significance using IMC 0612, "Power Reactor Inspection Reports", Appendix E, "Examples of Minor Issues." Example 4k is applicable to this issue in that the lack of evaluation showing that the quantity of combustible and flammable liquids stored near the dry cask storage system were bounded by the design basis in the UFSAR allowed for a credible unanalyzed fire and explosion scenario that could affect the important-to-safety dry cask storage system. The violation screened as a Severity Level IV NCV. Cross-cutting aspects are not assigned to traditional enforcement violations. (Section 4OA5.1)

REPORT DETAILS

Summary of Plant Status

Duane Arnold Energy Center (DAEC) operated at full power at the beginning of the inspection period. On October 23, 2017, the licensee commenced a reactor shutdown to repair a packing leak on a recirculation system sample line isolation valve. The licensee commenced a reactor start up on October 25, 2017. During the startup, the licensee experienced a reactor recirculation pump runback and a 12 percent reduction in reactor power when reactor vessel water level dropped to less than 186 inches. The runback is discussed in further detail in Section 4OA2.4 of this report. Operations personnel reestablished steady state full power operations on October 30, 2017. On December 9, 2017, the licensee lowered reactor power to 72 percent to perform a control rod sequence exchange and returned the plant to full power operations. The plant remained at full power for the remainder of the inspection period with the exception of brief down-power maneuvers to accomplish rod pattern adjustments or planned surveillance test activities.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R01 Adverse Weather Protection (71111.01)

.1 Winter Seasonal Readiness Preparations

a. Inspection Scope

The inspectors conducted a review of the licensee's preparations for winter conditions to verify that the plant's design features and implementation of procedures were sufficient to protect mitigating systems from the effects of adverse weather. Documentation for selected risk-significant systems was reviewed to ensure that these systems would remain functional when challenged by inclement weather. During the inspection, the inspectors focused on plant specific design features and the licensee's procedures used to mitigate or respond to adverse weather conditions. Additionally, the inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant specific procedures. Cold weather protection, such as heat tracing and area heaters, was verified to be in operation where applicable. The inspectors also reviewed corrective action program (CAP) items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their CAP in accordance with station corrective action procedures. Documents reviewed are listed in the Attachment to this report. The inspectors' reviews focused specifically on the following plant system due to its risk significance or susceptibility to cold weather issues:

- reactor building intake heating coils.

This activity constituted one winter seasonal readiness preparations sample as defined in Inspection Procedure (IP) 71111.01–05.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

- direct current systems batteries; and
- 'B' residual heat removal system.

The inspectors selected these systems based on their risk significance relative to the Reactor Safety Cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, UFSAR, Technical Specification (TS) requirements, outstanding 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 rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

These activities constituted two partial system walkdown samples as defined in IP 71111.04–05.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

The inspectors conducted fire protection walkdowns which were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- low level rad processing and storage facility el. 757' (Zones 21–F, 21–G, 21–H, 21–I, 21–J, 21–K, 21–L, 21–M, 21–O and 21–S) and radioactive release (Zones 21–G, 21–M and 21–S);

- control building el. 772';
- reactor building el. 757' (Zones 2–C, 2–E, and 2–F); and
- turbine building el. 780'.

The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and implemented adequate compensatory measures for out-of-service, degraded or inoperable fire protection equipment, systems, or features in accordance with the licensee's fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events with later additional insights, their potential to impact equipment which could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Using the documents listed in the Attachment to this report, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's CAP. Documents reviewed are listed in the Attachment to this report.

These activities constituted four quarterly fire protection inspection samples as defined in IP 71111.05–05.

b. Findings

No findings were identified.

1R06 Flooding (71111.06)

.1 Internal Flooding

a. Inspection Scope

The inspectors reviewed selected risk important plant design features and licensee procedures intended to protect the plant and its safety-related equipment from internal flooding events. The inspectors reviewed flood analyses and design documents, including the UFSAR, engineering calculations, and abnormal operating procedures to identify licensee commitments. The specific documents reviewed are listed in the Attachment to this report. In addition, the inspectors reviewed licensee drawings to identify areas and equipment that may be affected by internal flooding caused by the failure or misalignment of nearby sources of water, such as the fire suppression or the circulating water systems. The inspectors also reviewed the licensee's corrective action documents with respect to past flood-related items identified in the corrective action program to verify the adequacy of the corrective actions. The inspectors performed a walkdown of the following plant areas to assess the adequacy of watertight doors and verify drains and sumps were clear of debris and were operable, and that the licensee complied with its commitments:

- reactor building north east corner room, south east corner room, north west corner room, and south west corner room.

Documents reviewed during this inspection are listed in the Attachment to this report.

This inspection constituted one internal flooding sample as defined in IP 71111.06–05.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review of Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

On November 14, 2017, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification training. The inspectors verified that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and that training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of abnormal and emergency procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly licensed operator requalification program simulator sample as defined in IP 71111.11–05.

b. Findings

No findings were identified.

.2 Resident Inspector Quarterly Observation During Periods of Heightened Activity or Risk (71111.11Q)

a. Inspection Scope

On October 22, 2017, and October 25, 2017, the inspectors observed control room operators shut down and start up the reactor to complete a forced outage in response to increased drywell unidentified leakage. These activities required heightened awareness or were related to increased risk. The inspectors evaluated the following areas:

- licensed operator performance;

- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms (if applicable);
- correct use and implementation of procedures;
- control board (or equipment) manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications (if applicable).

The performance in these areas was compared to pre-established operator action expectations, procedural compliance and task completion requirements. Documents reviewed are listed in the Attachment to this report.

This activity constituted one quarterly licensed operator heightened activity/risk sample as defined in IP 71111.11–05.

b. Findings

No findings were identified.

.3 Biennial Written and Annual Operating Test Results (71111.11A)

a. Inspection Scope

The inspectors reviewed the overall pass/fail results of the Annual Operating Tests and the Biennial Written Examinations required by 10 CFR Part 55.59(a), that were administered by the licensee from November 6, 2017, through December 15, 2017. The results were compared to the thresholds established in Inspection Manual Chapter (IMC) 0609, Appendix I, "Licensed Operator Requalification Significance Determination Process," to assess the overall adequacy of the licensee's Licensed Operator Requalification Training (LORT) Program to meet the requirements of 10 CFR 55.59. (02.02)

This inspection constituted one annual licensed operator requalification examination results sample as defined in IP 71111.11–05.

b. Findings

No findings were identified.

.4 Biennial Review (71111.11B)

a. Inspection Scope

The following inspection activities were conducted during the weeks of December 4 and December 11, 2017, to assess: (1) the effectiveness and adequacy of the facility licensee's implementation and maintenance of its systems approach to training (SAT) based LORT Program put into effect to satisfy the requirements of 10 CFR 55.59; (2) conformance with the requirements of 10 CFR 55.46 for use of a plant referenced simulator to conduct operator licensing examinations and for satisfying experience requirements; and (3) conformance with the operator license conditions specified in 10 CFR 55.53. The documents reviewed are listed in the Attachment to this report.

- Licensee Requalification Examinations (10 CFR 55.59(c); SAT element 4 as defined in 10 CFR 55.4): The inspectors reviewed the licensee's program for development and administration of the LORT biennial written examination and annual operating tests to assess the licensee's ability to develop and administer examinations that are acceptable for meeting the requirements of 10 CFR 55.59(a).

  - The inspectors conducted a detailed review of one biennial requalification written examination set (Reactor Operator and Senior Reactor Operator) to assess content, level of difficulty, and quality of the written examination materials. (02.03)
  - The inspectors conducted a detailed review of ten Job Performance Measures (JPMs) and four simulator scenarios to assess content, level of difficulty, and quality of the operating test materials. (02.04)
  - The inspectors observed the administration of the annual operating test and biennial written examination to assess the licensee's effectiveness in conducting the examinations, including the conduct of pre-examination briefings, evaluations of individual operator and crew performance, and post-examination analysis. The inspectors evaluated the performance of one crew in parallel with the facility evaluators during two dynamic simulator scenarios, and evaluated various licensed crew members concurrently with facility evaluators during the administration of several JPMs. (02.05)
  - The inspectors assessed the adequacy and effectiveness of the remedial training conducted since the last requalification examinations and the training planned for the current examination cycle to ensure that they addressed weaknesses in licensed operator or crew performance identified during training and plant operations. The inspectors reviewed remedial training procedures and individual remedial training plans. (02.07)

- Conformance with Examination Security Requirements (10 CFR 55.49): The inspectors conducted an assessment of the licensee's processes related to examination physical security and integrity (e.g., predictability and bias) to verify compliance with 10 CFR 55.49, "Integrity of Examinations and Tests." The inspectors reviewed the facility licensee's examination security procedure and observed the implementation of physical security controls (e.g., access restrictions and simulator I/O controls) and integrity measures (e.g., security agreements, sampling criteria, bank use, and test item repetition) throughout the inspection period. (02.06)
- Conformance with Operator License Conditions (10 CFR 55.53): The inspectors reviewed the facility licensee's program for maintaining active operator licenses and to assess compliance with 10 CFR 55.53(e) and (f). The inspectors reviewed the procedural guidance and the process for tracking on-shift hours for licensed operators and which control room positions were granted watch-standing credit for maintaining active operator licenses. Additionally, medical records for seven licensed operators were reviewed for compliance with 10 CFR 55.53(I). (02.08)
- Conformance with Simulator Requirements Specified in 10 CFR 55.46: The inspectors assessed the adequacy of the licensee's simulation facility (simulator) for use in operator licensing examinations and for satisfying experience requirements. The inspectors reviewed a sample of simulator performance test records (e.g., transient tests, malfunction tests, scenario based tests, post-event

tests, steady state tests, and core performance tests), simulator discrepancies, and the process for ensuring continued assurance of simulator fidelity in accordance with 10 CFR 55.46. The inspectors reviewed and evaluated the discrepancy corrective action process to ensure that simulator fidelity was being maintained. Open simulator discrepancies were reviewed for importance relative to the impact on 10 CFR 55.45 and 55.59 operator actions as well as on nuclear and thermal hydraulic operating characteristics. (02.09)

- Problem Identification and Resolution (10 CFR 55.59(c); SAT element 5 as defined in 10 CFR 55.4): The inspectors assessed the licensee's ability to identify, evaluate, and resolve problems associated with licensed operator performance (a measure of the effectiveness of its LORT Program and their ability to implement appropriate corrective actions to maintain its LORT Program up to date). The inspectors reviewed documents related to licensed operator performance issues (e.g., recent examination and inspection reports including cited and non-cited violations; U.S. Nuclear Regulatory Commission (NRC) End of Cycle and Mid-Cycle reports; NRC plant issue matrix; licensee event reports; licensee condition/problem identification reports including documentation of plant events and review of industry operating experience). The inspectors also sampled the licensee's quality assurance oversight activities, including licensee training department self-assessment reports. (02.10)

This inspection constituted one Biennial Licensed Operator Requalification Program inspection sample as defined in IP 71111.11–05.

b. Findings

No findings were identified.

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

.1 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- risk evaluation associated with emergent high pressure coolant injection (HPCI) system work;
- work evaluation while in yellow risk with low pressure injection function, 'A' residual heat removal service water and 'A' control building chiller (CBC) systems inoperable and/or unavailable;
- 'A' standby diesel generator critical maintenance management activity; and
- risk evaluation associated with the loss of, and emergent work on, the 'B' instrument Vac inverter.

These activities were selected based on their potential risk significance relative to the Reactor Safety Cornerstones. As applicable for each activity, the inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4) and were accurate and complete. When emergent work was performed, the inspectors verified that the

plant risk was promptly reassessed and managed. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met.

Documents reviewed during this inspection are listed in the Attachment to this report.

These activities constituted four maintenance risk assessments and emergent work samples as defined in IP 71111.13–05.

b. Findings

No findings were identified.

1R15 Operability Determinations and Functional Assessments (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- ‘B’ main steam line automatic depressurization relief valve hi temperature indication during reactor startup;
- operable but nonconforming (OBN 02167905) cancelled for main steam isolation valve (MSIV) limit switch ZS4412A;
- HPCI Coupler Grease Extrusion;
- FLEX equipment inoperable and unavailable due to environmental protection agency features;
- standby filter Unit ‘A’ high temperature alarm;
- reactor building main intake heating coil deficiency found during site cold weather preparations; and
- diesel fuel oil storage tank water intrusion.

The inspectors selected these potential operability issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that 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 TS and UFSAR to the licensee’s evaluations to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment to this report.

This operability inspection constituted seven samples as defined in IP 71111.15–05.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

.1 Plant Modifications

a. Inspection Scope

The inspectors reviewed the following modification(s):

- traversing incore probe (TIP) digital upgrade.

The inspectors reviewed the configuration changes and associated 10 CFR 50.59 safety evaluation screening against the design basis, the UFSAR, and the TS, as applicable, to verify that the modification did not affect the operability or availability of the affected system(s). The inspectors, as applicable, observed ongoing and completed work activities to ensure that the modifications were installed as directed and consistent with the design control documents; the modifications operated as expected; post-modification testing adequately demonstrated continued system operability, availability, and reliability; and that operation of the modifications did not impact the operability of any interfacing systems. As applicable, the inspectors verified that relevant procedure, design, and licensing documents were properly updated. Lastly, the inspectors discussed the plant modification with operations, engineering, and training personnel to ensure that the individuals were aware of how the operation with the plant modification in place could impact overall plant performance. Documents reviewed are listed in the Attachment to this report.

This activity constituted one permanent plant modification sample as defined in IP 71111.18–05.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the following post-maintenance activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- repack residual heat removal service water loop ‘A’ supply header isolation valve V13–0023;
- lube and inspect gearbox and limit switch for the residual heat removal cross-tie, MO 2010–O;
- ‘A’ CBC temperature controller;
- ‘A’ CBC compressor discharge pressure transducer;

- ‘A’ CBC temperature load controller;
- ‘A’ essential service water strainer shaft replacement; and
- ‘A’ standby diesel generator critical maintenance management.

These activities were selected based upon the structure, system, or component's ability to impact risk. The inspectors evaluated these activities for the following (as applicable): the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was returned to its operational status following testing (temporary modifications or jumpers required for test performance were properly removed after test completion); and test documentation was properly evaluated. The inspectors evaluated the activities against TSs, the UFSAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with post-maintenance tests to determine whether the licensee was identifying problems and entering them in the CAP and that the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the Attachment to this report.

This activity constituted seven post-maintenance testing samples as defined in IP 71111.19–05.

b. Findings

No findings were identified.

1R20 Outage Activities (71111.20)

.1 Other Outage Activities

a. Inspection Scope

The inspectors evaluated outage activities for an unscheduled forced outage that began on October 23, 2017, and continued through the October 25, 2017. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed or reviewed the reactor shutdown and cooldown, outage equipment configuration and risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, control of containment activities, personnel fatigue management, startup and heatup activities, and identification and resolution of problems associated with the outage. The outage was initiated to address an approximate 1 gallon per minute increase in unidentified drywell leakage. The increased leakage was from a packing leak on a recirculation loop sample line isolation valve. During the return to full power following the forced outage, the licensee failed to maintain reactor water within procedurally specified limits. This resulted in a recirculation pump runback to 45 percent speed and an unplanned power reduction of 12 percent. The issue was discussed further in Section 4OA2.4 of this report.

Documents reviewed are listed in the Attachment to this report.

This activity constituted one other outage sample as defined in IP 71111.20–05.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- surveillance test procedure NS 160003A; 'A' residual heat removal service water system leakage surveillance test (Routine); and
- reactor core isolation cooling simulated automatic actuation test (Routine).

The inspectors observed in-plant activities and reviewed procedures and associated records to determine the following:

- did preconditioning occur;
- the effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis;
- plant equipment calibration was correct, accurate, and properly documented;
- as-left setpoints were within required ranges; and the calibration frequency was in accordance with TSs, the UFSAR, procedures, and applicable commitments;
- measuring and test equipment calibration was current;
- test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied;
- test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used;
- test data and results were accurate, complete, within limits, and valid;
- test equipment was removed after testing;
- where applicable for inservice testing activities, testing was performed in accordance with the applicable version of Section XI, American Society of Mechanical Engineers code, and reference values were consistent with the system design basis;
- where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure;

- where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;
- prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;
- equipment was returned to a position or status required to support the performance of its safety functions; and
- all problems identified during the testing were appropriately documented and dispositioned in the CAP.

Documents reviewed are listed in the Attachment to this report.

These inspections constituted two routine surveillance testing samples as defined in IP 71111.22–02 and –05. In addition, the inspectors did not identify any performance degradation in the RCS leakage for the entire cycle other than the previously discussed recirculation loop sample line isolation valve packing leak identified by the licensee. Because the licensee immediately shut down the reactor, located, and repaired the leak, after they had observed an increase in drywell particulate activity, before an increasing trend in unidentified leakage was established, the reactor coolant leak detection inspection sample was not performed as defined in IP 71111.22–02.

b. Findings

No findings were identified.

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspectors performed an in-office review of the latest revisions to the Emergency Plan, Emergency Action Levels (EALs), and EAL Bases document to determine if these changes decreased the effectiveness of the Emergency Plan. The inspectors also performed a review of the licensee's 10 CFR 50.54(q) change process and Emergency Plan change documentation to ensure proper implementation for maintaining Emergency Plan integrity.

The U.S. Nuclear Regulatory Commission review was not documented in a Safety Evaluation Report and did not constitute approval of licensee-generated changes; therefore, this revision is subject to future inspection. The specific documents reviewed during this inspection are listed in the Attachment to this report.

This EAL and Emergency Plan Change inspection constituted one sample as defined in Inspection Procedure 71114.04.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstones: Occupational and Public Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

.1 Radiological Hazard Assessment (02.02)

a. Inspection Scope

The inspectors assessed the licensee's current and historic isotopic mix, including alpha emitters and other hard-to-detect radionuclides. The inspectors evaluated whether survey protocols were reasonable to identify the magnitude and extent of the radiological hazards.

The inspectors determined if there have been changes to plant operations since the last inspection that may have resulted in a significant new radiological hazard for onsite individuals. The inspectors evaluated whether the licensee assessed the potential impact of these changes and implemented periodic monitoring, as appropriate, to detect and quantify the radiological hazard. The inspectors reviewed the last two radiological surveys from selected plant areas and evaluated whether the thoroughness and frequency of the surveys were appropriate for the given radiological hazard.

The inspectors conducted walkdowns of the facility, including radioactive waste processing, storage, and handling areas to evaluate material conditions and performed independent radiation measurements as needed to verify conditions were consistent with documented radiation surveys.

The inspectors assessed the adequacy of pre-work surveys for select radiologically risk-significant work activities.

The inspectors evaluated the radiological survey program to determine if hazards were properly identified. The inspectors discussed procedures, equipment, and performance of surveys with radiation protection staff and assessed whether technicians were knowledgeable about when and how to survey areas for various types of radiological hazards.

The inspectors reviewed work in potential airborne areas to assess whether air samples were being taken appropriately for their intended purpose and reviewed various survey records to assess whether the samples were collected and analyzed appropriately. The inspectors also reviewed the licensee's program for monitoring contamination which has the potential to become airborne.

These inspection activities constituted one complete sample as defined in IP 71124.01–05.

b. Findings

No findings were identified.

.2 Instructions to Workers (02.03)

a. Inspection Scope

The inspectors reviewed select radiation work permits used to access high radiation areas and evaluated the specified work control instructions or control barriers. The inspectors also assessed whether workers where made aware of the work instructions and area dose rates.

The inspectors reviewed electronic alarming dosimeter dose and dose rate alarm setpoint methodology. For selected electronic alarming dosimeter occurrences, the inspectors assessed the worker's response to the alarm, the licensee's evaluation of the alarm, and any follow-up investigations.

The inspectors reviewed the licensee's methods for informing workers of changes in plant operations or radiological conditions that could significantly impact their occupational dose.

The inspectors reviewed the labeling of select containers of licensed radioactive material that could cause unplanned or inadvertent exposure to workers.

These inspection activities constituted one complete sample as defined in IP 71124.01–05.

b. Findings

No findings were identified.

.3 Contamination and Radioactive Material Control (02.04)

a. Inspection Scope

The inspectors observed locations where the licensee monitors material leaving the radiologically controlled area and assessed the methods used for control, survey, and release of material from these areas. As available, the inspectors observed health physics personnel surveying and releasing material for unrestricted use.

The inspectors observed workers leaving the radiologically controlled area and assessed their use of tool and personal contamination monitors and reviewed the licensee's criterial for use of the monitors.

The inspectors assessed whether instrumentation was used at its typical sensitivity levels based on appropriate counting parameters or whether the licensee had established a de facto release limit.

The inspectors selected several sealed sources from the licensee's inventory records and assessed whether the sources were accounted for and verified to be intact. The inspectors also evaluated whether any transactions, since the last inspection, involving nationally tracked sources were reported in accordance with 10 CFR 20.2207.

These inspection activities constituted one complete sample as defined in IP 71124.01–05.

b. Findings

No findings were identified.

.4 Radiological Hazards Control and Work Coverage (02.05)

a. Inspection Scope

The inspectors evaluated ambient radiological conditions during tours of the facility. The inspectors assessed whether the conditions were consistent with applicable posted surveys, radiation work permits, and worker briefings.

The inspectors evaluated the adequacy of radiological controls, such as required surveys, radiation protection job coverage, and contamination controls. The inspectors evaluated the licensee's use of electronic alarming dosimeters in high noise areas as high radiation area monitoring devices.

The inspectors assessed whether radiation monitoring devices were placed on the individual's body consistent with licensee procedures. The inspectors assessed whether the dosimeter was placed in the location of highest expected dose or that the licensee properly employed a U.S. Nuclear Regulatory Commission approved method of determining effective dose equivalent.

The inspectors reviewed the application of dosimetry to effectively monitor exposure to personnel in work areas with significant dose rate gradients.

For select airborne area radiation work permits, the inspectors reviewed airborne radioactivity controls and monitoring, the potential for significant airborne levels, containment barrier integrity, and temporary filtered ventilation system operation.

The inspectors examined the licensee's physical and programmatic controls for highly activated or contaminated materials stored within pools and assessed whether appropriate controls were in place to preclude inadvertent removal of these materials from the pool.

These inspection activities constituted one complete sample as defined in IP 71124.01–05.

b. Findings

No findings were identified.

.5 High Radiation Area and Very High Radiation Area Controls (02.06)

a. Inspection Scope

The inspectors observed posting and physical controls for high radiation areas and very high radiation areas to assess adequacy.

The inspectors conducted a selective inspection of posting and physical controls for high radiation areas and very high radiation areas to assess conformance with performance indicators.

The inspectors reviewed procedural changes to assess the adequacy of access controls for high and very high radiation areas to determine whether procedural changes substantially reduced the effectiveness and level of worker protection.

The inspectors assessed the controls the high radiation areas greater than 1 rem/hour and areas with the potential to become high radiation areas greater than 1 rem/hour for compliance with TS and procedures.

The inspectors assessed the controls for very high radiation areas and areas with the potential to become very high radiation areas. The inspectors also assessed whether individuals were unable to gain unauthorized access to these areas.

These inspection activities constituted one complete sample as defined in IP 71124.01–05.

b. Findings

No findings were identified.

.6 Radiation Worker Performance and Radiation Protection Technician Proficiency (02.07)

a. Inspection Scope

The inspectors observed radiation worker performance and assessed their performance with respect to radiation protection work requirements, the level of radiological hazards present, and radiation work permit controls.

The inspectors assessed worker awareness of electronic alarming dosimeter set points, stay times, or permissible dose for radiologically significant work as well as expected response to alarms.

The inspectors observed radiation protection technician performance and assessed whether the technicians were aware of the radiological conditions and radiation work permit controls and whether their performance was consistent with training and qualifications for the given radiological hazards.

The inspectors observed radiation protection technician performance of radiation surveys and assessed the appropriateness of the instruments being used, including calibration and source checks.

These inspection activities constituted one complete sample as defined in IP 71124.01–05.

b. Findings

No findings were identified.

.7 Problem Identification and Resolution (02.08)

a. Inspection Scope

The inspectors assessed whether problems associated with radiological hazard assessment and exposure controls were being identified at an appropriate threshold and were properly addressed for resolution. For select problems, the inspectors assessed the appropriateness of the corrective actions. The inspectors also assessed the licensee's program for reviewing and incorporating operating experience.

The inspectors reviewed select problems related to human performance errors and assessed whether there was a similar cause and whether corrective actions taken resolve the problems.

The inspectors reviewed select problems related to radiation protection technician error and assessed whether there was a similar cause and whether corrective actions taken resolve the problems.

These inspection activities constituted one complete sample as defined in IP 71124.01–05.

b. Findings

No findings were identified.

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

.1 Radiological Work Planning (02.02)

a. Inspection Scope

The inspectors compared the results achieved with the intended dose established in the as-low-as-reasonably-achievable planning. The inspectors compared the person-hour estimates provided by work groups to the radiation protection group with the actual work activity time results, and evaluated the accuracy of these time estimates. The inspectors evaluated the reasons for any inconsistencies between intended and actual work activity doses.

The inspectors evaluated whether post-job reviews were conducted to identify lessons learned and entered into the licensee's CAP.

These inspection activities supplemented those documented in Inspection Report 05000331/2016004 and constituted one complete sample as defined in IP 71124.02–05.

b. Findings

No findings were identified.

.2 Verification of Dose Estimates and Exposure Tracking Systems (02.03)

a. Inspection Scope

The inspectors reviewed selected occasions with inconsistent or incongruent results from the licensee's intended radiological outcomes to determine whether the cause was attributed to a failure to adequately plan work activities, or failure to provide sufficient management oversight of in-plant work activities, or failure to conduct the work activity without significant rework, or failure to implement radiological controls as planned.

These inspection activities supplemented those documented in Inspection Report 05000331/2016004 and constituted one complete sample as defined in IP 71124.02–05.

b. Findings

No findings were identified.

.3 Implementation of As-Low-As-Reasonably-Achievable and Radiological Work Controls (02.04)

a. Inspection Scope

The inspectors compared the radiological results achieved with the intended radiological outcomes and verified that the licensee captured lessons learned for use in the next outage.

These inspection activities supplemented those documented in Inspection Report 05000331/2016004 and constituted one complete sample as defined in IP 71124.02–05.

b. Findings

No findings were identified.

2RS5 Radiation Monitoring Instrumentation (71124.05)

.1 Calibration and Testing Program (02.03)

a. Inspection Scope

The inspectors reviewed the methods and sources used to perform whole body count functional checks before daily use and assessed whether check sources were appropriate and aligned with the plant's isotopic mix. The inspectors reviewed whole body count calibration records since the last inspection and evaluated whether calibration sources were representative of the plant source term and that appropriate calibration phantoms were used. The inspectors looked for anomalous results or other indications of instrument performance problems.

These inspection activities supplemented those documented in Inspection Report 05000331/2017003 and constituted one complete sample as defined in IP 71124.05–05.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, and Occupational and Radiation Safety

4OA1 Performance Indicator Verification (71151)

.1 Safety System Functional Failures

a. Inspection Scope

The inspectors sampled licensee submittals for the Safety System Functional Failures performance indicator (PI) for the period from the fourth quarter 2016 through the third quarter 2017. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in Nuclear Energy Institute (NEI) Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013, and NUREG–1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73," were used. The inspectors reviewed the licensee's operator narrative logs, operability assessments, maintenance rule records, maintenance work orders, issue reports, event reports and NRC Integrated Inspection Reports for the period of October 2016 through September 2017, to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This activity constituted one safety system functional failures sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index—Residual Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index (MSPI) - Residual Heat Removal System PI, for the period from the fourth quarter 2016 through the third quarter 2017. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in NEI Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, MSPI derivation reports, event reports and NRC Integrated Inspection Reports for the period of October 2016 through the September 2017 to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable

NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This activity constituted one MSPI residual heat removal system sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.3 Mitigating Systems Performance Index—Cooling Water Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index - Cooling Water Systems PI for the period from the fourth quarter 2016 through the third quarter 2017. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in NEI Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, MSPI derivation reports, event reports and NRC Integrated Inspection Reports for the period of October 2016 through the September 2017 to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This activity constituted one MSPI cooling water system sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.4 Reactor Coolant System Specific Activity

a. Inspection Scope

The inspectors sampled licensee submittals for the reactor coolant system specific activity PI for the period from the third quarter 2016 through the third quarter 2017. The inspectors used PI definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 2013, to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee's reactor coolant system chemistry samples, TS requirements, issue reports, event reports and NRC Integrated Inspection Reports to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator. In addition to record reviews, the inspectors

observed a chemistry technician obtain and analyze a reactor coolant system sample. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one reactor coolant system specific activity sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.5 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors sampled licensee submittals for the Occupational Exposure Control Effectiveness PI for the period from the third quarter 2016 through the third quarter 2017. The inspectors used PI definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 2013, to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee's assessment of the PI for occupational radiation safety to determine if indicator related data was adequately assessed and reported. To assess the adequacy of the licensee's PI data collection and analyses, the inspectors discussed with radiation protection staff, the scope and breadth of its data review and the results of those reviews. The inspectors independently reviewed electronic personal dosimetry dose rate and accumulated dose alarms and dose reports and the dose assignments for any intakes that occurred during the time period reviewed to determine if there were potentially unrecognized occurrences. The inspectors also conducted walkdowns of numerous locked high and very high radiation area entrances to determine the adequacy of the controls in place for these areas. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one occupational exposure control effectiveness sample as defined in IP 71151–05.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify they were being entered into the licensee's CAP at an appropriate threshold, adequate attention was being given to timely corrective actions, and adverse trends were identified and addressed. Some minor issues were entered into the licensee's corrective action program as a result of the inspectors' observations; however, they are not discussed in this report.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead, by procedure they were considered an integral part of the inspections performed during the quarter.

b. Findings

No findings were identified.

.2 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also considered the results of daily inspector corrective action program item screening discussed in Section 4OA2.1 above, licensee trending efforts, and licensee human performance results. The inspectors' review nominally considered the 6-month period of June 2017 through November 2017, although some examples expanded beyond those dates where the scope of the trend warranted.

The review also included issues documented outside the CAP in major equipment problem lists, repetitive and/or rework maintenance lists, departmental problem/challenges lists, system health reports, quality assurance audit/surveillance reports, self-assessment reports, and Maintenance Rule assessments. The inspectors compared and contrasted their results with the results contained in the licensee's CAP trending reports. Corrective actions associated with a sample of the issues identified in the licensee's trending reports were reviewed for adequacy.

This activity constituted one semi-annual trend review inspection sample as defined in IP 71152.

b. Observations

The inspectors reviewed CR 02232776 which documented an issue in which operations department personnel received a high radiation area briefing prior to entering the steam tunnel to remove equipment tags while the reactor was shutdown. It was noted later that the operations personnel required a locked high radiation area key in order to access the steam tunnel and that the key had been signed out on the administrative key log rather than the locked high radiation area key log as required by site procedures. Licensee radiation protection confirmed that the area had not been down posted from a locked high radiation area and should have been briefed as such as required by procedures. These issues were considered minor violations as the licensee confirmed through surveys that the area radiation levels did not meet or exceed the threshold for a locked high radiation area. The inspectors challenged licensee management regarding shutdown procedures that should have down posted the steam tunnel area from a locked high radiation area to a high radiation area upon reactor shutdown, operator questioning attitude due to operators failing to recognize and stop when they encountered the locked high radiation area posting, and radiation protection department questioning attitude for signing out a locked high area radiation key after performing a briefing for a high radiation area.

The inspectors additionally reviewed CR 02232265, which documented that during a forced outage three individuals received internal depositions during valve repacking work. The reactor was in a forced outage and shutdown in order for the licensee to fix a valve packing leak inside the drywell. The licensee determined that because the valve was backseated and not actively leaking at the time that the individuals tasked with building the scaffolding structure up to the valve could do so without wearing respiratory protection. The licensee believed that requiring the scaffold workers to wear respiratory protection would encumber the individuals and add unnecessary dose. The inspectors questioned the licensee's decision making process by specifically challenging the reliance on the valve not actively leaking to make the decision on whether the scaffold builders should wear respiratory equipment.

The inspectors reviewed CR 02233479, which documented an issue where operations personnel contaminated two areas of the reactor building while draining the chemical waste tank. The inspectors challenged the operations department's decision to leave the area during an evolution that was actively draining a tank that contained contamination.

Due to the inspector's observations, the licensee documented the concerns in CR 02234985 and performed a Level 1 assessment for the potential trend in radiation protection performance site wide. Corrective actions from the Level 1 assessment included the formation of a cross functional team to develop, implement and assess a radiological safety improvement plan.

c. Findings

No findings were identified.

.3 Annual Follow-up of Selected Issues: Increasing Trend Identified in Reactor Coolant System Sulfates

a. Inspection Scope

The inspectors selected the following condition reports for in-depth review:

- CR 02236101; Cognitive Trend-Low Level Trend In Reactor Sulfates; and
- CR 02236241; Routine Surveys Reveal Higher Than Normal Dose Rates.

These condition reports were selected because they represented two potentially related events that the licensee evaluated in an effort to identify the source of an increasing trend in reactor water sulfates.

As appropriate, the inspectors verified the following attributes during their review of the licensee's corrective actions for the above condition reports and other related condition reports:

- complete and accurate identification of the problem in a timely manner commensurate with its safety significance and ease of discovery;
- consideration of the extent of condition, generic implications, common cause, and previous occurrences;
- evaluation and disposition of operability/functionality/reportability issues;

- classification and prioritization of the resolution of the problem commensurate with safety significance;
- identification of the root and contributing causes of the problem; and
- identification of corrective actions, which were appropriately focused to correct the problem;
- completion of corrective actions in a timely manner commensurate with the safety significance of the issue;
- effectiveness of corrective actions taken to preclude repetition; and
- evaluate applicability for operating experience and communicate applicable lessons learned to appropriate organizations.

The inspectors discussed the corrective actions and associated evaluations with licensee personnel.

This activity constituted one in-depth problem identification and resolution inspection sample as defined in IP 71152.

b. Observations and Assessments

The licensee demonstrated the ability to evaluate seemingly unrelated issues, refute inconsistent theories, apply operating experience, arrive at a likely conclusion and correct the deficiency. Specifically, the licensee conducted a level 1 core business assessment and determined that channeling in a reactor water clean-up demineralizer led to increased reactor coolant sulfate concentration and changed area radiation conditions. The licensee reapplied pre-coat to the demineralizer which returned reactor coolant sulfates and area radiation levels to the expected range.

c. Findings

No findings were identified.

.4 Annual Follow-up of Selected Issues: Trends in Operations Fundamentals

a. Inspection Scope

The inspectors selected the following condition report for in-depth review:

- CR 02233094; 45 Percent Recirc Pump A&B Runback.

The basis for this selection was an observed trend in operations fundamentals that included questioning attitude, operations leadership, and teamwork.

As appropriate, the inspectors verified the following attributes during their review of the licensee's corrective actions for the above condition reports and other related condition reports:

- complete and accurate identification of the problem in a timely manner commensurate with its safety significance and ease of discovery;
- consideration of the extent of condition, generic implications, common cause, and previous occurrences;
- evaluation and disposition of operability/functionality/reportability issues;

- classification and prioritization of the resolution of the problem commensurate with safety significance;
- identification of the root and contributing causes of the problem;
- identification of corrective actions, which were appropriately focused to correct the problem;
- completion of corrective actions in a timely manner commensurate with the safety significance of the issue;
- effectiveness of corrective actions taken to preclude repetition; and
- evaluate applicability for operating experience and communicate applicable lessons learned to appropriate organizations.

The inspectors discussed the corrective actions and associated evaluations with licensee personnel.

This activity constituted one in-depth problem identification and resolution inspection sample as defined in IP 71152.

Observations and Assessments

As discussed in the finding below, DAEC management took action to address self-revealed, licensee identified, and NRC identified weaknesses in questioning attitude, operations leadership and crew teamwork.

b. Findings

Failure to Maintain Reactor Water Level within Procedurally Required Level Band Results in Reactor Recirculation Pump Runback

Introduction: A self-revealed finding of very low safety significance and an associated Non-Cited Violation (NCV) of 10 Code of Federal Regulations (CFR) Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings", was identified on October 26, 2017, due to operations personnel failing to control reactor vessel water level as instructed by Integrated Plant Operating Procedure (IPOI) 2, "Startup," Revision 160, Section 3.0, "Reactor Startup." While at 55 percent reactor power during a reactor startup, with only one reactor feed pump running, the operating crew manipulated the reactor vessel water level automatic control set point which resulted in reactor water level dropping below the procedurally specified water level band. This resulted in a recirculation pump runback to 45 percent speed and an unplanned reactor power decrease from 55 to 43 percent.

Description: On October 26, 2017, DAEC was at 55 percent reactor power and increasing, following a forced outage. The crew was controlling reactor vessel water level in accordance with Integrated Plant Operating Procedure (IPOI) 2, "Startup," Revision 160, Section 3.0, "Reactor Startup," Step 8, which stated, in part, to "Control RPV [Reactor Pressure Vessel] Level between +186 and +196 inches." Reactor operators were controlling reactor water level at 191 inches with the reactor water level controller in automatic.

The control room crew was briefed and was preparing to start the second condensate pump. The brief included discussion on an expected reactor water level increase of approximately 2 inches when starting the second condensate pump. The brief also

discussed the high level alarm at 195 inches. However, there was no discussion of lowering the automatic reactor water level control set point to compensate for the anticipated increase in reactor water level. There was also no discussion of the automatic reactor recirculation pump runback that occurs if water level drops below 186 inches with only one reactor feed pump running.

After operators in the field reported that the prerequisites for starting the second condensate pump had been completed, and without notifying the control room supervisor, the auxiliary reactor operator lowered the reactor water automatic level control set point from 191 to 187.5 inches to provide additional margin to the high reactor water alarm at 195 inches. The operator was aware that it takes some time for the water level to stabilize at the new set point and that level may overshoot the new set point before stabilizing. However, the reactor operator was not concerned that the over shoot would initiate a recirculation pump runback because the operator anticipated starting the second condensate pump within 30 seconds of making the adjustment. Subsequently, the second condensate pump start was delayed for over two minutes when a field operator called the control room and postponed the pump start until the switchgear room was verified clear of personnel. During the delay, reactor water level decreased to less than 186 inches and initiated the reactor recirculation pump runback. Reactor power decreased from 55 to 43 percent. The licensee verified that reactor power stabilized at 43 percent and halted the reactor startup until a human performance review could be performed. There were no complications as a result the runback.

Analysis: The inspectors determined that the failure to maintain reactor water level between 186 inches and 195 inches as prescribed by IPOI 2, “Startup,” Revision 160, was reasonably within the licensee’s ability to foresee and correct. Specifically, the licensee manipulated the reactor vessel water level automatic control set point and allowed reactor water level to decrease to less than 186 inches. This resulted in an automatic recirculation pump runback and an unplanned reactor power decrease from 55 to 43 percent.

The performance deficiency was determined to be more than minor in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Screening,” dated September 7, 2012, because the finding impacted the Initiating Events Cornerstone Attribute of Human Performance and adversely affected the Cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during power operations. The inspectors applied IMC 0609, Attachment 4, “Initial Characterization of Findings,” to this finding. The inspectors answered “No” to all questions within Table 3 – Significance Determination Process (SDP) Appendix Router, and transitioned to IMC 0609, Appendix A, “The Significance Determination Process for Findings At-Power.” Per Exhibit 1 – Initiating Events Screening Questions, the inspectors answered “No” to the Question B Transient Initiator, and therefore, the finding screened as very low safety significance (Green).

The inspectors determined this finding affected the cross-cutting area of human performance in the aspect of teamwork, where individuals and work groups communicate and coordinate their activities within and across organization boundaries to ensure nuclear safety is maintained. Specifically, a reactor operator manipulated the reactor water level automatic control setpoint without notifying the control room supervisor, briefing the evolution, or obtaining a peer check. (H.4)

Enforcement: Title 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.

Integrated Plant Operating Procedure 2, "Startup," Revision 160, Section 3.0, "Reactor Startup," Step 8, stated, in part, to "Control RPV [Reactor Pressure Vessel] Level between +186 and +196 inches."

Contrary to the above, on October 26, 2017, the licensee failed to accomplish an activity affecting quality in accordance with IPOI 2. Specifically, the licensee lowered the reactor vessel level automatic control setpoint and allowed reactor water level to decrease outside the required level band, as specified in IPOI 2, which caused an automatic recirculation pump runback. This resulted in an unplanned reactor power decrease from 55 to 43 percent.

The licensee responded to the transient and verified that reactor power stabilized at 43 percent without complications. The licensee conducted a human performance review and recommenced power ascension with a new operating crew following normal shift changeover. Because the violation was of very low safety significance and was entered into the licensee's corrective action program as CR 02233094, this violation is being treated as an NCV, consistent with Section 2.3.2.a of the Enforcement Policy. **(NCV 05000331/2017004–01: Failure to Maintain Reactor Water Level within Procedurally Required Level Band Results in Reactor Recirculation Pump Runback)**

.5 Annual Follow-up of Selected Issues: Corrective Actions for Declining Performance Trends in Emergency Response Organization

a. Inspection Scope

The inspectors selected the following condition reports for in-depth review:

- CR 02204192; Communication Error During Notification Process Of Table Top.

This CR was chosen for review because in May 2017 the licensee evaluated a failed drill and exercise performance (DEP) PI opportunity as acceptable. In addition, DAEC management provided the inspectors with an incorrect regulatory interpretation of the DEP PI success criteria. This issue was documented in NRC Integrated Inspection Report 05000331/2017002.

As appropriate, the inspectors verified the following attributes during their review of the licensee's corrective actions for the above condition reports and other related condition reports:

- complete and accurate identification of the problem in a timely manner commensurate with its safety significance and ease of discovery;
- consideration of the extent of condition, generic implications, common cause, and previous occurrences;
- evaluation and disposition of operability/functionality/reportability issues;
- classification and prioritization of the resolution of the problem commensurate with safety significance;

- identification of the root and contributing causes of the problem;
- identification of corrective actions, which were appropriately focused to correct the problem;
- completion of corrective actions in a timely manner commensurate with the safety significance of the issue;
- effectiveness of corrective actions taken to preclude repetition; and
- evaluate applicability for operating experience and communicate applicable lessons learned to appropriate organizations.

The inspectors discussed the corrective actions and associated evaluations with licensee personnel.

This activity constituted one in-depth problem identification and resolution inspection sample as defined in IP 71152.

b. Observations and Assessments

The licensee initiated corrective actions to improve the emergency response organization. The licensee reorganized the Emergency Preparedness (EP) structure at DAEC and other NextEra sites to have the site EP manager report to the site licensing manager. The DAEC line organization has taken responsibility for emergency response readiness. The licensee has become more critical in assessing DEP PI performance. The inspectors have observed an increasing trend in emergency response sensitivity and readiness.

c. Findings

No findings were identified.

4OA5 Other Activities

.1 Operation of an Independent Spent Fuel Storage Installation at Operating Plants (60855.1)

a. Inspection Scope

(1) Certificate of Compliance 1004 Updated Final Safety Analysis Report Revision Control Unresolved Item

During the 2015 Independent Spent Fuel Storage Installation (ISFSI) inspection, the inspectors identified an unresolved item (URI) associated with the licensee's ISFSI Final Safety Analysis Report (FSAR) change management. Specifically, the licensee was maintaining its Campaign I NUHOMS 61BT casks to Certificate of Compliance (CoC) 072–01004, Amendment 8, which originally corresponded to FSAR, Revision 9. The licensee loaded and was maintaining its Campaign II NUHOMS 61BT casks to CoC 072–01004, Amendment 9, which originally corresponded to FSAR, Revision 10. The licensee used the 10 CFR 72.48, "Changes, Tests, and Experiments," process to monitor both Campaign I and Campaign II casks to FSAR, Revision 11. Revision 11 was issued upon approval of CoC 072–01004, Amendment 10. The inspectors

questioned whether it was acceptable to have casks loaded to two different CoC amendments being maintained to one FSAR revision.

(2) Independent Spent Fuel Storage Installation Program Review

The inspector reviewed documents, interviewed plant personnel, and performed a walkdown of the ISFSI. The inspector determined the licensee's compliance with 10 CFR Part 72, 10 CFR Part 50, the applicable cask CoC, cask TS, cask FSAR, and approved ISFSI procedures.

During the walkdown, the inspector evaluated the material condition of the ISFSI pad and Horizontal Storage Modules (HSMs) and performed independent radiation surveys. The inspector observed the licensee perform survey measurements of the ISFSI pad to monitor differential and absolute ISFSI pad settlement. The inspector also observed the licensee perform a daily visual TS surveillance of the HSMs. The inspector reviewed radiological surveys performed by the licensee and selected 10 CFR 72.48 reviews.

A review of corrective action reports written since the last ISFSI inspection indicated that the licensee was effectively identifying and correcting conditions adverse to quality.

b. Findings

(1) (Closed) Unresolved Item 07200032/2015001–01: Certificate of Compliance 1004 Final Safety Analysis Report Revision Control

The NRC staff has concluded that there is no specific regulatory prohibition on monitoring multiple CoC amendments to a single FSAR revision. However, the licensee must still be in compliance with the regulatory requirements of 10 CFR Part 72, in particular 10 CFR 72.48, which is the governing regulatory change process for implementing changes to the cask FSAR.

In addition to monitoring both Campaign I and Campaign II casks to FSAR Revision 11, the licensee maintains, as part of the cask licensing basis, DBD–F16–001, "Duane Arnold Energy Center Design Basis Document for the Dry Spent Fuel Storage Program," which includes both the 10 CFR 72.212 report and 10 CFR 72.48 screenings and evaluations.

The inspector found that the licensee was implementing an acceptable approach to cask FSAR changes as specified in NEI 96–07, Appendix B, Section B4.1.7, "Cask Design Changes Made by a CoC Holder and Adopted by a General Licensee." Specifically, when the CoC holder, Transnuclear (TN), performed an FSAR change, TN would communicate the change to the licensee with a FSAR Change Notice (FCN) and an associated 10 CFR 72.48 screening or evaluation. The licensee would then generate an Action Request to review the change communicated by the FCN.

The licensee would perform a licensing review which would either adopt the change in full, not adopt the change, adopt the change in part, or only adopt the change for casks associated with a single campaign, i.e., either Campaign I or Campaign II casks. Changes specifically associated with later cask amendments and cask procurement changes for casks that were already procured were not adopted by the licensee. Additionally, many changes that were only applicable to parts of CoC 072–01004

systems other than the chosen licensee cask system (i.e. NUHOMS 61BT casks, 102 HSMs, and OS197 transfer cask), were not adopted by the licensee.

For the changes that were adopted, the licensee performed a 10 CFR 72.48 screening or evaluation against the licensee's entire design basis document, including the 72.212 report. These licensee screening or evaluations were incorporated into the cask design basis document as a record of which changes were or were not adopted and the regulatory basis for adopting any changes. When an entire new FSAR revision was issued by TN, the licensee would perform a roll-up 72.48 screening to verify that all FCNs associated with the new revision had been evaluated by the process described above, evaluate any FCNs that had not already gone through the process, and would then reconcile the cask FSAR to the latest FSAR revision. This process was utilized to reconcile the FSAR for Campaign I casks from Revision 9 to Revision 10. Additionally, this process was utilized to reconcile the FSAR for Campaign I and Campaign II casks from Revision 10 to Revision 11. Since the reconciliation of both Campaign I and Campaign II casks to FSAR Revision 11, the licensee has not adopted any further changes against FSAR Revision 11.

Based upon the results of the NRC review, URI 07200032/2015001–01, "CoC 1004 FSAR Revision Control," is closed. No violations of NRC requirements were identified.

(2) <u>Failure to Evaluate Site Fire and Explosion Hazards in Accordance with 10 CFR 72.212(b)(6)</u>

<u>Introduction</u>: The inspector identified a Severity Level IV NCV of 10 CFR 72.212(b)(6), "Conditions of General License Issued under 72.210," for the failure of the licensee to determine whether reactor site parameters, specifically fire and explosion hazards, were enveloped by the cask design bases considered in the cask FSAR.

<u>Description</u>: The licensee has loaded ten 61BT casks under CoC 072–01004, Amendment 8, and ten 61BT casks under CoC 072–01004, Amendment 9. The licensee monitors both sets of casks to the TN NUHOMS FSAR, NUH–003, Revision 11. The licensee controlled combustible materials near these casks during storage operations in accordance with site Administrative Control Procedure (ACP) 1412.2, "Control of Combustibles," Revision 48. Attachment 2, Section 2.4.3 of this procedure describes requirements for the control of combustible liquids (e.g., diesel fuel, hydraulic fluid) at the ISFSI and Attachment 2, Section 2.4.4, describes requirements for the control of flammable liquids (e.g., gasoline, acetone) at the ISFSI.

The minimum separation requirements for combustible and flammable liquids as stated in Revision 48 of the procedure were as follows:

- Diesel fuel, turbine oil, EHC oil, etc.
  - 0–249 gallons; >10 feet
  - 250–6,000 gallons; >20 feet
  - >6,000 gallons; >30 feet
- Engine oils, hydraulic fluids, grease, etc.
  - 0–749 gallons; >10 feet
  - 750–12,000 gallons; >20 feet
  - >12,000 gallons; >30 feet

- Gasoline, acetone, etc.
    - o 0–600 gallons; >30 feet
    - o >600 gallons; >50 feet

The TN NUHOMS FSAR, Revision 11, Section K.4.6.5, evaluates a hypothetical fire accident for a NUHOMS 61BT dry shielded canister (DSC). This hypothetical fire consists of 300 gallons of diesel fuel, while the DSC is in a transfer cask, which also bounds the DSC in storage in an HSM. The FSAR does not specifically evaluate a hypothetical explosion accident, but does state in Section 3.3.6 that pressures due to explosions cannot exceed those pressures postulated by the FSAR for tornado missile or wind effects. The procedural limits in ACP 1412.2 are not clearly bounded by the existing analyses in the FSAR. The procedural limits exceed the 300 gallons of diesel fuel fire in the FSAR for combustible liquids, and the FSAR does not contain any hypothetical explosion accident with which to bound the flammable liquids.

In accordance with 10 CFR 72.212(b)(6), the licensee is required to determine whether reactor site parameters are enveloped by the cask design bases in the cask FSAR to ensure that structures, systems, and components important to safety will perform their safety functions during postulated accident conditions. The licensee did perform a site-specific analysis of fire and explosion hazards in Calculation 70–007–0012, "Duane Arnold Energy Center Fire Hazards Analysis for the Dry Spent Fuel Program," accepted June 9, 2003. However, the licensee did not demonstrate in this analysis that the procedural limits for combustible and flammable liquids were enveloped by the cask design bases in the FSAR. In this site-specific analysis these procedural limits were instead derived from National Fire Protection Association (NFPA) codes.

At the time of this inspection, no combustible or flammable liquids were being stored on the ISFSI pad or within the ISFSI protected area (PA). The combustible liquids from the vehicles used to transport DSCs to and from the ISFSI in total, as shown in Calculation 70–007–0012, are within the hypothetical fire accident described in the FSAR. The inspector reviewed the licensee's Combustible Material/Flammable Liquid Control Permits issued for the ISFSI and identified two instances in which combustible or flammable material was brought into the ISFSI PA that was not clearly bounded by the existing analyses in the FSAR. These instances were in 2010, when new HSMs were being installed, and in 2017, when a propane-powered lift was utilized.

Analysis: The licensee's failure to evaluate reactor site fire and explosion hazards in accordance with 10 CFR 72.212(b)(6) was a performance deficiency. In accordance with Section 2.2 of the NRC Enforcement Policy and IMC 0612, Appendix B, "Issue Screening," ISFSI facilities are not subject to the SDP and are not subject to the Reactor Oversight Process, so violations identified at ISFSIs are assessed using traditional enforcement. Traditional enforcement violations are not assessed for cross-cutting aspects.

The inspector determined that the violation was of more than minor significance using IMC 0612, "Power Reactor Inspection Reports", Appendix E, "Examples of Minor Issues." Example 4k was applicable to this example in that the lack of an evaluation to demonstrate the procedurally permitted combustible and flammable liquids were bounded by the design basis in the FSAR allowed for a credible unanalyzed fire and explosion scenario that could affect the important-to-safety dry cask storage system. Consistent with the guidance in Section 1.2.6.D of the NRC Enforcement Manual, if a

violation does not fit an example in the Enforcement Policy Violation Examples, it should be assigned a severity level: (1) commensurate with its safety significance, and (2) informed by a similar violation addressed in the violation examples. The inspector found no similar violations in the violation examples. As no combustible or flammable liquids were being stored on the ISFSI pad and the transport vehicles used to move the casks to the ISFSI were within the fire hazard analysis, the violation was determined to be of very low safety significance (Severity Level IV).

Enforcement: Title 10 CFR 72.212(b)(6) requires, in part, that the licensee review the Safety Analysis Report referenced in the CoC, prior to use of the general license, to determine whether or not the reactor site parameters are enveloped by the cask design bases. The results of this review must be documented in the evaluation made in 10 CFR 72.212(b)(5).

Contrary to the above, since June 9, 2003, the licensee failed to evaluate site-specific fire and explosion hazards as allowed by ACP 1412.2, “Control of Combustibles,” Revision 48, to determine whether or not those fire and explosion hazards were enveloped by the cask design bases. Specifically, the procedural limits in ACP 1412.2 were not clearly bounded by the existing analyses in the TN NUHOMS FSAR, Revision 11, and Calculation 70–007–0012, “Duane Arnold Energy Center Fire Hazards Analysis for the Dry Spent Fuel Program,” did not demonstrate that those procedural limits were enveloped by the cask design bases in the FSAR. The licensee entered this issue into its CAP as CR 02228514 and CR 02228558. Because this issue was of very low safety significance (Severity Level IV) and was entered into the licensee’s CAP, this violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. **(NCV 05000331/2017004–02, Failure to Evaluate Site Fire and Explosion Hazards in Accordance with 10 CFR 72.212(b)(6))**

4OA6 Management Meetings

.1 Exit Meeting Summary

On January 4, 2018, the inspectors presented the inspection results to Mike Strope, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

Interim exits were conducted for:

- The results of the ISFSI inspection were presented on October 6, 2017, to Mr. D. Curtland and other members of the licensee's management and staff;
- The inspection results for the Radiation Safety Program review with Mr. D. Curtland, Director Site Operations, on November 30, 2017;
- The results of the Emergency Preparedness Program inspection with Mr. M. Fritz, Emergency Preparedness Manager, conducted over the phone on December 14, 2017; and
- The results of the Biennial Operator Licensing Requalification inspection with Mr. D. Curtland, Director of Site Operations, and other members of the licensee's staff on December 15, 2017. The licensee acknowledged the issues presented.

The inspectors confirmed that none of the potential report input discussed was considered proprietary. Proprietary material received during the inspection was returned to the licensee.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

D. Curtland, Site Vice President
P. Hansen, Engineering Site Director
M. Davis, Licensing Manager
M. Fritz, Emergency Preparedness Manager
J. Karrick, Nuclear Oversight Supervisor
M. Strope, Operations Director
D. Morgan, Radiation Protection Manager
M. Casey, Chemistry Manager
J. Schwertfeger, Security Manager
C. Hill, Training Manager
B. Murrell, Licensing Senior Engineer
L. Swenzinski, Licensing Senior Engineer
P. Collingsworth, System Engineering
B. Lawrence, System Engineering Manager
B. Lindley, Reactor Engineering Supervisor
D. Garrison, Reactor Engineering
J. Azmeh, Design Engineering
D. Church, Engineering Programs Manager
R. Severson, BWRVIP Engineer
D. Slivon, ISI Programs Engineer
F. Dohmen, NDE Level III
T. Weaver, Senior Licensing Engineer
T. Granfors, Senior EP Coordinator
R. Palmer, Senior EP Coordinator
T. Ringgold, Senior EP Coordinator
D. Minnick, EP Drill Developer
P. Polfleit, NextEra Energy EP Manager
D. Bloomquist, Radiation Protection Supervisor
M. Walter, Operations Continuing Training Supervisor
E. Murray, Simulator/EP/NRC Examination Supervisor
W. Render, Operations Instructor/Examination Developer
J. Morris, Operations Instructor/Examination Develop
K. Gassman, Simulator Analyst

U.S. Nuclear Regulatory Commission

L. Kozak, Acting Chief, Reactor Projects Branch 1

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000331/2017004–01	NCV	Failure to Maintain Reactor Water Level within Procedurally Required Level Band Results in Reactor Recirculation Pump Runback (4OA2.4)
05000331/2017004–02	NCV	Failure to Evaluate Site Fire and Explosion Hazards in Accordance with 10 CFR 72.212(b) (6) (4OA5.1)

Closed

05000331/2017004–01	NCV	Failure to Maintain Reactor Water Level within Procedurally Required Level Band Results in Reactor Recirculation Pump Runback (4OA2.4)
05000331/2017004–02	NCV	Failure to Evaluate Site Fire and Explosion Hazards in Accordance with 10 CFR 72.212(b) (6) (4OA5.1)
07200032/2015001–01	URI	CoC 1004 FSAR Revision Control (4OA5.1)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspector reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

1R01 Adverse Weather Protection

- OP-AA-102-10002; Seasonal Readiness; Revision 20
- NG 270K; Plant Winterization Checklist; Revision 19
- CR 02235001; ‘B’ Main Intake Coil Glycol Leak

1R04 Equipment Alignment

- Operating Instruction (OI) 149A1; RHR System Electrical Lineup; Revision 4
- Drawing BECH-M119; P. & I. D. Residual Heat Removal System; Revision 85
- OI 149A4; “B” RHR System Valve Lineup; Revision 7
- ODI-13; Second Assistant Logs; Revision 50
- STP 3.8.4-02; Battery Connected Cell Checks; Revision 23
- STP 3.8.4-02A; Battery 1D1 & 1D5 Connected Cell Checks; Revision 5
- STP 3.8.4-02B; Battery 1D2 & 1D6 Connected Cell Checks; Revision 5
- STP 3.8.4-02C; Battery 1D4 & 1D93 Connected Cell Checks; Revision 3
- STP 3.8.4-05; Battery Inspections; Revision 12
- STP 3.8.4-05A; 1D1 And 1D5 Battery Inspections; Revision 2
- STP 3.8.4-05B; 1D2 And 1D6 Battery Inspections; Revision 3
- STP 3.8.4-05C; 1D4 Battery Inspection; Revision 2

1R05 Fire Protection

- Fire Hazards Analysis 400; Fire Hazards Analysis; Revision 23
- Pre-Fire Plan (PFP)-LL-757; Pre-Fire Plan LLRPSF; Revision 2
- PFP-CB-772; Pre-Fire Plan Control Building El. 772; Revision 1
- PFP-RB-757; Pre-Fire Plan Reactor Building El. 757; Revision 3
- PFP-RR-001; Pre-Fire Plan Radioactive Release; Revision 0
- PFP-TB-780; Pre-Fire Plan Turbine Building El. 780; Revision 4

1R06 Flood Protection Measures

- Radwaste Handling Procedure 3402.22; Radwaste Sump System; Revision 27
- Radwaste Handling Procedure 3402.8; Processing Contents of 1T-66 Chemical Waste Tank; Revision 18
- CR 02233479; Clean Area Contamination
- Drawing BECH-M137<1>; P. & I. D. Radwaste Sump System; Revision 39
- Drawing BECH-M137<2>; P. & I. D. Radwaste Sump System; Revision 17
- Drawing BECH-M138<1>; P. & I. D. Equipment Radwaste System (Closed); Revision 28
- Drawing BECH-M138<1a>; P. & I. D. Liquid Radwaste Treatment System; Revision 1
- Drawing BECH-M138<2>; P. & I. D. Equipment Radwaste System (Closed); Revision 17
- Drawing BECH-M137<2>; P. & I. D. Floor Drain Radwaste System; Revision 65

1R11 Licensed Operator Requalification Program

- CR 2231847; Increase In Unidentified Drywell Leakage
- OI 920; Drywell Sump System; Revision 48
- OI 698; Main Generator System; Revision 99
- OI 646; Extraction Steam System; Revision 67
- OI 264; Reactor Recirculation System; Revision 142
- IPOI 3; Power Operations (35% – 100% Rated Power); Revision 156
- IPOI 4; Shutdown; Revision 137
- OI 149; Residual Heat Removal System; Revision 165
- IPOI 2; Startup; Revision 160
- PDA OPS ESG 292
- ACP 103.10; Control of Time Critical Tasks; Revision 12
- ODI-039; Strategies for Successful Transient Mitigation; Revision 13
- OP-AA-100-1001-F02; License Activation/Reactivation Approval; Revision 1
- OP-032; Quarterly Personnel Watchstanding Verification; Revision 16
- TR-AA-220-1000; Systematic Approach to Training Process; Revision 0
- TR-AA-104; NextEra Energy Fleet Licensed Operator Continuing Training Program; Revision 9
- TR-AA-220-1004; Licensed Operator Continuing Training Annual Operating and Biennial Written Exams; Revision 2
- TR-AA-221-1000; Simulator Change Control; Revision 3
- TR-AA-230-1007; Conduct of Simulator Training and Validation; Revision 5
- TR-AA-230-1008; Simulator Scenario Based Testing and Validation; Revision 1

Condition Reports

- CR 02143830; Crew Failed Annual Operating Exam
- CR 02233094; 45% Recirc Pump ‘A’ & ‘B’ Runback
- CR 02100321; CB5960: Alliant Energy DC 20260 Switch Sequence
- CR 02100047; OPS In-Field Supervisor Out of Role
- CR 02104085; Procedure Use and Adherence Performance Trend Identified
- CR 02128777; 1Q2016 PAR [Performance Analysis Report] Determined Crew PU&A [Procedure Use and Adherence] is a New Adverse Trend
- CR 02160826; RFP [Refueling Procedure] 403 Checklist Requirements Not Met
- CR 02146439; Reset of SBDG Lube Oil HI/LO Temperature
- CR 02143225; Unexpected Alarm During Removal of Bed from Service
- CR 02155928; CV1718D Influent Valve Not Opened Per Procedure

Other

- ACP 103.10 Attachment 1; NFPA 805-Safe Shutdown Analysis, Shutdown Outside the Control Room; dated September 26, 2016
- ACP 103.10 Attachment 1; Blocking Open Control Room Back Cabinet Doors; September 7, 2016
- ACP 103.10 Attachment 1; Begin Aggressive Reactor Cooldown While in a Station Blackout Condition; September 7, 2016
- ACP 103.10 Attachment 1; Blocking Open Control Room Doors; September 7, 2016
- ACP 103.10 Attachment 1; Essential Switchgear Room Ventilation Instructions, and Block Open HPCI Room Upper and Lower Doors; September 7, 2016

- Focused Self-Assessment 02191812; NRC IP 71111.11 Pre-Inspection Assessment; September 11–14, 2017
- 2017 Duane Arnold Energy Center (DAEC) Licensed Operator Requalification (LOR) Program Biennial Written Examination, RO and SRO, for Week 5; December 2017
- 2017 DAEC LOR Program Annual Examination JPM Number 2.2.12-04; Revision 6
- 2017 DAEC LOR Program Annual Examination JPM Number 2.4.41-12; Revision 0
- 2017 DAEC LOR Program Annual Examination JPM Number 201001-03; Revision 14
- 2017 DAEC LOR Program Annual Examination JPM Number 202001-06; Revision 2
- 2017 DAEC LOR Program Annual Examination JPM Number 212000-01; Revision 7
- 2017 DAEC LOR Program Annual Examination JPM Number 217000-07; Revision 3
- 2017 DAEC LOR Program Annual Examination JPM Number 259002-06; Revision 1
- 2017 DAEC LOR Program Annual Examination JPM Number 295006-03; Revision 8
- 2017 DAEC LOR Program Annual Examination JPM Number 295024-07; Revision 0
- 2017 DAEC LOR Program Annual Examination JPM Number 295037-13; Revision 0
- 2016 DAEC LOR Program Annual Simulator Evaluation Scenario PDA OPS ESG 187; Revision 0
- 2016 DAEC LOR Program Annual Simulator Evaluation Scenario PDA OPS ESG 188; Revision 0
- 2016 DAEC LOR Program Annual Simulator Evaluation Scenario PDA OPS ESG 189; Revision 0
- 2017 DAEC LOR Program Annual Simulator Evaluation Scenario PDA OPS ESG 190; Revision 0
- 2017 DAEC LOR Program Annual Simulator Evaluation Scenario PDA OPS ESG 291; Revision 0
- 2017 DAEC LOR Program Annual Simulator Evaluation Scenario PDA OPS ESG 295; Revision 0
- 2017 DAEC LOR Program Annual Simulator Evaluation Scenario PDA OPS ESG 296; Revision 1
- Crew Simulator Evaluation Form; TR-AA- 230-1007-F01; July 12, 2016
- Individual Simulator Evaluation Forms; TR-AA- 230-1007-F02, for PDA OPS ESG-188; July 12, 2016
- STA Simulator Evaluation Form; TR-AA- 230-1007-F03, for PDA OPS ESG-188; July 12, 2016
- Training Remediation Forms; TR-AA- 230-1004-F04; Failure Date July 12, 2016
- Academic Review Board Form; TR-AA- 230-1004-F13; July 13, 2016
- Crew Simulator Evaluation Form; TR-AA- 230-1007-F01; July 15, 2016
- Individual Simulator Evaluation Forms; TR-AA- 230-1007-F02, for PDA OPS ESG-187 and 189; July 15, 2016
- Crew Simulator Evaluation Form; TR-AA- 230-1007-F01; December 12, 2017
- Individual Simulator Evaluation Forms; TR-AA- 230-1007-F02, for PDA OPS ESG-295 and 296; December 12, 2017
- STA Simulator Evaluation Forms; TR-AA- 230-1007-F03, for PDA OPS ESG-295 and 296; December 12, 2017
- 4.0; Stability/Steady State Tests; December 14, 2016
- A3.1; Real Time and Repeatability Test; November 28, 2016
- 3.4.3.3; Reactor Core Reload and Core Performance Testing [Fuel Cycle 26]; December 26, 2016
- TR 421; Manual SCRAM
- TR 424; Simultaneous Trip of Both Recirculation Pumps
- TR 427; Maximum Rate Power Ramp Down to Approximately 75% then Back Up to 100%

- Scenario Based Testing Packages for 2017 Duane Arnold Energy Center Licensed Operator Requalification Program Annual Simulator Evaluation Scenario PDA OPS ESG-295; Revision 0
- Scenario Based Testing Packages for 2017 Duane Arnold Energy Center Licensed Operator Requalification Program Annual Simulator Evaluation Scenario PDA OPS ESG-296; Revision 1
- Simulator vs Plant Differences Report; December 7, 2017

Simulator Work Requests

- 1684; Halt during SEG 2016-05S; January 20, 2016
- 1688; Halt during LOCT Evaluated Scenario; January 25, 2016
- 1719; Request New Malfunction be Developed for HPCI Hydraulic System Oil Leak; March 24, 2016
- 1741; Malfunction MS04, MSL Rupture Outside Primary Containment, Steam Line A, B, B, D Effects; May 18, 2016
- 1748; "D" Well Pump Tripped when Starting 'B' Circ Water Pump; June 3, 2016
- 1812; Malfunction ZZ03, Design Basis Earthquake Event; January 19, 2017
- 1822; Low Accum Press Annunc when Scramming Individual Rod; February 15, 2017
- 1843; Loss of Power to Intake Structure Buses 1B9 and 1B20 Should Activate Annunciator 1C23C, E-3; April 10, 2017
- 1860; Malfunction HP08, HPCI Steam Supply Line Break (Torus Room) Doesn't Heat Up Torus Room; May 9, 2017
- 1891; Torus Area Temperature Issue; August 9, 2017

<u>1R13 Maintenance Risk Assessments and Emergent Work Control</u>

- WO 40564594-01; HPCI Main Pump – Coupling Grease
- WO 40515732-01; Residual Heat Removal Service Water (RHRSW) Loop 'A' Supply Header Isolation
- Work Management (WM)-AA-203-1006; Critical Maintenance Management; Revision 7
- WM-AA-203; Online Scheduling Process; Revision 16
- Operations (OP)-AA-102-1003; Guarded Equipment; Revision 23
- Work Planning Guideline-2; On-Line Risk Management Guideline; Revision 70
- WO 40506630-01; 1G031/ENG
- WO 40398760-01; TS3213
- Abnormal Operating Procedure (AOP) 317; Loss of Vac Instrument Control Power; Revision 103
- CR 02238281; Unexpected Loss of 1Y21
- CR 02238292; Abnormal Indications on 1Y2A Following Restoration of Power

<u>1R15 Operability Evaluations</u>

- CR 01598986; PSV 4402 Has Elevated Temperature Indication
- CR 01598986-06; Root Cause Evaluation of PSV 4402 Leakage; December 12, 2010
- CR 01830710; PSV 4402 Tailpipe Temperature Rise During Reactor Start Up
- CR 02232755; PSV 4402 ADS Relief Valve Hi Temperature at 400 psig Reactor Pressure
- Operating Shift Log Entry 02-Oct-2017 1444
- CR 02167905; Trend-ZS 4412 Required Adjustment But Not Adjusted
- CAL-E94-005; Main Steam Isolation Valve 10 Percent Closed SCRAM-ZS4412A, ZS4413A, ZS4416A, ZS4418A, ZS4419A, And ZS4421A; Revision 3

- DGC-E111; Engineering Design Guide For The Duane Arnold Energy Center; Instrument Set Point Guide (GE-NE-901-007-0292); Revision 1
- Information Notice 85-84; Inadequate Inservice Testing Of Main Steam Isolation Valves
- NG-89-2155; MSIV Spring Only Closure; SIL No. 477; Revision 0
- NG-93-2470; Closure of Commitment #930059; Revision 0
- NG-93-3199; IST Program Relief Request VR-017; Revision 0
- NG-97-1809; MSIV Slow Closure Time; Revision 0
- STP 3.3.1.1-18; MSIV Limit Switch Calibration And Inspection; Revision 22
- Koppers Company Instruction Sheet Form 1900; Self-Aligning Couplers
- Operating Shift Log Entry 12-Oct-2017 1252
- CR 01988287; Grease Buildup On Inside Of Coupling Cover
- CR 02218868; Unusual Amount Of Grease On HPCI Gear Reduction Pedestal
- CR 02229276; HPCI Main Pump-Coupling Grease
- CR 02230197; FLEX FDR Vulnerability
- BECH-E113<037>; Heating And Ventilating Systems; Revision 14
- BECH-E113<039>; Heating And Ventilating Systems; Revision 05
- Nextera Energy Memo; Seasonal Readiness – Winter 2017; Dated September 24, 2017
- OP-AA-102-1002; Seasonal Readiness; Revision 21
- BECH-M173; Air Flow Diagram, Standby Filter Unit Control Building; Revision 61
- CR 02230549; Received SFU A HI TEMP Alarm
- CR 02231223; ‘A’ SFU Temperature Controller Failures
- M071-017<1>; Wiring Schematic Tate Reynolds Company; Revision 7
- M071-017<2>; Wiring Schematic Tate Reynolds Company; Revision 4
- CR 02239288; IT-034 Aux Boiler Fuel Oil “Storage Tank Water Intrusion”

1R18 Plant Modifications

- EC-284591; Modification Acceptance Test Replace TIP DCU Modification; Revision 1
- CR 02236450; ‘A’ TIP Veeder-Root Counter Erratic
- CR 02236451; New TIP DCU Failed Common Channel Test During MAT
- CR 02236452; TIP DCU Computer Points Fail To Respond
- CR 02236797; PPC Interface Card Failed To Respond To The New TIP DCU
- CR 02236809; Work On A TIP DCU Required Caution Tag Removed

1R19 Post Maintenance Testing

- WO 40515732-01; Repack RHRSW Loop ‘A’ Supply Header Isolation; V13-0023
- Operating Shift Log Entries 31-Oct-2017 133 and 31-Oct-2017 141
- WO 40536010-01; MO 2010-O, RHR Loop A/B Cross-Tie Header Isolation Valve
- STP 3.5.1-02A; ‘A’ LPCI System Operability Tests; Revision 18
- Operating Shift Log Entry 30-Oct-2017 1001
- WO 40514858-01; ‘A’ CB Chiller Temperature Load Controller
- WO 40435243-01; ‘A’ Chiller Compressor Discharge Press Transducer
- WO 40458303-05; ‘A’ CB Chiller Temperature Controller
- STP 3.7.5-01A; ‘A’ Control Room Building Chiller Operability; Revision 4
- WO 40404639; Replace 15098A Strainer Shaft
- WO 40552296 RTS Task; ‘A’ Standby Diesel Generator; Revision 0
- WO 40552296 RTS Task; ‘A’ Standby Diesel Generator; Revision 1
- STP NS54001A; ‘A’ ESW System Class 3 Leakage Inspection; Revision 8
- STP NS540002A; ‘A’ Emergency Service Water Operability Test; Revision 38
- STP 3.8.1-06A; ‘A’ Standby Diesel Generator Operability Test (Fast Start); Revision 24

1R20 Refueling and Other Outage Activities

- CR 2231982; External Leakage From Valve, Unknown As To Cause Of Leakage
- OI 920; Drywell Sump System; Revision 48
- OI 698; Main Generator System; Revision 99
- OI 646; Extraction Steam System; Revision 67
- OI 264; Reactor Recirculation System; Revision 142
- IPOI 3; Power Operations (35-100% Rated Power); Revision 156
- IPOI 4; Shutdown; Revision 137
- IPOI 2; Start-up; Revision 160
- AOP 683; Abnormal Safety Relief Valve Operation; Revision 19
- BECH-M116; P.& I.D.; Reactor Recirculation System
- FSK-05592<1>; Recirc. Sample Piping To SX-4640; Revision 3
- CR 02231847; Increase In Unidentified Drywell Leakage
- OSM Logs 22-Oct-2017
- OSM Logs 25-Oct-2017
- OSM Logs 26-Oct-2017
- CR 02233094; 45% Recirc Pump A&B Runback
- CR 02235465; Use of Robust Barrier When In Single Reactor Feed Pump

1R22 Surveillance Testing

- STP NS160003A; ‘A’ RHRSW System Leakage Inspection; Revision 4
- LMS Curricula Status Check; Daniel Dietrich; 10/31/2017 1:21:41 PM
- STP 3.5.3-04; RCIC Simulated Auto Actuation Test; Revision 19

1EP4 Emergency Action Level and Emergency Plan Changes

- Duane Arnold Energy Center Emergency Plan, Section B; Revisions 37 and 38
- Duane Arnold Energy Center Emergency Plan, Section F; Revisions 28 and 29
- Duane Arnold Energy Center Emergency Plan, Section G; Revisions 23 and 24
- Duane Arnold Energy Center Emergency Plan, Section H; Revisions 34 and 35
- Duane Arnold Energy Center Emergency Plan, Section N; Revisions 22 and 23
- EP-AA-100-1007; Evaluation of Changes to the Emergency Plan, Supporting Documents and Equipment (10 CFR 50.54(Q)); Revision 3
- Emergency Plan Implementing Procedure (EPIP) 4.3; Rescue and Emergency Repair Work; Revisions 10 and 11
- EPIP 6.1; Drill and Exercise Program; Revisions 5 and 6
- EPIP Form Par-01; Revisions 4 and 5
- NUC EPR 2001; 10 CFR 50.54(q) Qualification; Revision 0
- PCR 2126248 and 2126257; “10 CFR 50.54(q) Screening Form, “Editorial Changes to Emergency Plan Sections G and H” Screen; June 29, 2016
- PCR 2126248 and 2126257; 10 CFR 50.54(q) Evaluation Form, “Editorial Changes to Emergency Plan Sections G and H” Evaluation; June 29, 2016
- PCR 2148555, 2093308, 2148561 and 2148562; 10 CFR 50.54(q) Screening Form, “Revise Emergency Plan and Procedure Changes for Identifying and Activating the Alternate Emergency Operations Facility” Screen and Evaluation; September 14, 2016
- PCR 2148555, 2093308, 2148561 and 2148562; 10 CFR 50.54(q) Evaluation Form, “Revise Emergency Plan and Procedure Changes for Identifying and Activating the Alternate Emergency Operations Facility” Evaluation; September 14, 2016

- PCR 2126262 and 2126183; 10 CFR 50.54(q) Screening Form, “Revisions of Emergency Plan Section N and EPIP 6.1” Screen; June 18, 2016
- PCR 2126262 and 2126183; 10 CFR 50.54(q) Evaluation Form, “Revisions of Emergency Plan Section N and EPIP 6.1” Evaluation; June 18, 2016
- PCR 2194687; 10 CFR 50.54(q) Screening Form, “EPIP Form PAR-01 Revision” Screen; April 26, 2017
- PCR 2167080 and 2191249; 10 CFR 50.54(q) Screening Form, “Revise Emergency Plan, Section F, “Emergency Communications” Screen; March 30, 2017
- PCR 2179494 and 2194525; 10 CFR 50.54(q) Evaluation Form, “Revise Emergency Plan, Section F, “Emergency Communications” Evaluation; March 30, 2017
- PCR 2179494 and 2194525; 10 CFR 50.54(q) Screening Form, “Revise EPIP 4.3, Subsection 3.5, and Emergency Plan Section B” Screen; March 23, 2017
- PCR 2179494 and 2194525; 10 CFR 50.54(q) Evaluation Form, “Revise EPIP 4.3, Subsection 3.5” Evaluation, and Emergency Plan Section B” Evaluation; March 23, 2017
- PCR 2169498; 10 CFR 50.54(q) Screening Form, “Revise Emergency Plan, Section I, Figure I-1” Screen; April 6, 2017
- PCR 2169498; 10 CFR 50.54(q) Evaluation Form, “Revise Emergency Plan, Section I, Figure I-1” Evaluation; April 6, 2017
- CR 02148555; DAEC Plan “H” – Emergency Facilities Staffing; August 16, 2016
- CR 02167080; Replacement of 800 MHZ Radio System; November 2, 2016
- CR 02169498; EPZ Boundary Change; November 15, 2016
- CR 02188898; Failed RSPS – PAR Declaration; March 1, 2017
- CR 02194525; DAEC Plan “B” – Emergency Response Organization; March 28, 2017

<u>2RS1 Radiological Hazard Assessment and Exposure Controls</u>

- RP-AA-103-1001; Posting Requirements for Radiological Hazards; Revision 4
- RP-AA-107-1004; Procedure for Radioactive Source Controls and Leak Checking; Revision 0
- STP NS999901; Sealed Source Leak Test; July 10, 2017
- JL Shepherd & Associates Leak Test/Contamination Check Certification; March 13, 2017
- Survey PDA-M-20170831-2; RM1, New JL Shepherd Survey; August 31, 2017
- Ni-63 Smear Analysis; July 19, 2017
- RB131 Surveys; October 24-27, 2016
- Recirculation Pump Replacement Surveys; October 14-27, 2016
- RWCU Surveys; November 10-28, 2017
- RWP 16-4080; Main Steam Isolation Valve in Support of Drywell Refuel Outage 25; Revision 00
- RWP 16-4060; PSV and MSRV Maintenance in Support of Refuel Outage 25; Revision 00
- RWP 16-4140; Recirc Pump Seal Work in Support of Drywell Refuel Outage 25; Revision 01
- RWP 17-0116; Non-Routine Maintenance: Valves, Pumps and Pipe Replacement; Revision 00
- HPP 3103.01; HP Survey Performance and Frequencies; Revision 48
- CR 02162965; Diver Dosimetry System has Intermittent Faults; October 16, 2016
  CR 02164457; Level 1 Assessment – RP Tagging and Postings; October 22, 2016
- Radiation Protection Work Plan; HPCI/RCIC Runs; August 23, 2016
- CR 02232776; RP Technician Improperly Applied HRA Controls to Steam Tunnel Entry; October 25, 2017
- RWP 17-0210 and Associated ALARA Documents; RCIC Run Inspections: High Radiation Areas; Multiple Dates
- RP-AA-103-1002; High Radiation Area Controls; Revision 7
- Survey PDA-M-20170221-3; RB061 DW017 DW023 – RCIC Run; February 28, 2017
- Survey PDA-M-201750226-2; RB061 DW017 DW023 – RCIC Run; February 26, 2015

- RWP 17-0116 and Associated ALARA Documents; Non-Routine Maintenance: Valves, Pumps and Pipe Replacement; Multiple Dates
- RWP 17-4180 and Associated ALARA Documents; M1: Mechanical Valve Maintenance and Repair in Support of Drywell – Forced Outage; Multiple Dates

2RS2 Occupational ALARA Planning and Controls

- RWP 16-4080 and Associated ALARA Documents; Main Steam Isolation Valve in Support of Drywell Refuel Outage 25; Multiple Dates
- RWP 16-4060 and Associated ALARA Documents; PSV and MSRV Maintenance in Support of Refuel Outage 25; Multiple Dates
- RWP 16-4140 and Associated ALARA Documents; Recirc Pump Seal Work in Support of Drywell Refuel Outage 25; Multiple Dates
- RP-AA-104-1000; ALARA Implementing Procedure; Revision 13
- CR 02205931; Level 1 Assessment – ALARA Deep Dive
- CR 02205900; Inconsistencies between Outage RWPs and ALARA Plans; May 18, 2017
- CR 02175874; ALARA Roll-Up CR from RFO25; December 19, 2016

2RS5 Radiation Monitoring Instrumentation

- HPP 3110.72; Calibration of the FASTSCAN Whole Body Counter; Revision 5
- AT 02216097; Questions to Address with Whole Body Counter Calibration; July 19, 2017
- Whole Body Counter Calibration Report; Advanced Measurement Technology Inc.; October 3, 2017
- CR 02228882; Level 1 Assessment – Whole Body Counter CE Validation; October 5, 2017

4OA1 Performance Indicator Verification

- ACP 1402.4; NRC, WANO & MOPR Performance Indicator Reporting; Revision 22
- NRC PI Data Calculation, Review And Approval; Safety System Functional Failures; Fourth Quarter 2016; January 11, 2017
- NRC PI Data Calculation, Review And Approval; Safety System Functional Failures; First Quarter 2017; April 17, 2017
- NRC PI Data Calculation, Review And Approval; Safety System Functional Failures; Second Quarter 2017; July 20, 2017
- NRC PI Data Calculation, Review And Approval; Safety System Functional Failures; Third Quarter 2017; October 9, 2017
- NRC PI Data Calculation, Review And Approval; MSPI Residual Heat Removal System, and MSPI Cooling Water System 1; Fourth Quarter 2016; January 11, 2017
- NRC PI Data Calculation, Review And Approval; MSPI Cooling Water System 2; Fourth Quarter 2016; January 11, 2017
- NRC PI Data Calculation, Review And Approval; MSPI Residual Heat Removal System, and MSPI Cooling Water System 1; First Quarter 2017; April 7, 2017
- NRC PI Data Calculation, Review And Approval; MSPI Cooling Water System 2; First Quarter 2017; April 7, 2017
- NRC PI Data Calculation, Review And Approval; MSPI Residual Heat Removal System, and MSPI Cooling Water System 1; Second Quarter 2017; July 17, 2017
- NRC PI Data Calculation, Review And Approval; MSPI Cooling Water System 2; Second Quarter 2017; July 17, 2017
- NRC PI Data Calculation, Review And Approval; MSPI Residual Heat Removal System, and MSPI Cooling Water System 1; Third Quarter 2017; October 8, 2017

- NRC PI Data Calculation, Review And Approval; MSPI Cooling Water System 2; Third Quarter 2017; October 8, 2017
- Reactor Oversight Process: NRC PI Data Calculation, Review and Approval; RCS Activity (RCSA); Various Dates
- Reactor Oversight Process: NRC PI Data Calculation, Review and Approval; Occupational Exposure Effectiveness; Various Dates

4OA2 Problem Identification and Resolution

- CR 02204192; Communication Error During Notification Process Of Table Top
- CR 02236101; Low Level Trend In Reactor Sulfates
- CR 02236241; Routine Survey Reveal Higher Than Normal Dose Rates
- CR 02236258; Potential Flowpath From 'B' RWCU To Phase Separators
- CR 02233094; 45% Recirc Pump A&B Runback
- CR 02235465; Use Of Robust Barrier When In Single Reactor Feed Pump
- CR 02232776; RPT Improperly Applied HRA Controls for Steam Tunnel Entry
- CR 02233479; Clean Area Contamination
- CR 02232265; 3 Workers Receive Internal Deposition
- CR 02234985; Level 1 Assessment – Potential Trend RP Performance

4OA5 Other

- 2015 Annual Radiological Environmental Operating Report
- 2016 Annual Radiological Environmental Operating Report
- 70-007-0012; Duane Arnold Energy Center Fire Hazards Analysis for the Dry Spent Fuel
- Program; Revision 0
- 72.48 Screening 72-127; August 16, 2004
- 72.48 Screening 72-144; November 4, 2004
- 72.48 Screening 72-145; December 16, 2004
- 72.48 Screening 72-153; December 14, 2004
- 72.48 Screening 72-168; February 24, 2005
- 72.48 Screening 72-171; May 3, 2005
- 72.48 Screening 72-172; May 3, 2005
- 72.48 Screening 72-195; May 16, 2007
- 72.48 Screening 72-197; May 11, 2007
- 7248SCRN-9857; 10 CFR 72.48 Screening of EC 271080
- 7248SCRN-10815; February 20, 2013
- 7248SCRN-10864; April 9, 2013
- ACP 118.0; Conduct of the Duane Arnold Energy Center On-Site Dry Spent Fuel Storage Program; Revision 21
- ACP 1412.2; Control of Combustibles; Revision 48, 49
- AOP 901; Earthquake; Revision 31
- AOP 903; Severe Weather; Revision 59
- AOP 914; Security Events; Revision 63
- CR 02200005; Gap at Bottom of ISFSI Security Fences
- CR 02205703; ISFSI Supplier Records Not Found in QA Records
- CR 02228019; High Risk Activity Not Identified Until T-0
- CR 02228348; 2017 NRC ISFSI: 10 CFR 72.104 and 10 CFR 190 Compliance
- CR 02228514; 2017 NRC ISFSI Combustible Control
- CR 02228558; ACP 1412.2 – Control of Combustibles

- DBD-F16-001; Duane Arnold Energy Center Design Basis Document for the Dry Spent Fuel Storage Program; Revision 16
- EAL-01; Duane Arnold Energy Center Emergency Action Level Matrix; Revision 10
- EAL-02; Duane Arnold Energy Center Emergency Action Level Matrix; Revision 9
- EBD-E; EAL Bases Document: ISFSI Abnormal Events Category; Revision 2
- ER-AA-201-2001; System and Program Health Reporting; Revision 12
- FPL-1; Quality Assurance Topical Report; Revision 21
- HPP 3103.01; HP Survey Performance and Frequencies; Revision 47
- HPP 3104.13; Dry Cask Storage Job Coverage and Decontamination; Revision 13
- ISFSI Quarterly Walkdown; August 28, 2017
- Level 1 Assessment for LICA 02169987 2017 NRC ISFSI Inspection Self-Assessment; August 31, 2017
- LI-AA-101-1002; 10 CFR 72.48 Changes, Tests, and Experiments; Revision 04
- NUH-003; Updated Final Safety Analysis Report for the Standardized NUHOMS Horizontal Modular Storage System for Irradiated Nuclear Fuel; Revision 6, 7, 8, 9, 10, 11
- OI 581; HSM Temperature Monitoring; Revision 6
- OP-031; Operations Department Weekly Housekeeping Area Inspection; Revision 24
- PDA 17-002; Independent Spent Fuel Storage Installation Audit; June 1, 2017
- PR-AA-100-1003; Independent Spent Fuel Storage Installation / Dry Fuel Storage Project Management; Revision 4
- STP 3.0.0-01; Surveillance Test Procedure Instrument Checks; Revision 162
- STP NS810101; ISFSI – HSM Bird Screen Inspection; Revision 0
- SY-AA-102-1021; Conduct of Personnel Searches; Revision 9

LIST OF ACRONYMS USED

ACP	Administrative Control Procedure
CAP	Corrective Action Program
CBC	Control Building Chiller
CFR	*Code of Federal Regulations*
CoC	Certificate of Compliance
CR	Condition Report
DAEC	Duane Arnold Energy Center
DEP	Drill Exercise Performance
DSC	Dry Shielded Canister
EAL	Emergency Action Level
EP	Emergency Preparedness
FCN	FSAR Change Notice
FSAR	Final Safety Analysis Report
HPCI	High Pressure Core Injection
HSM	Horizontal Storage Module
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IPOI	Integrated Plant Operating Procedure
ISFSI	Independent Spent Fuel Storage Installation
JPM	Job Performance Measure
LORT	Licensed Operator Requalification Training
MSIV	Main Steam Isolation Valve
MSPI	Mitigating Systems Performance Index
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NRC	U.S. Nuclear Regulatory Commission
PA	Protected Area
PI	Performance Indicator
SAT	Systems Approach to Training
SDP	Significance Determination Process
TIP	Traversing Incore Probe
TN	Transnuclear
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
WO	Work Order