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

August 11, 2016

EA-16-174

Mr. Timothy S. Rausch
President and Chief Nuclear Officer
Susquehanna Nuclear, LLC
769 Salem Blvd - NUCSB3
Berwick, PA 18603-0467

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION – INTEGRATED INSPECTION REPORT 05000387/2016002 AND 05000388/2016002

Dear Mr. Rausch:

On June 30, 2016, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Susquehanna Steam Electric Station (SSES), Units 1 and 2. The enclosed report documents the inspection results, which were discussed on July 8, 2016 with you and other members of your staff.

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

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

Separately, a violation involving a failure to set secondary containment during operations with a potential for draining the reactor vessel (OPDRVs) was identified during the Unit 1 refueling outage. Specifically, from March 16, 2016 to April 11, 2016, while all other Technical Specifications (TSs) were met, Susquehanna conducted several OPDRVs without establishing secondary containment operability, which is a violation of TS 3.6.4.1, "Secondary Containment." NRC issued Enforcement Guidance Memorandum (EGM) 11- 003, "Enforcement Guidance Memorandum on Dispositioning Boiling Water Reactor (BWR) Licensee Noncompliance with TS Containment Requirements during Operations with a Potential for Draining the Reactor Vessel," on October 4, 2011, allowing for the exercise of enforcement discretion for such OPDRV-related TS violations, when certain criteria are met. Because the NRC has determined the licensee has met the criteria and the violations occurred during the discretion period described in the EGM, the NRC is exercising enforcement discretion and will not issue enforcement action for these violations. The EGM, which was most recently revised on January 15, 2016, also requires that licensees receiving discretion must submit a license amendment request (LAR) to accept the NRC's generic change to the Standard TS that will allow a graded approach to OPDRV requirements. The LAR must be submitted within twelve months of NRC publication of the generic change in the Federal Register.

If you contest the non-cited violations in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at SSES. In addition, if you disagree with the cross-cutting aspect assigned to any finding, 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 Regional Administrator, Region I, and the NRC Resident Inspector at SSES.

In accordance with Title 10 of the *Code of Federal Regulations* (CFR) 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC's Public Document Room or from the Publicly Available Records component of the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC's website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Michael L. Scott, Director
Division of Reactor Projects

Docket Nos. 50-387 and 50-388
License Nos. NPF-14 and, NPF-22

Enclosure:
Inspection Report 05000387/2016002
and 05000388/2016002 w/Attachment
Supplementary Information

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Letter to Mr. Timothy Rausch from Michael L. Scott, dated August 11, 2016.

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION – INTEGRATED INSPECTION REPORT 05000387/2016002 AND 05000388/2016002

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REGION I

Docket Nos.:	50-387 and 50-388
License Nos.:	NPF-14 and NPF-22
Report No.:	05000387/2016002 and 05000388/2016002
Licensee:	Susquehanna Nuclear, LLC (Susquehanna)
Facility:	Susquehanna Steam Electric Station, Units 1 and 2
Location:	Berwick, Pennsylvania
Dates:	April 1, 2016 through June 30, 2016
Inspectors:	J. Greives, Senior Resident Inspector T. Daun, Resident Inspector L. Dumont, Reactor Inspector C. Graves, Health Physicist P. Meier, Project Engineer N. Embert, Operations Engineer E. H. Gray, Senior Reactor Inspector
Approved By:	Daniel L. Schroeder, Chief Reactor Projects Branch 4 Division of Reactor Projects

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SUMMARY

IR 05000387/2016002, 05000388/2016002; April 1, 2016 to June 30, 2016; Susquehanna Steam Electric Station, Units 1 and 2; Maintenance Effectiveness, Surveillance Testing, Drill Evaluation, Radiological Hazard Assessment and Exposure Controls, and Follow-Up of Events and Notices of Enforcement Discretion.

This report covered a three-month period of inspection by resident inspectors and announced baseline inspections performed by regional inspectors. The inspectors identified seven non-cited violations (NCVs), all of which were of very low safety significance (Green and/or Severity Level IV). The significance of most findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, “Significance Determination Process (SDP)”, dated April 29, 2015. Cross-cutting aspects are determined using IMC 0310, “Aspects Within Cross-Cutting Areas,” dated December 4, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC’s Enforcement Policy, dated February 4, 2015. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NRC technical report designation (NUREG) - 1649, “Reactor Oversight Process,” Revision 5.

Cornerstones: Mitigating Systems

- Green. A self-revealing finding of very low safety significance (Green) and associated NCV of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” for failure to correct a condition adverse to quality. Specifically, on March 23, 2016, the ‘A’ emergency diesel generator (EDG) failed its technical specification (TS) surveillance test in that the emergency switchgear room cooler, 1V222A, started immediately when the EDG loaded onto the emergency bus following a simulated loss of off-site power (LOOP) and simulated Emergency Core Cooling System (ECCS) Initiation, rather than sequencing onto the bus as intended by design. Susquehanna identified the direct cause of the failure was due to a misadjustment of the mechanism-operated cell (MOC) linkage switch (S1) in the ‘A’ EDG output breaker to the 1A 4 kilovolt (kV) bus, which provides the electrical logic to the 1V222A load timer. The repeat failure was entered into the corrective action program (CAP) as CR-2016-08643, the MOC linkage was realigned, and the functions satisfactorily tested.

  The finding was determined to be more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the failure to correct the degraded condition rendered the ‘A’ EDG inoperable for longer than the TS allowed outage time. In accordance with IMC 0609.04, “Initial Characterization of Findings,” dated June 19, 2012, and Exhibit 2 of IMC 0609, Appendix A, “The SDP for Findings At-Power,” dated June 19, 2012, the inspectors determined that this finding required a detailed risk assessment because the finding represents an actual loss of function of a single train for greater than the TS allowed outage time. Specifically, the ‘A’ EDG was inoperable from July 19, 2010 until April 2, 2016, because TS requires functioning of the sequencing timers for the EDG to be operable. In coordination with a Region 1 Senior Risk Analyst, the issue was qualitatively screened as Green (very low safety significance) based on the low initiating event frequency associated with a loss of coolant accident (LOCA) co-incident with a LOOP event, and observed successful EDG function during multiple LOOP/LOCA tests over the period in question.

This would result in a delta core damage frequency substantially less than E-6. Additionally, it was reasonable to conclude that the 'A' EDG remained available to perform its function given the minimal increased load on the machine as evidenced during the performance of the LOOP-LOCA surveillance testing in 2012, 2014, and 2016.

This finding had a cross-cutting aspect in the area of Problem Identification and Resolution, Evaluation, because Susquehanna did not thoroughly evaluate the issue to ensure that the resolution addressed the cause and extent of conditions commensurate with their safety significance. Specifically, Susquehanna corrected a suspected condition without appropriate troubleshooting until the third identical failure of the 1V222A load timer. [P.2] (Section 1R12)

- Green. A self-revealing finding of very low safety significance (Green) and associated NCV of TS 5.4.1.a, "Procedures," was identified when Susquehanna failed to implement procedures for loading EDGs promptly following extended unloaded operation. Specifically, Susquehanna did not load the 'B' EDG promptly following over 6 hours of unloaded operation which resulted in the slow starting time during the subsequent surveillance test due to insufficient fuel delivery caused by clogged fuel injectors. The failure was entered into the CAP as CR-2016-13220 and the EDG was run loaded for an extended period to ensure any unburned fuel had been removed from the machine.

  The finding was more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems Cornerstone and affected the objective to ensure the reliability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, the failure to load the 'B' EDG following extended operation unloaded resulted in the slow starting time of the EDG during subsequent surveillance testing due to clogged fuel injectors. The inspectors evaluated the finding in accordance with Exhibit 2 of IMC 0609, Appendix A, "The SDP for Findings At-Power," dated June 19, 2012 and determined that it was of very low safety significance (Green) because it did not affect the design or qualification of the EDG, did not represent a loss of system function, and did not represent a loss of a single train for greater than its TS allowed outage time. The finding is related to the cross-cutting area of Human Performance, Consistent Process, because Susquehanna did not use a consistent, systematic approach to make decisions which incorporated risk insights. Specifically, Susquehanna did not appropriately coordinate the loaded run of the 'B' EDG with maintenance on the 'C' EDG to ensure 'B' EDG availability was not unnecessarily challenged. [H.13] (Section 1R22)

- Green. An NRC-identified finding of very low safety significance (Green) and associated NCV of TS 5.4.1.a, "Procedures," was identified when Susquehanna failed to implement procedures for controlling the high pressure coolant injection (HPCI) system. Specifically, operators overrode automatic initiation of the system prior to inserting a manual scram, contrary to the requirements of OP-252-001, "HPCI System," and OP-AD-300, "Administration of Operations." This was entered into the CAP as CRs 2016-12854 and 2016-13118 and 2016-13136, the operator's involved in the event were remediated, and lessons learned communicated to other station personnel.

  The finding was more than minor because it was associated with the Human Performance attribute of the Mitigating Systems Cornerstone and affected the objective to ensure the availability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, overriding the HPCI system prior to initiating a plant scram rendered the system unavailable to respond to a level transient or failure of the non-safety related feedwater system. The inspectors evaluated the finding in accordance with Exhibit 2 of IMC 0609, Appendix A, "The SDP for Findings At-Power," dated June 19, 2012 and determined that it required a detailed risk assessment because it represented a loss of the single train system's function. The Region 1 SRA performed a detailed risk evaluation using the Susquehanna Unit 2 standardized plant analysis risk (SPAR) Model, version 8.23. The issue was conservatively modeled with a HPCI failure to start due to the system automatic start signal being overridden.

The change in core damage frequency per year was determined to be in the E-10 range due to the very short duration the system auto start feature was defeated. Therefore the issue was determined to be of very low safety significance (Green). The finding is related to the cross-cutting area of Human Performance, Procedure Adherence because Susquehanna did not follow processes, procedures and work instructions. Specifically, operators did not ensure that their actions were appropriately authorized by procedures when taking action to override a key safety system prior to a plant transient. [H.8] (Section 4OA3)

Cornerstone: Barrier Integrity

- Green. A self-revealing Green finding and associated violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," and TS 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," was identified when Susquehanna did not promptly identify a condition adverse to quality. Despite observing abnormal behavior during local leak rate testing following replacement in May 2014, Susquehanna did not take any action to ensure that certain Reactor Water Cleanup (RWCU) system PCIVs passed their subsequent testing. Consequently, these valves failed their in-service and local leak rate test in March 2016 when they failed to close upon securing system flow. The failure was caused by an internal interference between the check valve hinge and body. Following the failures in March 2016, Susquehanna repaired the valves and successfully performed local leak rate testing, restoring operability of the PCIVs. The repeat failure was entered into the CAP as CRs 2016-06960 and 2016-09940.

  The finding was determined to be more than minor because it was associated with the Structure, System, and Component (SSC) and Barrier Performance attribute of the Barrier Integrity cornerstone and adversely affected the cornerstone objective of providing reasonable assurance that physical design barriers (containment) protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to identify a condition adverse to quality during post-maintenance testing resulted in two PCIVs being rendered inoperable for longer than the TS allowed outage time. In accordance with IMC 0609.04, "Initial Characterization of Findings," dated June 19, 2012, and Exhibit 2 of IMC 0609, Appendix A, "The SDP for Findings At-Power," dated June 19, 2012, the inspectors determined that this finding is of very low safety significance (Green) because the performance deficiency did not involve the hydrogen recombiners and did not result in an actual open pathway in the physical integrity of reactor containment. Specifically, the redundant valve for each penetration remained operable during the period in which these two valves were inoperable. This finding had a cross-cutting aspect in the area of Human Performance, Conservative Bias, because Susquehanna did not use decision making practices that emphasized prudent choices over those that are simply allowable. Specifically, Susquehanna decided to accept elevated seat leakage for two new PCIVs, assuming that they could be declassified as PCIVs. [H.14] (Section 1R12)

Cornerstone: Emergency Preparedness

- Green. An NRC-identified finding of very low safety significance (Green) and associated NCV of 10 CFR 50.54(q)(2),"Emergency Plans" was identified when Susquehanna failed to identify that an incorrect notification of wind direction was made to the senior state official (SSO) during a full-scale drill. This failure was entered into the CAP as CRs 2016-14303 and 2016-14128, ERO personnel involved in the incorrect communication and the drill controllers that failed to identify the deficiency were remediated, and lessons learned communicated to other emergency response organization personnel.

  The finding was more than minor because it is associated with the emergency response organization (ERO) Performance attribute of the Emergency Preparedness Cornerstone and affected the cornerstone objective to ensure that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. Specifically, the failure of Susquehanna personnel to effectively identify an exercise weakness associated with a risk significant planning standard (RSPS) caused a missed opportunity to identify and correct a drill-related performance deficiency. The inspectors assessed the issue using the Emergency Preparedness SDP, Appendix B to IMC 0609, dated September 23, 2014.

  Susquehanna's failure to critique the inaccurate notification met the NRC's definition of a weakness in a full-scale drill. However, because four previous notifications had accurately reported the wind direction and the miscommunication was inconsistent with the correct protective actions recommendation (PAR) that was communicated simultaneously, in consultation with a senior emergency preparedness inspector, inspectors determined the communication would likely have been corrected prior to the offsite response organizations (OROs) acting on the incorrect information, did not result in an incorrect PAR, and therefore determined that that the failure to critique the drill weakness only constituted a degradation of the planning standard (PS) function. Therefore the finding is characterized as having very low safety significance (Green). The finding is related to the cross-cutting area of Problem Identification and Resolution, Identification, in that Susquehanna did not identify a RSPS issue completely, accurately, and in a timely manner commensurate with the safety significance. Specifically, during the full-scale drill, Susquehanna failed to recognize and critique that a RSPS was not met and did not place this issue into the CAP until prompted by inspectors. [P.1] (Section 1EP6)

Cornerstone: Occupational and Public Radiation Safety

- Green. A Green self-revealing NCV of TS 5.7.1, High Radiation Area Controls, was identified when a worker did not comply with a radiological posting barrier and other access control requirements for high radiation area (HRA) entry. Specifically, on December 26, 2015, a security officer entered into a posted HRA without proper authorization. This was entered into the CAP as CR-2015-33947, the HRA barrier was moved further out, and a shield rack was placed in front of the condenser bay door to reduce radiation dose rates.

  The finding was determined to be more than minor based on similarity to example 6.h in IMC 0612, Appendix E, and it is associated with Human Performance attribute of the Occupational Radiation Safety Cornerstone and affected the cornerstone objective to ensure adequate protection of the worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. Specifically, the individual violated the HRA posting, radiation work permit (RWP) and briefing requirements designed to protect the worker from unnecessary radiation exposure. Using IMC 0609, Appendix C, "Occupational Radiation Safety SDP," dated August 19, 2008, the finding was determined to be of very low safety significance (Green) because it did not involve: (1) as low as is reasonably achievable (ALARA) occupational collective exposure planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. The finding is related to the cross-cutting area of Problem Identification and Resolution, Resolution, in that the organization did not ensure that corrective actions to address the cause of repetitive electronic dosimeter alarms in this area of the plant and had not been sufficiently evaluated and had not enhanced radiological controls to prevent this issue from recurring. [P.3] (Section 2RS1)

- Green. A Green self-revealing NCV of TS 5.7.2, High Radiation Area Controls, was identified when workers entered the wrong reactor unit condenser bay (Unit 2) that was posted and controlled as a locked high radiation area (LHRA). Specifically, on May 3, 2016, four Susquehanna staff were briefed to enter the Unit 1 condenser bay to check for steam leaks during start up, however the staff entered the Unit 2 condenser bay during full power operations in error and received electronic dosimeter alarms. This was entered into the CAP as CR-2016-11944, the use of master keys for routine entry into LHRA was discontinued, and a radiation safety stand down was conducted.

  The finding was determined to be more than minor based on a similar example 6.h in IMC 0612, Appendix E, and it is associated with Human Performance attribute of the Occupational Radiation Safety Cornerstone and affected the cornerstone objective to ensure adequate protection of the worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. Specifically, Susquehanna staff violated the RWP and briefing requirements designed to protect workers from unnecessary radiation exposure. Using IMC 0609, Appendix C, "Occupational Radiation Safety SDP," dated, August 19, 2008, the finding was determined to be of very low safety significance (Green) because it did not involve: (1) ALARA occupational collective exposure planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. The finding was self-revealing because Susquehanna was made aware of the situation as a result of an electronic dose rate alarm. The finding is related to the cross-cutting area of Human Performance, Teamwork because the workers did not conduct peer checking and recognize and communicate that they were in the wrong reactor unit for the work they were conducting. Specifically, four Susquehanna staff were briefed to enter the Unit 1 condenser bay to check for steam leaks during start up, however the staff entered the Unit 2 condenser bay. [H.4] (Section 2RS1)

REPORT DETAILS

Summary of Plant Status

Unit 1 began the inspection period in mode 5 for the 1R19 refueling and maintenance outage. Following the completion of refueling and maintenance activities, operators commenced a reactor startup on April 21, 2016. During startup activities, a hydrogen seal ring on the main turbine generator failed which required a shutdown on April 23, 2016 for repairs. Following repairs to the hydrogen seal ring, operators commenced a reactor startup on May 2, 2016 and achieved 100 percent power on May 10, 2016. On May 13, 2016, operators reduced power to approximately 70 percent to perform a rod pattern adjustment. Full power was achieved again on May 15, 2016. Operators maintained the unit at or near 100 percent power until June 5, 2016 when they commenced a planned shutdown to investigate elevated unidentified leakage in the drywell. Following the completion of the maintenance activity, operators commenced a reactor startup on June 26, 2016. Operators returned the unit to 100 percent power on June 30, 2016.

Unit 2 began the inspection period at 100 percent power. On April 7, 2016, operators reduced power to approximately 70 percent to perform a rod sequence exchange and returned to full power on April 9, 2016. On April 29, 2016, operators reduced power to 78 percent to clean the condenser waterboxes. Operators returned the unit to 100 percent power on May 1, 2016. On May 12, 2016, operators inserted a reactor recirculation pump (RRP) runback to approximately 70 percent due to an electrical transient that resulted in a loss of drywell cooling. On May 13, 2016 operators inserted a manual reactor scram when drywell cooling could not be restored. Following the completion of repairs, operators commenced a reactor startup on May 16, 2016. Operators returned the unit to 100 percent power on May 18, 2016, and remained at or near 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01 – 1 sample)

.1 Summer Readiness of Offsite and Alternate Alternating Current (ac) Power Systems

a. Inspection Scope

The inspectors performed a review of plant features and procedures for the operation and continued availability of the offsite and alternate ac power system to evaluate readiness of the systems prior to seasonal high grid loading. The inspectors reviewed Susquehanna's procedures affecting these areas and the communications protocols between the transmission system operator and Susquehanna. This review focused on changes to the established program and material condition of the offsite and alternate ac power equipment. The inspectors assessed whether Susquehanna established and implemented appropriate procedures and protocols to monitor and maintain availability and reliability of both the offsite ac power system and the onsite alternate ac power system. The inspectors evaluated the material condition of the associated equipment by interviewing the responsible system engineer, reviewing condition reports and open work orders, and walking down portions of the offsite and ac power systems including the 500 kV, 230 kV, and T-10 switchyards.

b. Findings

No findings were identified.

1R04 Equipment Alignment

.1 Partial System Walkdowns (71111.04 – 3 samples)

a. Inspection Scope

The inspectors performed partial walkdowns of the following systems:

- Unit 1, fuel pool cooling during common residual heat removal (RHR) out-of-service window on April 1, 2016
- Unit 1, reactor core isolation cooling (RCIC) following turbine internal inspection on May 2, 2016
- Common, 'A' EDG while 'E' EDG unavailable for substitution on June 29, 2016

The inspectors selected these systems based on their risk-significance relative to the reactor safety cornerstones at the time they were inspected. The inspectors reviewed applicable operating procedures, system diagrams, the Updated Final Safety Analysis Report (UFSAR), TS, work orders, CRs, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have impacted the system's performance of its intended safety functions. The inspectors also performed field walkdowns of accessible portions of the systems to verify system components and support equipment were aligned correctly and were operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no deficiencies. The inspectors also reviewed whether Susquehanna staff had properly identified equipment issues and entered them into the CAP for resolution with the appropriate significance characterization.

b. Findings

No findings were identified.

.2 Full System Walkdown (71111.04S – 1 sample)

a. Inspection Scope

On June 13, 2016, the inspectors performed a complete system walkdown of accessible portions of the emergency service water system to verify the existing equipment lineup was correct. The inspectors reviewed operating procedures, surveillance tests, drawings, equipment line-up check-off lists, and the UFSAR to verify the system was aligned to perform its required safety functions. The inspectors also reviewed electrical power availability, component lubrication and equipment cooling, hanger and support functionality, and operability of support systems. The inspectors performed field walkdowns of accessible portions of the systems to verify as-built system configuration matched plant documentation, and that system components and support equipment remained operable.

The inspectors confirmed that systems and components were aligned correctly, free from interference from temporary services or isolation boundaries, environmentally qualified, and protected from external threats.

The inspectors also examined the material condition of the components for degradation and observed operating parameters of equipment to verify that there were no deficiencies. Additionally, the inspectors reviewed a sample of related CRs and work orders to ensure Susquehanna appropriately evaluated and resolved any deficiencies.

b. Findings

No findings were identified.

1R05 Fire Protection

.1 Resident Inspector Quarterly Walkdowns (71111.05Q – 5 samples)

a. Inspection Scope

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

- Common, 'E' EDG building (fire zone 0-41E) on May 19, 2016
- Unit 1, drywell (fire zone 1-4F) on June 6, 2016
- Common, 'A' EDG (fire zone 0-41A) on June 29, 2016
- Unit 1, access corridor (fire zone 1-2B) on June 30, 2016
- Unit 1, standby liquid control area (fire zone 1-5A-S) on June 30, 2016

b. Findings

No findings were identified.

1R06 Flood Protection Measures (71111.06 – 1 sample)

.1 Internal Flooding Review

a. Inspection Scope

The inspectors reviewed the UFSAR, the site flooding analysis, and plant procedures to identify internal flooding susceptibilities for the site. The inspectors review focused on the 749' elevation of the Unit 2 reactor building, which includes the 'A' and 'B' safety-related switchgear rooms, both divisions of reactor protection system instrumentation and the standby liquid control system. It verified the adequacy of equipment seals located below the flood line, floor and water penetration seals, watertight door seals, common drain lines and sumps, sump pumps, level alarms, control circuits, and temporary or removable flood barriers. It assessed the adequacy of operator actions that Susquehanna had identified as necessary to cope with flooding in this area and also reviewed the CAP to determine if Susquehanna was identifying and correcting problems associated with both flood mitigation features and site procedures for responding to flooding.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program and Licensed Operator Performance (71111.11Q – 3 samples)

.1 Quarterly Review of Licensed Operator Requalification Testing and Training

a. Inspection Scope

The inspectors observed licensed operator simulator training on May 24, 2016, which included a loss of control room annunciators followed by high vibrations on the main turbine and resultant reactor scram. Following the reactor scram, a failure of the ‘A’ reactor recirculation piping produced a LOCA in the drywell. The inspectors evaluated operator performance during the simulated event and verified completion of risk significant operator actions, including the use of abnormal and emergency operating procedures. The inspectors assessed the clarity and effectiveness of communications, implementation of actions in response to alarms and degrading plant conditions, and the oversight and direction provided by the control room supervisor. The inspectors verified the accuracy and timeliness of the emergency classification made by the shift manager and the TS action statements entered by the unit supervisor. Additionally, the inspectors assessed the ability of the crew and training staff to identify and document crew performance problems.

b. Findings

No findings were identified.

.2 Quarterly Review of Licensed Operator Performance in the Main Control Room (2 samples)

a. Inspection Scope

On April 21, 2016, inspectors observed the control room operators perform a planned reactor startup from the Unit 1 refueling outage. Additionally, on May 16, 2016, inspectors observed the control room operators perform a reactor startup following a Unit 2 reactor scram. The inspectors observed the reactivity control briefing to verify that it met the criteria specified in OP-AD-002, “Standards for Shift Operations,” Revision 57, OP-AD-300, “Administration of Operations,” Revision 5, and OP-AD-338, “Reactivity Manipulations Standards and Communication Requirements,” Revision 31.

The inspectors observed the crews during the evolutions to verify that procedure use, crew communications, control board component manipulations, and coordination of activities in the control room met established standards.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12Q – 3 samples)

a. Inspection Scope

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

- Unit 1, 1X210 and 1X230 doble testing and inspection on March 29, 2016 and March 31, 2016
- Unit 1, MOC switch failure on April 1, 2016
- Unit 1, repeat failures of feedwater flushing line PCIVs on April 13, 2016

b. Findings

.1 Introduction: A self-revealing Green finding and associated violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," and TS 3.6.1.3, "PCIVs," was identified when Susquehanna did not promptly identify a condition adverse to quality. Despite observing abnormal behavior during local leak rate testing following replacement in May 2014, Susquehanna did not take any action to ensure that certain RWCU system PCIVs passed their subsequent testing. Consequently, these valves failed their in-service and local leak rate test in March 2016 when they failed to close upon securing system flow. The failure was caused by an internal interference between the check valve hinge and body.

Description: 141F038A and 141F039B isolate the RWCU system from the 'A' and 'B' feedwater lines, respectively, and are credited for operation during a feedwater line break accident. In this case, these two valves provide the second isolation barrier from each feedwater line and would be required to function to automatically isolate primary containment if the redundant PCIVs, 141818A and 141818B, failed to close.

In May 2014, 141F039A and 141F039B were replaced as part of a modification to the RWCU system. Post-maintenance local leak rate testing indicated the valves were not seating as expected. In particular, 141F039A was unable to be pressurized initially following replacement and required several attempts to rapidly pressurize the test volume in an effort to get the valve to seat. After these several attempts, leakage was measured at 941 sccm. Additionally, seat leakage for 141F039B post-replacement was measured at 3950 sccm, in excess of its administrative limit of 2250 sccm. CR-2014-16323 was generated to evaluate the results and stated that the "new check valves in the RWCU return to feedwater appear to be defective." Susquehanna developed a technical decision making (TDM) document to review the possible options and determine a course of action. Options included 1) accept valves as-is, 2) perform additional testing, 3) perform an internal inspection of the valves and 4) accept valve as-is while pursuing declassifying these valves from PCIVs.

Susquehanna's evaluation determined that option 4 was the preferred option, in part because it was assumed that the valves were the third boundary for this primary containment penetration and could be declassified as PCIVs. If this assumption were incorrect, the preferred option would have been to perform an internal inspection of the valves, a choice that would have been consistent with the majority of stations consulted as part of the decision making process. At the time, however, the action to declassify the valves was assumed to be reasonable and no physical work was performed on the valves.

Actions were generated to administratively remove the valves from the program with an assigned due date of September 26, 2014. This action was extended a total of seven times to January 13, 2017. Its supporting engineering evaluation, which was extended three times and closed on May 2, 2016, identified that the valves were the second barrier credited in a feedwater line break under EC-059-1026, "Containment Isolation Design Requirements for the Feedwater Penetrations," and therefore were required to be credited as PCIVs.

In March 2016, during the subsequent biennial local leak rate test, both valves failed to close and were declared inoperable as PCIVs. Investigation identified an internal interference on both valves between hinge and valve body such that the disk failed to close when flow through the valve was secured. Susquehanna entered both failures into the CAP as CRs 2016-06960 and 2016-09940.

The inspectors determined that Susquehanna had a reasonable opportunity to identify a condition adverse to quality, associated with internal interference between the check valve hinge and body. However, in May 2014, Susquehanna did not implement their TDM process in an adequate manner that ensured the PCIVs would remain operable for the following operating cycle. Specifically, had Susquehanna engaged design engineering during the development of the TDM, declassifying the valves as PCIVs would likely not have been assumed as reasonable and, therefore, Susquehanna's decision making process would have directed an internal inspection of the valves at that time making it likely that the interference problem would have been discovered prior to their operability being required. Inspectors determined that this would have provided a reasonable opportunity to have identified the condition adverse to quality.

Analysis: The failure to identify and correct a condition adverse to quality in a timely manner, associated with PCIVs for the Unit 1 RWCU system, was a performance deficiency. The finding was determined to be more than minor because it was associated with the SSC and Barrier Performance attribute of the Barrier Integrity cornerstone and adversely affected the cornerstone objective of providing reasonable assurance that physical design barriers (containment) protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to identify a condition adverse to quality during post-maintenance testing resulted in two PCIVs being rendered inoperable for longer than their TS allowed outage time. In accordance with IMC 0609.04, “Initial Characterization of Findings,” dated June 19, 2012, and Exhibit 2 of IMC 0609, Appendix A, “The SDP for Findings At-Power,” dated June 19, 2012, the inspectors determined that this finding was of very low safety significance (Green) because the performance deficiency did not involve the hydrogen recombiners and did not result in an actual open pathway in the physical integrity of reactor containment. Specifically, the redundant valve for each penetration remained operable during the period in which these two valves were inoperable.

This finding had a cross-cutting aspect in the area of Human Performance, Conservative Bias, because Susquehanna did not use decision making practices that emphasized prudent choices over those that are simply allowable. Specifically, Susquehanna decided to accept elevated seat leakage for two new PCIVs, assuming that they could be declassified as PCIVs. [H.14]

Enforcement: 10 CFR 50, Appendix B, Criterion XVI, “Corrective Actions,” requires that measures be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. Contrary to this, from May 2014 through March 2016, Susquehanna failed to identify a condition adverse to quality, associated with internal interference between the check valve hinge and body of RWCU system PCIVs, in a timely manner.

TS 3.6.1.3 requires 141F039A and 141F039B, the PCIVs for the RWCU system, to be operable in Mode 1 and requires action be taken within 4 hours to isolate the respective containment penetration if either is determined to be inoperable. Contrary to this, both valves were inoperable from May 2014 through March 2016 without the containment penetration being isolated with a closed, deactivated valve. Following the failures in March 2016, Susquehanna repaired the valves and successfully performed local leak rate testing, restoring operability of the PCIVs.

Because it was of very low safety significance (Green) and has been entered into the CAP as CRs 2016-06960 and 2016-09940, this finding is being treated as a NCV in accordance with section 2.3.2 of the NRC’s Enforcement Policy.
(**NCV 05000387/2016002-01 Failure to Promptly Identify a Condition Adverse to Quality Associated with Primary Containment Isolation Valves**)

.2 Introduction: A self-revealing finding of very low safety significance (Green) and associated NCV of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” for failure to correct a condition adverse to quality was identified. Specifically, on March 23, 2016, the ‘A’ EDG failed its TS surveillance test because the emergency switchgear room cooler, 1V222A, started immediately when the EDG loaded onto the emergency bus following a simulated LOOP and simulated ECCS Initiation.

Susquehanna identified the direct cause of the failure was due to a misadjustment of the MOC linkage switch (S1) in the ‘A’ EDG output breaker to the 1A 4kV bus, which provides the electrical logic to the 1V222A load timer.

Description: Susquehanna Unit 1 and Unit 2 TS Limiting Condition of Operation (LCO) 3.8.1, “AC Sources- Operating,” requires, in part, four EDGs be operable in modes 1, 2, and 3. TS surveillance requirements (SRs) 3.8.1.11 and 3.8.1.18 verifies, in part, that the EDGs auto-start from a standby condition and energizes auto-connected loads through individual load timers. Table 3.8.1-1 of the TS basis lists the allowable value for the load timer of 1V222A as, greater than or equal to 54 seconds. SR 3.0.1 states, in part, that failure to meet a surveillance constitutes a failure to meet the LCO.

On March 23, 2016, the ‘A’ EDG failed its TS surveillance test because the 1V222A room cooler started immediately when the EDG loaded onto the emergency bus following a simulated LOOP and simulated ECCS initiation. In response to the failure, Susquehanna disabled the automatic initiation capability for 1V222A which restored operability to the ‘A’ EDG. Susquehanna initiated troubleshooting and on April 1, 2016 identified that the contacts on S1 were not properly made up.

This condition essentially bypassed the load timer for this load and allowed it to start immediately upon receiving a LOCA initiation signal. On April 2, 2016 adjustments were made to S1 to correct the misalignment of the contacts.

An evaluation was conducted and the same failure of the load timer for 1V222A was identified in May 2012 and April 2014. During these events, the load timer was replaced and the load timer tested by jumpering out the LOOP initiation logic and testing the time delay independently. Inspectors determined that the failures in 2012 and 2014 provided a reasonable opportunity to identify the cause of the failure and correct the condition adverse to quality.

In each of the events where the 1V222A load sequenced on early, the ‘A’ EDG was able to meet its TS requirements to achieve steady state voltage and frequency in the required time. Susquehanna performed an evaluation under AR-2016-12383 concluding there was no impact of the ‘A’ EDG to perform its safety functions with the described adverse condition.

The repeat failure was entered into the CAP as CR-2016-08643. Susquehanna identified the direct cause of the failure as misadjustment of the MOC linkage and attributed it to the same mechanism that occurred in 2012 and 2014. Additionally, Susquehanna identified the apparent cause of the failure was inadequate post maintenance testing (PMT) following a breaker replacement on July 19, 2010. Susquehanna determined the condition was reportable as a condition prohibited by plant TSs in accordance with 10 CFR 50.73(a)(2)(i)(B), because the ‘A’ EDG was inoperable for longer than the TS allowed outage time specified in TS 3.8.1, and 10 CFR 50.73(a)(2)(v)(D), as a condition that could have prevented fulfilment of a safety function.

Analysis: Inspectors determined that not identifying and correcting a condition adverse to quality in a timely manner associated with a misaligned MOC switch on the ‘A’ EDG output breaker to the 1A 4kV bus was a performance deficiency within Susquehanna’s ability to foresee and correct and should have been prevented. The finding was determined to be more than minor because it was associated with the Equipment

Performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the failure to correct the degraded condition rendered the ‘A’ EDG inoperable for longer than the TS allowed outage time. In accordance with IMC 0609.04, “Initial Characterization of Findings,” dated June 19, 2012, and Exhibit 2 of IMC 0609, Appendix A, “The SDP for Findings At-Power,” dated June 19, 2012, the inspectors determined that this finding required a detailed risk assessment because the finding represents an actual loss of function of a single train for greater than the TS allowed outage time. Specifically, the ‘A’ EDG was inoperable from July 19, 2010 until April 2, 2016.

In coordination with a Region 1 Senior Risk Analyst, the issue was qualitatively screened as Green (very low safety significance) based on the low initiating event frequency associated with a LOCA co-incident with a LOOP event. This would result in a delta core damage frequency substantially less than E-6. Additionally, it was reasonable to conclude that the ‘A’ EDG remained available to perform its function given the minimal increased load on the machine as evidenced during the performance of the LOOP-LOCA surveillance testing in 2012, 2014, and 2016.

This finding had a cross-cutting aspect in the area of Problem Identification and Resolution, Evaluation, because Susquehanna did not thoroughly evaluate the issue to ensure that the resolution addressed the cause and extent of conditions commensurate with their safety significance. Specifically, Susquehanna corrected a suspected condition without appropriate troubleshooting until the third identical failure of the 1V222A load timer. [P.2]

<u>Enforcement:</u> 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” requires that measures be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly corrected. Contrary to the above, despite failing TS surveillances in May 2012 and April 2014, which was ultimately determined to be caused by a misalignment of the MOC linkage switch (S1) in the ‘A’ EDG output breaker to the 1A 4kV bus which provides the electrical logic to the 1V222A load timer, implementation of the CAP did not assure that this condition adverse to quality was promptly corrected following each failure.

Susquehanna Unit 1 and Unit 2 TS LCO 3.8.1, “AC Sources - Operating,” requires, in part, four EDGs be operable in modes 1, 2, and 3 and requires that a EDG be restored within 72 hours of being declared inoperable. Contrary to the above, from July 19, 2010 through April 2, 2016, the ‘A’ EDG was inoperable.

Because this violation was of very low safety significance (Green), and Susquehanna has entered this performance deficiency into the CAP as CR-2016-08643, the NRC is treating this as a NCV in accordance with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000387; 388/2016002-02; Failure to Promptly Correct a Condition Adverse to Quality with ‘A’ EDG MOC Switch)**

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 – 4 samples)

a. Inspection Scope

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

- Unit 1, reactor vessel leak check on April 14, 2016
- Common, elevated risk during ESW system outage window on May 26, 2016
- Unit 1, emergent repair of pressure boundary leakage from 'B' RRP lower seal vent line and local power range monitor (LPRM) 24-09 instrument housing on June 9, 2016
- Unit 2, yellow risk during automatic depressurization system drywell pressure bypass timer calibrations on June 28, 2016

b. Findings

No findings were identified.

1R15 Operability Determinations and Functionality Assessments (71111.15 – 6 samples)

a. Inspection Scope

The inspectors reviewed operability determinations for the following degraded or non-conforming conditions based on the risk significance of the associated components and systems:

- Unit 1, failure of 1V222A to sequence on to 'A' EDG during LOCA/LOOP testing on March 23, 2016
- Unit 1, 'B' RHR heat exchanger vent motor operated valve torque switch incorrectly on April 11, 2016
- Common, breaker failure for 'A' SGTS heater on April 25, 2016
- Unit 1, elevated vibration on HPCI auxiliary oil pump identified on April 26, 2016
- Unit 1, 'D' MSIV stroke time outside acceptance limits on April 27, 2016
- Unit 2, HV255F006 stem leakage on May 17, 2016

The inspectors evaluated the technical adequacy of the operability determinations to assess whether TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TS and UFSAR to Susquehanna's evaluations to determine whether the components or systems were operable. The inspectors confirmed, where appropriate, compliance with bounding limitations associated with the evaluations. Where compensatory measures were required to maintain operability, such as in the case of operator workarounds (OWAs), the inspectors determined whether the measures in place would function as intended and were properly controlled by Susquehanna.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18 – 3 samples)

.1 Permanent Modifications

a. Inspection Scope

'B' RRP Cooler Assembly Replacement

The inspectors evaluated a modification to the 'B' RRP by engineering change package 1999985, "Replace RxR Pump Cooler Assembly." The inspectors verified that the design bases, licensing bases, and performance capability of the affected systems were not degraded by the modification. In addition, the inspectors reviewed modification documents associated with the upgrade and design change, including modification of associated piping to make it more resistant to vibration induced fatigue failure.

American Society of Mechanical Engineers Code Repair of LPRM 24-09 Instrument Housing

The inspectors evaluated an American Society of Mechanical Engineers code repair to the LPRM 24-09 instrument housing necessitated by a through-wall leak and implemented by engineering change package 2002083, "Repair Through Wall Leak on Unit 1 LPRM 24-09 Housing." The inspectors verified that the design bases, licensing bases, and performance capability of the affected systems were not degraded by the modification. In addition, the inspectors reviewed modification documents associated with the upgrade and design change, code repair forms, vendor quality control documents, as well as the fracture mechanics evaluations used to determine weld overlay size.

Unit 1, RHR low-pressure coolant injection (LPCI) injection cross-tie Modification

The inspectors evaluated a modification to the piping arrangement to cross-tie the RHR injection check valves implemented by engineering change package 1846732, "Ensure Positive Seating of RHR Injection Check Valves HV151F050 A and B." The inspectors verified that the design bases, licensing bases, and performance capability of the affected systems were not degraded by the modification. In addition, the inspectors reviewed modification documents associated with the design change, including containment isolation and reactor coolant pressure boundaries, pipe break considerations, operational margin for LPCI bypass flow, and welding specifications. The inspectors performed field walkdowns of the associated piping additions, hanger supports, and valve locations within primary containment. The inspectors also reviewed revisions to design basis calculations generated as a result of this modification.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19 – 6 samples)

a. Inspection Scope

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

- Unit 1, 'B' MSIV seat repair on April 1, 2016
- Unit 1, 'C' RHR pump motor replacement on April 6, 2016 and April 7, 2016
- Unit 1, RHR cross-connect modification on April 20, 2016
- Unit 1, RCIC turbine inspection on April 22, 2016
- Unit 1, repair of reactor recirculation sample inboard PCIV on June 8, 2016
- Unit 1, pressure seal leak on the hinge pin cover for 'B' feedwater line check valve, 141F010B, on June 13, 2016

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors reviewed the station's work schedule and outage risk plan for the Unit 1 maintenance and refueling outage 1R19, which was conducted March 11 through May 3, 2016, and the Unit 1 forced maintenance outage, which was conducted from June 5 through June 27, 2016. The inspectors reviewed Susquehanna's development and implementation of outage plans and schedules to verify that risk, industry experience, previous site-specific problems, and defense-in-depth were considered.

During the outages, the inspectors observed portions of the shutdown and cooldown processes and monitored controls associated with the following outage activities:

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

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22 – 5 samples)

a. Inspection Scope

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

The inspectors reviewed the following surveillance tests:

- Unit 1, standby liquid control vessel injection on April 7, 2016
- Unit 1, RHR testing from the remote shutdown panel on April 9, 2016
- Unit 1, integrated leak rate testing on April 12, 2016
- Unit 1, division II RHR logic system functional testing on April 16, 2016
- Common, 10-year simultaneous start of all EDGs on May 20, 2016

b. Findings

Introduction: A self-revealing finding of very low safety significance (Green) and associated NCV of TS 5.4.1.a, "Procedures," was identified for Susquehanna's failure to implement procedures for loading EDGs promptly following extended unloaded operation. Specifically, Susquehanna did not load the 'B' EDG promptly following over 6 hours of unloaded operation which resulted in the slow starting time during the subsequent surveillance test due to insufficient fuel delivery caused by clogged fuel injectors.

Description: The EDGs automatically start in emergency mode on a loss of offsite power or upon receiving a signal associated with a LOCA. Upon receiving a start signal, the EDGs are required to reach rated frequency and voltage within 10 seconds to ensure that assumptions in the accident analyses are met.

On May 20, 2016, the 'B' EDG did not reach rated frequency within the 10 second requirement of TSs while being tested in accordance with a surveillance requirement; consequently, the 'B' EDG was declared inoperable. In review of the event, Susquehanna determined that the most likely cause of the slow starting time was insufficient fuel delivery caused by clogged fuel injectors due to extended idle operation. Specifically, on May 13, 2016, all four operated in emergency mode when a high drywell pressure signal was received associated with a loss of drywell cooling. Since offsite power remained available during the event, the EDGs operated unloaded for an extended period (over 6 hours each).

Step 2.2.2 of OP-024-001, "Diesel Generators," requires that, "for every 6 hours at < 50 percent load, run diesel at > 75 percent load for 30 minutes immediately prior to shutdown, per Final Safety Analysis Report (FSAR) 8.3.14." FSAR section 8.3.14 states that "any diesel generator continuously operated at loads of less than 50 percent will be loaded in accordance with the manufacturer's recommendations to remove any built up combustion products." This is further echoed in FSAR question 040-82 which states, in part, that "the consequences of no load or light load operation are potential equipment failure due to the gum and varnish deposits."

Susquehanna reviewed the performance history of the 'B' EDG and noted that it was not run promptly following extended operation in an unloaded condition. The other three EDGs had been started and loaded to full load for 90 minutes prior to performance of the surveillance on May 20th and all reached rated frequency and voltage within the required 10 seconds during surveillance testing. In the case of the 'B' EDG, Susquehanna deferred running the EDG loaded due to an unexpected failure on the 'C' EDG, which rendered it inoperable and required substituting the spare 'E' EDG in for 'C'.

Susquehanna preferred to wait until the ‘C’ EDG was restored prior to running the ‘B’ EDG because a spare was unavailable if the ‘B’ were to fail. Inspectors determined that this decision was not prudent since EDGs remain operable while being run in the test mode of operation. Instead, the decision to delay a loaded run until the next monthly surveillance likely resulted in the inoperability of the ‘B’ EDG.

Analysis: Failure to implement requirements to load the EDGs promptly following extended unloaded operations was a performance deficiency that was within Susquehanna’s ability to foresee and correct and should have been prevented. The finding was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and affected the objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, the failure to load the ‘B’ EDG following extended operation unloaded likely resulted in the slow starting time of the EDG during subsequent surveillance testing due to clogged fuel injectors. The inspectors evaluated the finding in accordance with Exhibit 2 of IMC 0609, Appendix A, “The SDP for Findings At-Power,” dated June 19, 2012 and determined that it was of very low safety significance (Green) because it did not affect the design or qualification of the EDG, did not represent a loss of system function, and did not represent a loss of a single train for greater than its TS allowed outage time.

The finding is related to the cross-cutting area of Human Performance, Consistent Process, because Susquehanna did not use a consistent, systematic approach to make decisions which incorporated risk insights. Specifically, Susquehanna did not appropriately coordinate the loaded run of the ‘B’ EDG with maintenance on the ‘C’ EDG to ensure ‘B’ EDG availability was not unnecessarily challenged. [H.13]

Enforcement: TS 5.4.1.a, “Procedures,” requires in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in RG 1.33. RG 1.33, Appendix A requires procedure for operating the EDGs. OP-024-001, “Diesel Generators,” states to run an EDG at greater than 75 percent load for 30 minutes immediately prior to shut down for every 6 hours it is run at less than 50 percent load. Contrary to the above, following 6 hours of unloaded operation on May 13, 2016, Susquehanna did not run the ‘B’ EDG loaded which resulted in the ‘B’ EDG failing to reach rated frequency within the required 10 seconds when it was tested on May 20, 2016. Susquehanna’s immediate corrective actions included running the EDG loaded for an extended period to ensure any unburned fuel had been removed from the machine. Because the violation is of very low safety significance and has been entered into Susquehanna’s CAP as CR-2016-13220, this violation is being treated as an NCV, consistent with section 2.3.2.a of the Enforcement Policy. **(NCV05000387; 388/2016002-03, Failure of B EDG to Reach Rated Frequency within 10 Seconds)**

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06 – 1 sample)

.1 Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of a routine Susquehanna emergency drill on May 24, 2016 to identify any weaknesses and deficiencies in the classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the simulator and emergency operations facility to determine whether the event classification, notifications, and protective action recommendations were performed in accordance with procedures. The inspectors also attended the station drill critique to compare inspector observations with those identified by Susquehanna staff in order to evaluate Susquehanna's critique and to verify whether the Susquehanna staff was properly identifying weaknesses and entering them into the CAP.

b. Findings

Introduction: An NRC-identified finding of very low safety significance (Green) and associated NCV of 10 CFR 50.54(q)(2),"Emergency Plans" was identified for Susquehanna's failure to identify that an incorrect notification of wind direction was made to the SSO during a full-scale drill.

Description: On May 24, 2016, inspectors observed a full-scale emergency plan drill. As part of the drill scenario, there were five notifications to offsite agencies that were counted and assessed under the drill and exercise performance (DEP) performance indicator (PI). In particular, the first four notifications were for declared emergencies (UE, Alert, SAE and GE) and the fifth was notification of the site's PAR. While observing the fifth notification for the PAR, inspectors noted that the wind direction was not appropriately communicated as coming "from 229 degrees." Specifically, the individual that was acting as the SSO recorded and repeated back a wind direction of "from 29 degrees." This was not corrected by the recovery manager who made the communication. Inspectors observed the critique process and noted that this issue was not discussed as a weakness or area for follow-up. Additionally, inspectors noted that Susquehanna assessed the DEP PI opportunity as a success.

Inspectors questioned Susquehanna about whether the discrepancy was noted during the assessment of the drill or whether the two forms (the transmitted form and the one in which the SSO had written down the transmission) were compared following the drill. In response to inspectors' questions, Susquehanna determined that 1) the issue was not appropriately critiqued and 2) that it resulted in a DEP PI failure for the opportunity. Susquehanna entered the failure to critique into the CAP as CRs 2016-14303 and 2016-14128.

To assess the potential significance of the issue, the inspectors reviewed the other notifications for the drill. In particular, the inspectors assessed that the wind direction was appropriately communicated as “from 229 degrees” in each of the previous four notifications by comparing the transmitted form from the emergency response facility to the received communication as documented by the OROs.

Additionally, inspectors noted that the wind direction of 29 degrees was inconsistent with the evacuation sectors of the PAR and therefore the error would have been identifiable by the OROs.

Nuclear Energy Institute (NEI) 99-02, “Regulatory Assessment PI Guideline,” Revision 7, states under the DEP PI that a notification is considered accurate if the notification form is appropriate to the event to include wind direction and speed. A clarifying note in the guidance states that “Minor discrepancies in the wind speed and direction provided on the emergency notification form need not count as a missed notification opportunity provided the discrepancy would not result in an incorrect PAR being provided.”

Though Susquehanna determined that the discrepancy in wind direction, 29 versus 229 degrees, would not be considered minor, inspectors determined that this note was important in assessing the risk significance of the finding when considering that it was being transmitted in a correct PAR and that the wind direction had been accurately transmitted in the notification of the General Emergency minutes prior the PAR notification.

Analysis: Failure to identify performance weaknesses by emergency responders during a full-scale drill was a performance deficiency that was within Susquehanna’s ability to foresee and correct and should have been prevented. The finding was more than minor because it is associated with the ERO Performance attribute of the Emergency Preparedness Cornerstone and affected the cornerstone objective to ensure that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. Specifically, the failure of Susquehanna personnel to effectively identify an exercise weakness associated with a RSPS caused a missed opportunity to identify and correct a drill-related performance deficiency. The inspectors assessed the issue using the Emergency Preparedness SDP, Appendix B to IMC 0609, dated September 23, 2014. Inspectors noted two examples provided in table 5.14-1 that were similar to the performance deficiency:

- White Finding: An example of a loss of PS function occurs when the critique process fails to identify a weakness associated with a RSPS that is determined by the NRC to be a DEP PI opportunity failure during a full-scale drill.

- Green Finding: An example of a degradation of the PS function occurs when the critique process fails to identify a weakness associated with a RSPS that is determined by the NRC to be a DEP PI opportunity success during a full-scale drill.

In this case, Susquehanna's failure to critique the inaccurate notification met the NRC's definition of a weakness in a full-scale drill. However, because four previous notifications had accurately reported the wind direction and the miscommunication was inconsistent with the correct PAR that was communicated simultaneously, in consultation with a senior emergency preparedness inspector, inspectors determined the communication would likely have been corrected prior to the OROs acting on the incorrect information, did not result in an incorrect PAR, and therefore determined that that the failure to critique the drill weakness only constituted a degradation of the PS function. Therefore the finding is characterized as having very low safety significance (Green).

The finding is related to the cross-cutting area of Problem Identification and Resolution, Identification, in that Susquehanna did not identify a RSPS issue completely, accurately, and in a timely manner commensurate with the safety significance. Specifically, during the full-scale drill, Susquehanna failed to recognize and critique that a RSPS was not met and did not place this issue into the CAP until prompted by inspectors. [P.1]

Enforcement: 10 CFR 50.54(q)(2) requires, in part, that a licensee shall follow and maintain the effectiveness of an emergency plan that meets the requirements in appendix E to 10 CFR 50 and, for nuclear power reactor licensees, the PSs of 10 CFR 50.47(b). 10 CFR 50.47(b)(14) requires, in part, periodic drills be conducted to develop and maintain key skills, and deficiencies identified as a result of drills be corrected.

Section IV.F.2.g of Appendix E to 10 CFR Part 50 requires that all training, including drills, shall provide for formal critiques in order to identify weak or deficient areas that need correction. Additionally, it requires that any identified weaknesses or deficiencies be corrected.

Contrary to the above, during the May 27, 2016, critique of the May 26, 2016, full-scale emergency planning drill, Susquehanna did not identify an ERO performance weakness. Specifically, Susquehanna did not identify that an inaccurate notification was made when a wrong wind direction was communicated with the PAR. Susquehanna's immediate corrective actions included remediating the ERO personnel involved in the incorrect communication and the drill controllers that failed to identify the deficiency, as well as communicating lessons learned to other emergency response organization personnel. Because the violation is of very low safety significance and has been entered into Susquehanna's CAP as CRs 2016-14303 and 2016-14128, this violation is being treated as an NCV, consistent with section 2.3.2.a of the Enforcement Policy. **(NCV05000387; 388/2016002-04, Failure to Critique an Incorrect PAR Notification)**

2. RADIATION SAFETY

Cornerstone: Occupational and Public Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01 – 4 samples)

a. Inspection Scope

The inspectors reviewed Susquehanna's performance in assessing and controlling radiological hazards in the workplace. The inspectors used the requirements contained in 10 CFR 20, TSs, applicable regulatory guides, and the procedures required by TSs as criteria for determining compliance.

Inspection Planning

The inspectors reviewed the PIs for the occupational radiation safety cornerstone, radiation protection (RP) program audits, and reports of operational occurrences in occupational radiation safety since the last inspection.

Radiological Hazard Assessment (1 sample)

The inspectors conducted independent radiation measurements during walk-downs of the facility and reviewed the radiological survey program; air sampling and analysis; continuous air monitor use; recent plant radiation surveys for radiological work activities; and any changes to plant operations since the last inspection to verify survey adequacy of any new radiological hazards for onsite workers or members of the public.

Instructions to Workers (1 sample)

The inspectors reviewed HRA work permit controls and use; observed containers of radioactive materials and assessed whether the containers were labeled and controlled in accordance with requirements.

The inspectors reviewed several occurrences where a worker's electronic personal dosimeter alarmed. The inspectors reviewed Susquehanna's evaluation of the incidents, documentation in the CAP, and whether compensatory dose evaluations were conducted when appropriate. The inspectors verified follow-up investigations of actual radiological conditions for unexpected radiological hazards were performed.

Contamination and Radioactive Material Control (1 sample)

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

Problem Identification and Resolution (1 sample)

The inspectors evaluated whether problems associated with radiation monitoring and exposure control (including operating experience) were identified at an appropriate threshold and properly addressed in the CAP.

b. Findings

.1 Introduction: A Green self-revealing NCV of TS 5.7.1, High Radiation Area Controls, was identified when a worker did not comply with a radiological posting barrier and other access control requirements for HRA entry. Specifically, on December 26, 2015, a security officer conducting a tour of Unit 1, entered a posted HRA without proper authorization.

Description: On December 26, 2015, a security officer was conducting a familiarization tour of the Unit 1 Turbine Building Elev. 699 and approached the alcove area outside a HRA barrier in front of a condenser bay door. The security officer leaned into the posted HRA barrier to ascertain if the door was a security door when the electronic dosimeter alarmed. The security officer promptly left the area and informed RP about the dosimeter alarm. The peak dose rate on the electronic dosimeter was found to be 198 mrem/hr, which confirmed that the security officer had entered a HRA. The security officer was using RWP 2015-0022, Activity 1, with electronic dosimeter settings of 10 mrem for accumulated dose and 80 mrem/hr for dose rate, and this RWP did not allow access to HRAs nor had the security officer been briefed on radiological conditions in the area, which is required for the HRA entry. Immediate corrective actions taken included moving the HRA barrier further out and placing a shield rack in front of the condenser bay door to reduce radiation dose rates. The issue was entered into Susquehanna's CAP as CR-2015-33947. Susquehanna staff indicated that there has been a history of personnel receiving electronic dosimeter alarms at this location in the plant.

TS 5.7.1.b requires that activities in a HRA be controlled by means of a RWP that authorizes entry. TS 5.7.1.e requires that for individuals not qualified in radiation procedures or escorted by such a person, entry into a HRA be authorized only after radiological conditions in the work area have been determined and personnel have been briefed on those conditions. NDAP-QA-0626, "Radiologically Controlled Area Access and RWP System," Revision 42, implements these same requirements and requires adherence to HRA posting requirements prior to entry. The inspectors determined that these TS and procedural requirements were not met.

Analysis: The failure to adhere to a HRA posting requiring a HRA RWP and a radiological briefing prior to entry is a performance deficiency that was reasonably within Susquehanna's ability to foresee and correct and should have been prevented. The finding was self-revealing because Susquehanna was made aware of the situation after an electronic dose rate alarm. The finding was determined to be more than minor based on similarity to example 6.h in IMC 0612, Appendix E, and it is associated with Human Performance attribute of the Occupational Radiation Safety Cornerstone and affected the cornerstone objective to ensure adequate protection of the worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. Specifically, the individual violated the HRA posting, and missed reviewing the RWP and briefing requirements designed to protect the worker from unnecessary radiation exposure. Using IMC 0609, Appendix C, "Occupational Radiation Safety SDP," dated August 19, 2008, the finding was determined to be of very low safety significance (Green) because it did not involve: (1) ALARA occupational collective exposure planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose.

The cause of the finding is related to the cross-cutting aspect of Problem Identification and Resolution, Resolution, in that the organization did not ensure that corrective actions to address the cause of repetitive electronic dosimeter alarms in this area of the plant and had not been sufficiently evaluated and had not enhanced radiological controls to prevent this issue from recurring. [P.3]

Enforcement: TS 5.7.1.b requires that activities in a HRA be controlled by means of a RWP that authorizes entry. TS 5.7.1.e requires that, for individuals not qualified in radiation procedures or escorted by such a person, entry into a HRA be authorized only after radiological conditions in the work area have been determined and personnel are briefed on these conditions.

Contrary to this, on December 26, 2015, a security officer conducting a familiarization tour of Unit 1 entered into a HRA while signed in on RWP 2015-0022, Activity 1 that did not authorize entry to a HRA and had not been briefed on the radiological conditions in the HRA area. Immediate corrective actions taken included moving the HRA barrier further out and placing a shield rack in front of the condenser bay door to reduce radiation dose rates. Because this finding was determined to be of low safety significance (Green) and was entered into Susquehanna's CAP (CR-2015-33947), this violation is being treated as an NCV consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV05000387; 388/2016002-05, Entry into a High Radiation Area without Radiological Briefing)**

.2 Introduction: A Green self-revealing NCV of TS 5.7.2, High Radiation Area Controls, was identified when workers erroneously entered the wrong reactor unit condenser bay (Unit 2) that was posted and controlled as a LHRA. Specifically, on May 3, 2016, four Susquehanna staff were briefed to enter the Unit 1 condenser bay to check for steam leaks during start up, however the staff entered the Unit 2 condenser bay during full power operations and received electronic dosimeter alarms.

Description: On May 3, 2016, four Susquehanna staff were briefed to enter the Unit 1 condenser bay to check for steam leaks during start up; however, the staff erroneously entered the Unit 2 condenser bay and received electronic dosimeter dose rate alarms of approximately 1050 mrem/hr. The group exited after receiving the alarms and reported to radiation protection. Event followup determined that the group entered the wrong reactor unit condenser bay and that actual LHRA conditions existed (> 1 Rem/hr dose rates) during the personnel entry. Susquehanna procedure allowed for the use of a master key to allow routine entry into a LHRA and therefore the workers were able to enter the Unit 2 LHRA by mistake. Further investigation found that the radiation measuring instrument being used by the RP technician during the entry, was under responding to the radiation dose and did not provide an early indication of higher than expected dose rates and did not prevent the personnel entry. Prior to entry, the work group did not confirm that they were in the appropriate reactor unit for the work they were about to conduct.

TS 5.7.2.b requires that activities in a HRA with dose rates greater than or equal to 1.0 rem/hr at 30 centimeters from the source be controlled by means of an RWP that includes specification of radiation dose rates in the immediate work area and other appropriate RP equipment and measures. NDAP-QA-0626, "Radiologically Controlled Area Access and RWP System," Revision 44, implements these requirements and step 5.6.5 and Attachment P, "High Radiation/Locked High Radiation/Very High Radiation Area Briefing Checklist," requires, in part, a radiological briefing from RP prior to entering the LHRA that includes a discussion of the required RWP, work area radiation levels, and electronic dosimeter dose alarm and dose rate alarm settings. The inspectors determined that these requirements were not met for entry in to the operating Unit 2 condenser bay.

Analysis: The failure to adhere to a LHRA posting requiring a LHRA RWP and a radiological briefing prior to entry was a performance deficiency that was reasonably within Susquehanna's ability to foresee and correct and should have been prevented. The finding was determined to be more than minor based on a similar example 6.h in IMC 0612, Appendix E, and it is associated with Human Performance attribute of the Occupational Radiation Safety Cornerstone and affected the cornerstone objective to ensure adequate protection of the worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. Specifically, Susquehanna staff violated the RWP and briefing requirements designed to protect workers from unnecessary radiation exposure. Using IMC 0609, Appendix C, "Occupational Radiation Safety SDP," dated, August 19, 2008, the finding was determined to be of very low safety significance (Green) because it did not involve: (1) ALARA occupational collective exposure planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. The finding was self-revealing because Susquehanna was made aware of the situation as a result of an electronic dose rate alarm.

The cause of the finding is related to the cross-cutting aspect of Human Performance, Teamwork because the workers did not conduct peer checking and recognize and communicate that they were in the wrong reactor unit for the work they were conducting. Specifically, four Susquehanna staff were briefed to enter the Unit 1 condenser bay to check for steam leaks during start up; however, the staff entered the Unit 2 condenser bay. [H.4]

Enforcement: TS 5.7.2.b requires that activities in a HRA with dose rates greater than or equal to 1.0 rem/hr at 30 centimeters from the source be controlled by means of a RWP that includes specification of radiation dose rates in the immediate work area and other appropriate RP equipment and measures.

Contrary to this requirement, on May 3, 2016, four Susquehanna staff were briefed to enter the Unit 1 condenser bay to check for steam leaks during start up, however the staff entered the Unit 2 condenser bay which was operating at 100 percent power and was a HRA with dose rates greater than 1.0 rem/hr at 30 centimeters without proper RWP authorization and without specification of radiation dose rates in the work area. Immediate corrective actions were to suspend the use of master keys for routine entry into LHRA and conduct a radiation safety stand down. Corrective actions included revising RP-121-1006, Locking and Key Control, to discontinue issuance of master keys for routine LHRA entries. Because this finding was determined to be of low safety significance (Green) and was entered into Susquehanna's CAP (CR-2016-11944), this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy. **(NCV05000387; 388/2016002-06, Entry into a Locked High Radiation Area without an Appropriate RWP/Radiological Briefing)**

2RS2 Occupational ALARA Planning and Controls (71124.02 – 2 samples)

a. Inspection Scope

From March 28 through April 1, 2016, inspectors assessed Susquehanna's performance with respect to maintaining occupational individual and collective radiation exposures ALARA. The inspectors used the requirements contained in 10 CFR 20, applicable regulatory guides, TSs, and procedures required by TSs as criteria for determining compliance.

Radiological Work Planning

The inspectors selected the following radiological work activities based on exposure significance for review:

- RWP 20161120, RWCU complex and back wash receiving tank room: general work, inspections, and non-RWCU valve breach
- RWP 20161162, fuel pool heat exchanger room general work
- RWP 20161222, replace feedwater heaters and associated activities
- RWP 20161370, nozzle and vessel in-service inspection and associated support

For each of these activities, the inspectors reviewed: ALARA work activity evaluations; exposure estimates; exposure reduction requirements.

Verification of Dose Estimates and Exposure Tracking Systems

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

Radiation Worker Performance (1 Sample)

The inspectors observed radiation worker and RP technician performance during radiological work to evaluate worker ALARA performance according to specified work controls and procedures. Workers were interviewed to assess their knowledge and awareness of planned and/or implemented radiological and ALARA work controls.

Problem Identification and Resolution (1 sample)

The inspectors evaluated whether problems associated with ALARA planning and controls were identified at an appropriate threshold and properly addressed in the CAP.

b. Findings

No findings were identified.

2RS3 <u>In-Plant Airborne Radioactivity Control and Mitigation</u> (71124.03 – 1 sample)

a. Inspection Scope

From March 28 through April 1, 2016, the inspectors reviewed the control of in-plant airborne radioactivity and the use of respiratory protection devices in these areas. The inspectors used the requirements in 10 CFR 20, RG 8.15, RG 8.25, NUREG/CR-0041, TS, and procedures required by TS as criteria for determining compliance.

Inspection Planning

The inspectors reviewed the UFSAR to identify ventilation and radiation monitoring systems associated with airborne radioactivity controls and respiratory protection equipment staged for emergency use. The inspectors also reviewed respiratory protection program procedures and current PIs for unintended internal exposure incidents.

Engineering Controls (1 sample)

The inspectors reviewed operability and use of both permanent and temporary ventilation systems, and the adequacy of airborne radioactivity radiation monitoring in the plant based on location, sensitivity, and alarm set-points.

b. Findings

No findings were identified.

2RS5 <u>Radiation Monitoring Instrumentation</u> (71124.05 – 3 samples)

a. Inspection Scope

The inspectors reviewed performance in assuring the accuracy and operability of radiation monitoring instruments used to protect occupational workers during plant operations and from postulated accidents. The inspectors used the requirements in 10 CFR 20; RGs; applicable industry standards; and procedures required by TSs as criteria for determining compliance.

Inspection Planning

The inspectors reviewed: UFSAR; Radiation RP audits; records of in-service survey instrumentation; and procedures for instrument source checks and calibrations.

Walkdowns and Observations (1 sample)

The inspectors conducted walk-downs of plant area radiation monitors and continuous air monitors. The inspectors assessed material condition of these instruments and that the monitor configurations aligned with the UFSAR. The inspectors checked the calibration and source check status of various portable radiation survey instruments and contamination detection monitors for personnel and equipment.

Calibration and Testing Program (1 sample)

For the following radiation detection instrumentation, the inspectors reviewed the current detector and electronic channel calibration, functional testing results alarm set-points and the use of scaling factors: laboratory analytical instruments, whole body counter, containment high-range monitors, portal monitors; personnel contamination monitors; small article monitors; portable survey instruments; area radiation monitors; electronic dosimetry; air samplers; and continuous air monitors. The inspectors reviewed the calibration standards used for portable instrument calibrations and response checks to verify that instruments were calibrated by a facility that used National Institute of Science and Technology traceable sources.

Problem Identification and Resolution (1 sample)

The inspectors verified that problems associated with radiation monitoring instrumentation (including failed calibrations) were identified at an appropriate threshold and properly addressed in the CAP.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Reactor Coolant System Specific Activity and Reactor Coolant System Leak Rate (4 samples)

a. Inspection Scope

The inspectors reviewed Susquehanna’s submittal for the reactor coolant system specific activity and reactor coolant system leak rate performance indicators for both Unit 1 and Unit 2 for the period of April 1, 2015, through March 31, 2016. To determine the accuracy of the PI data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, “Regulatory Assessment PI Guideline,” Revision 7. The inspectors also reviewed reactor coolant system sample analysis and control room logs of daily measurements of reactor coolant system leakage, and compared that information to the data reported by the performance indicator. Additionally, the inspectors observed surveillance activities that determined the reactor coolant system identified leakage rate, and chemistry personnel taking and analyzing a reactor coolant system sample.

b. Findings

No findings were identified.

.2 Occupational Exposure Control Effectiveness (1 sample)

a. Inspection Scope

The inspectors reviewed Susquehanna submittals for the occupational radiological occurrences PI for the first quarter 2015 through the fourth quarter 2015. The inspectors used PI definitions and guidance contained in the NEI Document 99-02, Revision 7, to determine the accuracy of the PI data reported. The inspectors reviewed electronic personal dosimetry accumulated dose alarms, dose reports, and dose assignments for any intakes that occurred during the time period reviewed to determine if there were potentially unrecognized PI occurrences. The inspectors conducted walk-downs of various Locked High and Very High Radiation Area entrances to determine the adequacy of the controls in place for these areas.

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152 – 2 samples)

.1 Routine Review of Problem Identification and Resolution Activities

a. Inspection Scope

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

b. Findings

No findings were identified.

.2 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a semi-annual review of site issues to identify trends that might indicate the existence of more significant safety concerns. As part of this review, the inspectors included repetitive or closely-related issues documented by Susquehanna in trend reports, site performance indicators, major equipment problem lists, system health reports, maintenance rule assessments, and maintenance or CAP backlogs. The inspectors also reviewed Susquehanna's CAP database for the first and second quarters of 2016 to assess CRs written in various subject areas (equipment problems, human performance issues, etc.), as well as individual issues identified during the NRCs daily CR review (Section 4OA2.1). The inspectors reviewed the Susquehanna quarterly trend report for the first quarter of 2016, conducted under LS-125-1009, "Station Trending Manual," Revision 2, to verify that Susquehanna personnel were appropriately evaluating and trending adverse conditions in accordance with applicable procedures.

b. Findings and Observations

No findings were identified.

Degraded Fire Barriers. The inspectors identified a trend in adverse conditions which impact fire barriers (e.g. fire doors). Specifically, inspectors noted that since June 2014, the average number of CRs generated and assigned to system 12, Buildings and Facilities, rose from 13 to 22 per month. Inspectors reviewed the CRs and noted that the bulk of CRs generated and assigned to the system were related to fire door deficiencies. Inspectors noted that the station tracks fire impairments in the daily status package, ensuring timely response to each individual condition identified. However, the higher rate of issue identification may indicate that additional corrective actions are required to resolve the potential adverse trend.

Human Performance Events. Inspectors continued to note an adverse trend in the number of human performance events. This trend was noted previously in the semi-annual review of trends documented in IRs 2013-005 (ML14045A295) and 2014-003 (ML14225A018). In review of CAP data since this was last documented in a semi-annual trend, inspectors noted that the number of prompt human performance investigations has continued to rise from 9 to 12 per month, on average, representing a 33 percent increase. Additionally, inspectors noted that the significance of these events has risen as well, contributing to four of the findings and/or reportable events documented in this inspection report.

.3 Annual Sample: Trend in Fire Protection Program Related Events

a. Inspection Scope

The inspectors performed an in-depth review of Susquehanna's apparent cause evaluation (ACE) associated with CR 2015-04357. This ACE was generated as a result of Susquehanna's failure to control the storage of transient combustibles in accordance with the fire protection program requirements (NCV 05000387; 388/2015001-02, Control of Transient Combustible Materials.) In addition, the inspectors reviewed Susquehanna's gap analysis evaluation related to CR-2014-3525. This CR was created because Susquehanna failed to implement their fire risk management and integrated risk management procedure (NCV 05000387; 388/2014005-01, Risk Management Actions Not Implemented). The objective of the Susquehanna's gap analysis was to ensure that all aspects of the NCV were appropriately and adequately addressed.

The inspectors assessed Susquehanna's problem identification threshold, compensatory actions, and the prioritization and timeliness of corrective actions to determine whether Susquehanna was appropriately identifying, characterizing, and correcting problems associated with these issues. The inspectors also assessed Susquehanna's corrective actions to prevent recurrence.

In addition, the inspectors toured both units, reviewed documentation associated with Susquehanna's fire protection program, and interviewed Susquehanna's fire personnel in order to assess the effectiveness of the recommended and implemented corrective actions.

b. Findings and Observations

No findings were identified.

The inspectors concluded that Susquehanna took the appropriate actions to identify and evaluate the causes of both NCVs. Susquehanna determined that one of the apparent causes of their failure to control the storage of transient combustibles was a lack of questioning attitude by the maintenance staff. They also concluded that the inadequate use of NDAP-QA-0503, Housekeeping Transient Material and Internal Cleanliness, by Susquehanna's personnel was an additional apparent cause. In addition, Susquehanna determined that lack of oversight at the job site was a contributing factor of the issue.

While reviewing Susquehanna's fire protection procedures, the inspectors identified that the fire marshal roles and responsibilities were not defined in the responsibilities section. Susquehanna acknowledged the observation and entered it into the CAP as CR-2016-04838. Susquehanna is in the process of adding these roles and responsibilities into their fire protection procedures. The inspectors determined that Susquehanna's overall response to these issues was timely and commensurate with its safety significance, and the actions taken and planned were reasonable to resolve the deficiencies identified in both NCVs.

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153 – 8 samples)

.1 Plant Events

a. Inspection Scope

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

- Unit 2, reactor scram due to loss of 480V motor control center and associated loss of drywell cooling on May 13, 2016

b. Findings

Introduction: An NRC-identified finding of very low safety significance (Green) and associated NCV of TS 5.4.1.a, “Procedures,” was identified for Susquehanna’s failure to implement procedures for controlling the HPCI system. Specifically, operators overrode automatic initiation of the system prior to inserting a manual scram, contrary to the requirements of OP-252-001, “HPCI System,” and OP-AD-300, “Administration of Operations.”

Description: The HPCI system is designed to automatically initiate on low reactor vessel water level. Though its primary purpose is to maintain reactor vessel inventory after small loss of coolant accidents, the HPCI system is capable of responding to restore reactor water level if the non-safety related feedwater system and RCIC systems were to fail during a plant transient. TS 3.5.1, “ECCS- Operating,” in part, requires HPCI to be operable in Mode 1.

On May 13, 2016, operators manually scrammed Unit 2 in response to rising drywell pressure and temperature due to an electrical transient causing a loss of reactor building chilled water and closed loop cooling water to the drywell. Prior to inserting the manual scram and initiating the plant transient, operators overrode automatic initiation of the HPCI system by performing section 2.16 of OP-252-001, “HPCI System,” which rendered the system inoperable. Though HPCI is not normally expected to initiate during a reactor scram if the non-safety related integrated control system (ICS) responds appropriately to control reactor water level, this action rendered the HPCI system incapable of automatic initiation in the event of a failure of the ICS.

Inspectors identified that this action was taken while reviewing operator logs following the scram. Inspectors reviewed OP-252-001 and questioned crew supervision whether the prerequisites for section 2.16, “Overriding HPCI Injection,” were met prior to performance. Specifically, step 2.16.1 states that HPCI can be throttled, stopped, prevented or inhibited under the cognizance of a Unit Supervisor or Shift Supervisor only if: 1) it is directed by an emergency operating procedure (EOP), 2) the system is not operating correctly as confirmed by at least two independent indications, or 3) adequate core cooling has been assured by at least two independent indications. This prerequisite is echoed in OP-AD-300, “Administration of Operations.” Additionally, OP-AD-300, Attachment M, “Transient Mitigation Strategies,” section G, “Scram Choreography,” states that once the mode switch is placed in shutdown (i.e. the scram has been initiated), the reactor operator can override HPCI if automatic initiation has occurred and adequate core cooling has been assured. Inspectors determined that operators had not implemented these two procedures correctly when HPCI was overridden prior to initiating the plant transient and verifying that adequate core cooling existed with the main feedwater system.

The inspectors noted that, in parallel with inspection activities, the operating crew identified that their actions associated with overriding HPCI were not prudent as part of the post-event critique and therefore, inspectors considered whether this performance deficiency should be characterized as licensee-identified. However, inspectors determined that Susquehanna failed to fully identify the facts surrounding the event or identify that the crew’s actions were contrary to written procedures. Inspectors noted that this deficiency was not documented in the emergency notification (EN51925) and was not adequately identified by Susquehanna during their post-event review or discussed in the pre-startup plant operational review committee meeting. Therefore, inspectors are considering this performance deficiency as NRC-identified because inspector intervention was required to ensure that adequate corrective actions were taken to address the operator performance issue.

Susquehanna entered the issue into the CAP as CRs 2016-12854 and 2016-13118 and 2016-13136 and performed a prompt human performance evaluation to determine: 1) why the procedure was not implemented correctly and 2) why the deficient operator performance was not identified during the post-event review. Immediate corrective actions included remediating the operators involved in the event and communicating lessons learned to other station personnel.

Analysis: Failure to implement procedures for operation of the HPCI system was a performance deficiency that was within Susquehanna's ability to foresee and correct and should have been prevented. The finding was more than minor because it was associated with the human performance attribute of the Mitigating Systems Cornerstone and affected the objective to ensure the availability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, overriding the HPCI system prior to initiating a plant scram rendered the system unavailable to respond to a level transient or failure of the non-safety related feedwater system. The inspectors evaluated the finding in accordance with Exhibit 2 of IMC 0609, Appendix A, "The SDP for Findings At-Power," dated June 19, 2012 and determined that it required a detailed risk assessment because it represented a loss of the single train system's function. The Regional Senior Reactor Analyst performed a detailed risk evaluation using the Susquehanna Unit 2 SPAR Model, version 8.23. The issue was conservatively modeled with a HPCI failure to start due to the system automatic start signal being overridden.

The change in core damage frequency per year was determined to be in the E-10 range due to the very short duration the system auto start feature was defeated. Therefore the issue was determined to be of very low safety significance (Green).

The finding is related to the cross-cutting area of Human Performance, Procedure Adherence because Susquehanna did not follow processes, procedures and work instructions. Specifically, operators did not ensure that their actions were appropriately authorized by procedures when taking action to override a key safety system prior to a plant transient. [H.8]

Enforcement: TS 5.4.1.a, "Procedures," requires, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in RG 1.33. RG 1.33, Appendix A requires procedure for operating HPCI as well as administration procedures to control bypass of safety functions. OP-252-001, "HPCI System," and OP-AD-300, "Administration of Operations," only allow overriding automatic HPCI system initiation under the cognizance of a Unit Supervisor or Shift Supervisor if 1) it is directed by an emergency operating procedure (EOP), 2) the system is not operating correctly as confirmed by at least two independent indications, or 3) adequate core cooling has been assured by at least two independent indications. Contrary to the above, on May 13, 2016, operators overrode automatic HPCI system initiation without ensuring that one of the three criteria specified in OP-252-001 and OP-AD-300 were met.

Susquehanna's immediate corrective actions included remediating the operators involved in the event and communicating lessons learned to other station personnel. Because the violation is of very low safety significance and has been entered into Susquehanna's CAP as CRs 2016-12854 and 2016-13118 and 2016-13136, this violation is being treated as an NCV, consistent with section 2.3.2.a of the Enforcement Policy. **(NCV05000388/2016002-07, HPCI Overridden Prior to Manual Reactor Scram)**

.2 (Closed) LER 05000387/2016-006-00: Implementation of Enforcement Guidance Memorandum (EGM) 11-003, Revision 3

From March 16 through April 11, 2016, Susquehanna performed OPDRVs without establishing secondary containment integrity. An OPDRV is an activity that could result in the draining or siphoning of the reactor pressure vessel (RPV) water level below the top of fuel, without crediting the use of mitigating measures to terminate the uncovering of fuel. TS 3.6.4.1, "Secondary Containment" requires that secondary containment be operable and is applicable during OPDRVs. The required action for this specification if secondary containment is inoperable in this condition of applicability is to initiate actions to suspend OPDRVs immediately. Therefore, failing to maintain secondary containment operability during OPDRVs without initiating actions to suspend the operation was considered a condition prohibited by TSs as defined by 10 CFR 50.73(a)(2)(i)(B).

As reported in LER 05000387/2016-006, Susquehanna conducted the following OPDRVs during the period of secondary containment inoperability:

- Recirculation system maintenance;
- RWCU system maintenance;
- RHR system local leak-rate test and maintenance;
- Hydraulic control unit and control rod drive system maintenance;
- LPRM replacement; and
- Control rod drive mechanism replacements

NRC EGM 11-03, "Enforcement Guidance Memorandum On Dispositioning BWR Licensee Noncompliance With TS Containment Requirements During Operations With A Potential For Draining The Reactor Vessel," Revision 3, provides, in part, for the exercise of enforcement discretion only if the licensee demonstrates that it has met four specific criteria during an OPDRV activity. The inspectors' assessments of Susquehanna's implementation of these four criteria during the LPRM replacement activity are described below:

1) The inspectors observed that, as required by the EGM, the OPDRV activities were logged in the control room narrative logs and that the log entries appropriately documented actions being taken to ensure water inventory was maintained and defense-in-depth criteria were in place.

2) The inspectors noted that the reactor vessel water level was maintained above the RHR high water level setpoint of 22 feet. The inspectors also noted that at least one safety-related pump was the standby source of makeup designated in the control room narrative logs for the evolutions. Susquehanna logged that the worst case estimated time to drain the reactor cavity to the RPV flange was greater than the EGM criteria of 24 hours.

3) The inspectors verified that the OPDRVs were not conducted in Mode 4 and that Susquehanna maintained secondary containment operability for the refueling floor while moving irradiated fuel during OPDRVs. The inspectors noted that Susquehanna had contingency plans in place for isolating the potential leakage paths, should difficulty arise during various maintenance activities. Additionally, the inspectors verified that two independent means of measuring RPV water level (one alarming) were available for identifying the onset of loss of inventory events.

4) Inspectors verified that all other TSs were met during OPDRVs with secondary containment inoperable.

TS 3.6.4.1 is applicable during OPDRVs and requires that secondary containment be operable. TS 3.6.4.1, action C.3, requires operators to initiate actions to suspend OPDRVs immediately upon discovery that secondary containment is inoperable. Contrary to the above, between March 16, 2016 through April 11, 2016, Susquehanna did not maintain secondary containment operable while performing OPDRVs. Because the violations were identified during the discretion period described in EGM 11-003, and the licensee has met the four criteria specified above, the NRC is exercising enforcement discretion in accordance with Section 3.5, "Violations Involving Special Circumstances," of the NRC Enforcement Policy and, therefore, will not issue enforcement action for this violation. In accordance with EGM 11-003, each licensee that receives discretion must submit a license amendment request within 12 months of the NRC staff's publication in the Federal Register of the notice of availability for a generic change to the Standard TSs to provide more clarity to the term OPDRV. The inspectors observed that Susquehanna is tracking the need to submit a license amendment request as AR-2015-01726. This LER is closed.

.3 (Closed) Licensee Event Report (LER) 05000388;387/2015-015-00: Loss of Safety Function due to Inoperability of Both Trains of the Control Room Emergency Outside Air Supply System

On October 3, 2014, both trains of Control Room Emergency Outside Air Supply System (CREOASS) were rendered inoperable when Susquehanna executed two separate surveillances concurrently. Susquehanna personnel did not recognize the October 3, 2014 event to be reportable as a loss of safety function at the time the event occurred in 2014. Instead, the NRC resident inspectors identified the event on November 24, 2015, while performing an extent of condition review of LER 387 (388)/2015-06-00 described above. After identification of the event, Susquehanna determined that the cause was the same as that identified in the previous LER and that no new additional corrective actions were necessary.

The enforcement aspects of this finding are discussed in section 4OA2 of IR 05000387; 388/2016-001 (ML16132A421) and section 1R15 of IR 05000387; 388/2015-004 (ML16040A197). The inspectors did not identify any new issues during the review of the LER. This LER is closed.

.4 <u>(Closed) Licensee Event Report (LER) 05000387/2016-009-00</u>: Valid Primary Containment Isolation Actuation during Local Leak Rate Testing due to Human Performance Error

On March 31, 2016, while performing lineups for local leak rate testing, Susquehanna received primary containment isolation system (PCIS) and secondary containment isolation system (SCIS) actuations, in addition to initiations of Standby Gas Treatment Systems (SGTS) and CREOASS. In review of the event, Susquehanna determined that separate tests that affected each division of PCIS and SCIS were being lined up concurrently. Additionally, Susquehanna determined that performance of these tests simultaneously had not been authorized and was the result of a human performance error. Susquehanna determined the actuations of SGTS and CREOASS were the result of both systems receiving valid signals from PCIS/SCIS and therefore determined the event was reportable in accordance with 10 CFR 50.72(b)(3(iv)(A) and 10 CFR 50.73(a)(2)(iv)(A). Susquehanna entered the event into the CAP as CR-2016-08541 and determined the cause was less than adequate procedure use and adherence by Operations staff. Primary corrective actions included coaching and remediation of the individual involved in the event. Inspectors assessed the human performance error and determined that it was a violation of regulatory requirements. However, inspectors determined that the violation was of minor safety significance because it resulted in the actuation of systems and therefore did not have an adverse effect on the cornerstone objectives. The inspectors reviewed the LER as well as Susquehanna's evaluation and corrective actions and did not identify any new issues. This LER is closed.

.5 <u>(Closed) Licensee Event Report (LER) 05000388/2015-007-00</u>: Primary Containment Isolation Valve Failure Due to Sticking/Sluggish Solenoid Valve

On September 30, 2015, the Unit 1 reactor water sample outboard containment isolation valve failed to stroke closed within the TS limit of 2 seconds. Susquehanna entered the issue into the CAP as CR-2015-26590 and evaluated the failure of the valve. Susquehanna determined that the failure was directly attributable to inadequate corrective actions following a failure on July 1, 2015. As such, Susquehanna determined that there was firm evidence that the valve had been inoperable for longer than the TS allowed outage time and consequently determined that the failure was reportable as a condition prohibited by TSs per 10 CFR 50.73((a)(2)(i)(B). In addition to the cause of inadequate corrective actions, Susquehanna determined that the preventative maintenance task was inadequate for this valve because it did not require periodic replacement of the air-operated valve's associated solenoid valve. Corrective actions included replacing the solenoid valve to restore valve operability and revising the generic preventative maintenance task for this valve and its counterpart on Unit 2 to include replacement of the solenoid valve during periodic valve overhauls and diagnostic testing. The enforcement aspects of this LER are discussed in section 1R12 of IR 05000387; 388/2015-004 (ML16040A197). The inspectors reviewed did not identify any new issues during the review of the LER. This LER is closed.

.6 (Closed) Licensee Event Report (LER) 05000387/2015-006-00 and 05000387; 388/2015-006-01: Loss of Safety Function due to Inoperability of Both Trains of the SGTS and a Loss of Safety Function of the Control Room Emergency Outside Air Supply System (CREOASS) due to Air Flow Controller found in Manual

On September 29, 2015, Susquehanna commenced a surveillance test which rendered the 'B' trains of SGTS and CREOASS inoperable. While performing this test, Susquehanna separately commenced testing which rendered the function of the RPV switches associated with PCIS actuations and SCIS actuations, which initiates the logic to start SGTS and CREOASS. Simultaneous performance of these two surveillance rendered both trains of each system inoperable, a loss of safety function of both systems. Susquehanna entered the issue into the CAP as CRs 2015-26442, 2015-26455 and 2015-26475 and reported the loss of safety function in EN 51432. Susquehanna's evaluation determined that the concurrent inoperability of both trains were not recognized when scheduling the two surveillances and noted that instrumentation surveillances are written to test all four instruments channels in the same procedures. Contributing to this, Susquehanna determined that the station had misinterpreted a note that allows delayed entry into the TS action statement to mean that the associated equipment could be considered operable despite not being able to perform their safety function. Corrective actions planned or completed include revising all instrumentation surveillances to clearly identify impacts and effects as well as performing training with applicable station personnel. The enforcement aspects of this LER are discussed in Section 1R15 of IR 05000387; 388/2015-004 (ML16040A197). The inspectors reviewed did not identify any new issues during the review of the LER. These LERs are closed.

.7 (Closed) Licensee Event Report (LER) 05000387; 388/2016-008-00: Inoperability of Diesel Generator Due to Misalignment of MOC Switch Contacts Due to Inadequate Post Maintenance Testing

On April 1, 2016, Unit 2 entered TS 3.8.1 due to the 'A' EDG being inoperable due to misaligned contacts on a MOC switch resulting in fan 1V222A starting without a time delay. Based upon review of the history and cause, the condition likely existed since the supply breaker was replaced on July 19, 2010. Since the condition results in inability to meet SR 3.8.1.18, the A EDG was inoperable during periods when 1V222A would have been available to start without the time delay.

This event was reported as a condition prohibited by TSs, in accordance with 10 CFR 50.73(a)(2)(i)(B), since the inoperability existed for a period of time greater than allowed by the TSs. A review of historical information for the last three years also identified instances in which one of the other EDGs (B, C, or D) was inoperable. Based on this information, this condition was also reported as a condition that could have prevented fulfillment of a safety function in accordance with 10CFR50.73(a)(2)(v)(D).

Susquehanna determined the direct cause of the MOC switch contacts not aligning properly was mis-adjustment of the MOC switch linkage. Additionally, Susquehanna determined that an inadequate PMT for the breaker swap was performed in 2010 because the PMT did not check MOC switch contact alignment. Corrective actions included aligning the MOC switch contacts and revising procedures to include visual inspection to ensure the MOC switch contacts are properly aligned during PMT.

The inspectors reviewed this LER, Susquehanna's evaluation, and associated corrective actions. The enforcement aspects of this finding are discussed in Section 1R12. This LER is closed.

4OA5 Other Activities

Institute of Nuclear Power Operations (INPO) Report Review

a. Inspection Scope

The inspectors reviewed the final report for the INPO plant assessment of Susquehanna conducted in November 2015. The inspectors evaluated these reports to ensure that NRC perspectives of Susquehanna performance were consistent with any issues identified during the assessments. The inspectors also reviewed these reports to determine whether INPO identified any significant safety issues that required further NRC follow-up.

b. Findings

No findings were identified.

4OA6 Meetings, Including Exit

On July 8, 2016, the inspectors presented the inspection results to Mr. T. Rausch, President and Chief Nuclear Officer, and other members of the Susquehanna staff. The inspectors verified that no proprietary information was retained by the inspectors or documented in this report.

ATTACHMENT: SUPPLEMENTARY INFORMATION

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

J. Franke, Site Vice President
B. Franssen, Plant Manager
J. Barnhardt, Dosimetry Supervisor
B. Bridge, Radiation Protection Manager
T. Creasy, System Engineering Manager
C. Fisher, Fire Marshal
F. Hickey, Senior Health Physicist
J. Jennings, Manager- Nuclear Regulatory Affairs
D. Jones, Maintenance General Manager
D. Lamarca, Operations Manager
D. Lock, Programs Engineer Manager
P. Scanlan, Maintenance Manager
A. Schrad, Diesel Engineer
J. Scranton, Engineer
B. Sprung, Regulatory Affairs
S. Maguire, Fire Protection System Engineer
T. McCarthy, Fire Protection Engineer
G. Merenich, Radiation Instruments Supervisor
E. OTruba, Radiation Operations Supervisor
R. Rodrigues-Gilroy, Radiation Operations Supervisor
C. Young, Operations Shift Manager

LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

Opened/Closed

05000387/2016002-01	NCV	Failure to Promptly Identify a Condition Adverse to Quality Associated with Primary Containment Isolation Valves (Section 1R12)
05000387;388/2016002-02	NCV	Failure to Promptly Correct a Condition Adverse to Quality with ‘A’ EDG MOC Switch (Section 1R12)
05000387;388/2016002-03	NCV	Failure of B EDG to Reach Rated Frequency within 10 Seconds (Section 1R22)
05000387;388/2016002-04	NCV	Failure to Critique an Incorrect PAR Notification (Section 1EP6)
05000387;388/2016002-05	NCV	Entry into a High Radiation Area without Radiological Briefing (Section 2RS1)
05000387;388/2016002-06	NCV	Entry into a Locked High Radiation Area without Radiological Briefing (Section 2RS1)

05000388/2016002-07	NCV	HPCI Overridden Prior to Manual Reactor Scram (Section 4OA3)

Closed

05000387/2016-006-00	LER	Implementation of Enforcement Guidance Memorandum (EGM) 11-003, Revision 3 (Section 4OA3)
05000388;387/2015-015-00	LER	Loss of Safety Function due to Inoperability of Both Trains of the Control Room Emergency Outside Air Supply System (Section 4OA3)
05000387/2016-009-00	LER	Valid Primary Containment Isolation Actuation during Local Leak Rate Testing due to Human Performance Error (Section 4OA3)
05000388/2015-007-00	LER	Primary Containment Isolation Valve Failure Due to Sticking/Sluggish Solenoid Valve (Section 4OA3)
05000387/2015-006-00/ 05000387;388/2015-006-01	LER	Loss of Safety Function due to Inoperability of Both Trains of the Standby Gas Treatment Systems (SGTS) and a Loss of Safety Function of the Control Room Emergency Outside Air Supply System (CREOASS) due to Air Flow Controller found in Manual (Section 4OA3)
05000387;388/2016-008-00	LER	Inoperability of Diesel Generator Due to Misalignment of MOC Switch Contacts Due to Inadequate Post Maintenance Testing (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures
GO-100-014, Unit 1 Hot Weather Operation, Revision 3
ON-103-001, GRID Instabilities, Revision 8
NDAP-00-1913, Seasonal Readiness, Revision 5
ON-NATPHENOM-001, Severe Weather/Natural Phenomena, Revision 2

Condition Reports (*NRC identified)
CR-2016-04038
CR-2016-15321

Section 1R04: Equipment Alignment

Procedures
NDAP-QA-0503, General Housekeeping, Transient Material and Internal Cleanliness, Revision 39
SO-024-001A, Monthly Diesel Generator 'A' Operability Test, Revision 26
SE-024-A01, Diesel Generator A Integrated Surveillance Test, Revision 7
SM-024-006A, Emergency Diesel Generator A 60 Month Electrical Inspection, Revision 2
SI-024-301, 5 Year Calibration of 'A' Diesel Generator Lube Oil Low Pressure Switches PSL-03468A1, A2, A3, A4, Revision 11
SM-024-A02, 60 Month 4KV Emergency Diesel Generator A Differential Relay Calibration, Revision 10
OP-024-001, Diesel Generators, Revision 81
OP-024-005, Diesel Generator Start Log, Revision 8

Condition Reports (*NRC identified)

CR-2015-12804	CR-2015-12916	CR-2016-02098	CR-2016-03167
CR-2016-08441	CR-2016-08826*	CR-2016-08828*	CR-2016-08831*
CR-2016-08832*	CR-2016-13440	CR-2016-14085	CR-2016-14105

Drawings
M-153, SES Unit 1 P&ID Fuel Pool Cooling and Clean-Up, Sheet 1, Revision 42
M-153, SES Unit 1 P&ID Fuel Pool Cooling and Clean-Up, Sheet 2, Revision 14
M-2153, SES Unit 2 P&ID Fuel Pool Cooling and Clean-Up, Sheet 1, Revision 33
M-111, SES Common P&ID ESW System, Sheet 1, Revision 50
M-111, SES Unit 1 P&ID ESW System "A" Loop, Sheet 2, Revision 53
M-2111, SES Unit 2 P&ID ESW System "A" Loop, Sheet 1, Revision 46

Maintenance Orders/ Work Orders

1162815	1195250	1195252	1195748	1675792	1989960
1997775					

Section 1R05: Fire Protection

Procedures
FP-013-236, 'E' Diesel Generator Building Fire Zone 0-41E Elevation 656'-6", 675-6", 708'-0", Revision 6
FP-113-100, Drywell (I-400, I-516, I-607) Fire Zone 1-4F Elevation 704' Thru 807', Revision 3
OI-013-001, Fire Protection Component Technical Data, Revision 33
NDAP-QA-0503, General Housekeeping, Transient Material and Internal Cleanliness, Revision 39
NDAP-QA-0442, Control of Ignition Sources, Cutting, Welding and Hot Work Permits, Revision 14
NDAP-QA-0440, Control of Transient Combustible/Hazardous Materials, Revision 19
NDAP-QA-0446, Fire Barrier Program, Revision 9
NDAP-QA-0449, Fire Protection Program, Revision 14
NDAP-QA-0444, Fire Alarm Response, Revision 5
NDAP-QA-0443, Firewatch Procedure, Revision 13

Miscellaneous

FP-013-189, Diesel Generator Bay ‘A’ Fire Zone 0-41A Elevations 677’, 660’ and 710’, Revision 4

FP-113-109, Remote Shutdown Panel Room (I-109), Revision 5

FP-113-119, Circulation Space (I-500) and Adjacent Rooms, Revision 6

Section 1R06: Flood Protection Measures

Drawings

C-2737, Unit 2 Reactor Building Station Flood Barrier Plan on El. 749’-1”, Sheet 1, Revision 2

Miscellaneous

EC-RISK-0539, Internal Flooding Analysis for PRA, Revision 2

EC-FLOD-0001, Internal Flooding Evaluations for Moderate Energy Pipe Cracks and Sprinkler system Actuations, Revision 3

NDAP-0A-0409

Section 1R11: Licensed Operator Requalification Program

Procedures

GO-200-002, Plant Startup, Heatup and Power Operation, Revision 88

Condition Reports (*NRC identified)

CR-2016-14070*

Section 1R12: Maintenance Effectiveness

Condition Reports (*NRC identified)

CR-1572815	CR-2014-12451	CR-2014-13556	CR-2014-16323
CR-2016-06960	CR-2016-07659	CR-2016-08987*	CR-2016-09940
CR-2016-12047*	CR-2016-13058*		

Action Requests

AR-2014-19673	AR-2015-16934	AR-2015-17319	AR-2016-07508

Drawings

M-141, SES Unit 1 P&ID Nuclear Boiler, Sheet 2, Revision 19

E-214, SES Common Schematic Diagram HVAC Control Strc Chilled Water System Loop Reset, Sheet 9A, Revision 6

V-176, SES Unit 1 Logic Diagram Reactor Bldg Zone 1 HVAC SWGR LC Rm CLG Fan IV-222A, Sheet 12, Revision 1

Maintenance Orders/ Work Orders

905365	1085198	1256443	1291117	1310732	1440203
1575292	1578728	1586540	1661338	1716982	1719979
1980428					

Miscellaneous

EC-059-1026, Containment Isolation Design Requirements for the Feedwater Penetrations (X-9 A/B) to Resolve CR 96-1407 (96-00046)

IOM 211, Indoor Metal Clad Switchgear 5kv Class/E or 4kv Switchgear for Engineered Safeguard Systems”, Revision 28

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures
NDAP-QA-0480, ASME Section XI System and Component Pressure Testing, Revision 9
SE-100-002, ASME Class I Boundary System Leakage Test, Revision 25
NDAP-QA-1902, Integrated Risk Management, Revision 23
PSP-26, Online and Shutdown Nuclear Risk Assessment Program, Revision 17
OP-AD-300, Administration of Operations, Revision 15
SI-283-322, Quarterly Calibration of Automatic Depressurization System (ADS) Timers B21C-K5A and B21C-K5B, Revision 15
SI-283-329, Quarterly Calibration of ADS Drywell Pressure Bypass Timers B21C-K114A,B,C,D, Revision 9

Condition Reports (*NRC identified)

CR-2015-30901	CR-2016-10222	CR-2016-10223	CR-2016-10231
CR-2016-10235	CR-2016-10210	CR-2016-10213*	CR-2016-14553
CR-2016-14663	CR-2016-14729	CR-2016-14755	

Drawings
M-143, SES Unit 1 P&ID Reactor Recirculation, Sheet 4, Revision 1

Maintenance Orders/ Work Orders

1588654	1639423	1946794	1947746	1955113	1955123
1983622	1983623	1984404			

Section 1R15: Operability Determinations and Functionality Assessments

Procedures
SO-070-A01, Monthly Standby Gas Treatment Train A, Revision 3
SI-183-325, 24 Month Calibration of MSIV RPS Limit Switches ZS-14122A-D and ZS-14128A-D, Revision 22
NDAP-QA-0703, Operability Determinations and Functionality Assessments, Revision 29
LS-120, Issue Identification and Screening Process, Revision 6

Condition Reports (*NRC identified)

CR-2016-09791	CR-2016-10615*	CR-2016-11018	CR-2016-11038
CR-2016-11078	CR-2016-11099	CR-2016-11204	CR-2016-11217
CR-2016-12701			

Drawings
FF62069, 26” MSIV, Sheet 2, Revision 3
E-301, Schematic Diagram Computer Digital Inputs Unit 1, Sheet 101, Revision 11
E-170, SES Unit 1 Schematic Diagram NSSSS Main Steam Inboard Isolation Valve Indication, Sheet 7, Revision 11
E-170, Block Diagram NSSSS Isolation Logic Unit 1, Sheet 1, Revision 26
E-157, Block Diagram Reactor Protection System Trip Ckts and RPS Valves Unit 1, Sheet 2, Revision 20
FF110100, 14”-900 Weld Ends Pressure Seal, Flex Wedge, Carbon Steel, Gate Valve with SB-3-150 Limitorque Oper., Sheet 1701, Revision 10

Maintenance Orders/ Work Orders

106227	1817099	1947648	1989377	1989571	1989794
1994704	1994711				

Miscellaneous
EC-070-0013, SBGT System Charcoal Filters, Revision 1
Engineering change 1989557
IOM, Flex –Wedge Gate Valves, Globe Valves, and Check Valves, Revision 5
M-VLV-420, MOV Data Detail, Limit Switch Settings and Torque Switch Settings for HV 255F006, Revision 1

Section 1R18: Plant Modifications

Procedures
ME-0RF-102, LPRM Replacement, Revision 23

Condition Reports (*NRC identified)

CR-394803	CR-766227	CR-1014272	CR-1553574
CR-1576913	CR-388387	CR-2015-30901	CR-2016-09391*
CR-2016-09394*	CR-2016-09399*	CR-2016-06766	CR-2016-07811
CR-2016-14544	CR-2016-14739	CR-2016-14876	CR-2016-15930
CR-2016-16043			

Maintenance Orders/ Work Orders
1946794

Action Requests

AR-2015-02093	AR-2015-16817	AR-2016-14797

Drawings
SPDCA110-3, SES Unit 1 Reactor Building RHR Drain and Crosstie Line, Sheet 1
SPDCA110-4, SES Unit 1 Reactor Building RHR NSSS, Sheet 1, Revision 1
SPDCA110-5, SES Unit 1 Isometric Reactor Building RHR Crosstie Line, Sheet 1
SPDCA110-6, SES Unit 1 Isometric Reactor Building RHR Crosstie Line, Sheet 1
SPDCA110-7, SES Unit 1 Isometric Reactor Building RHR Crosstie Line, Sheet 1
SPDCA110-H11, SES Unit 1 Pipe Support Reactor Building Inside Containment, Sheet 1
M-151, SES Unit 1 P&ID RHR, Sheet 3, Revision 32
M-151, SES Unit 1 P&ID RHR, Sheet 1, Revision 72
M-143, SES Unit 1 P&ID Reactor Recirculation, Sheet 4, Revision 1

Miscellaneous
EC-1846732, Ensure Positive Seating of RHR Testable Check Valves HV151F050A & HV151F050B
Hot Box 16-09
EC-049-0001, Pressure Drops in RHR System for Various Modes of Operation, Revision 10
EC-PIPE-16377, Unit 1 RHR 50 Valve Crosstie Line, Revision 0
EC-049-0035, RHR TS Test Pressure, Revision 5
EC-2000892, Modify SPDCB126-1 Piping
EC/BTT-1999985, Replace RxR Pump Cooler Assembly
003N7138-R0, Fracture Mechanics Evaluations for the Determination of Weld Overlay Size for In-Core Monitor Housing for SSES Unit 1
EC-2002083, Repair through Wall Leak on Unit 1 LPRM 24-09 (LPRM12409) Housing
003N7113, ICMH Weld Overlay ASME Code Section XI Repair Plan, Revision 0
Code Program Form Number 16-162-2001157-068
BOP-UT-16-083, UT Calibration/Examination
EC-035-1027, Determination of Maximum Hole Size Permitted under NRC EGM 11-003
EC-22A2019, In-Core Housing, Stress Report, Revision 3

Section 1R19: Post-Maintenance Testing

Procedures
MT-EO-059, Static or Dynamic testing of Motor Operated Valves Using Quiklook LL. Revision 5
MT-083-012, MSIV Diagnostic Testing, Revision 8
SM-183-004, MSIV Fail Safe Testing, Revision 3
TP-149-080, Initial Start and Run-in of New or Repaired RHR Pump Motor, Revision 5
SO-149-A02, Quarterly RHR System Flow Verification Division I, Revision 25
MT-050-003, RCIC Pump Turbine Disassembly and Reassembly, Revision 15
SO-150-005, 24 Month RCIC Flow Verification, Revision 21
SO-150-002, Quarterly RCIC Flow Verification, Revision 53
TP-150-004, RCIC Turbine Overspeed Trip Testing with Auxiliary Steam, Revision 18
SE-149-203, 1035 PSIG Leak Rate Testing of RHR Shutdown Cooling Isolation Valves, Revision 18
SE-149-202, 1035 PSIG Leak Rate Testing of LPCI Loop B Injection Pressure Isolation Valves, Revision 14
SE-149-201, 1035 PSIG Leak Rate Testing of LPCI Loop A Injection Pressure Isolation Valves, Revision 15

Condition Reports (*NRC identified)

CR-1701593	CR-2014-26599	CR-2016-06619	CR-2016-06871
CR-2016-09317	CR-2016-10218	CR-2016-10270	CR-2016-10310
CR-2016-10337	CR-2016-10395	CR-2016-10714	CR-2016-10775
CR-2016-10801	CR-2016-10894	CR-2016-13075	CR-2016-14391
CR-2016-14679			

Action Requests
AR-2016-14522

Maintenance Orders/ Work Orders

1251997	1279794	1846878	1891735	1891750	1891755
1968710	1983475	1981187	1984434	1999987	2000299

Miscellaneous
IOM, Reactor Cooling Isolation Cooling Turbine Unit 1, Revision 22

Section 1R20: Refueling and Other Outage Activities

Procedures
GO-100-004, Plant Shutdown to Minimum Power, Revision 77
GO-100-005, Plant Shutdown to Hot/Cold Shutdown, Revision 69
GO-100-006, Cold Shutdown, Defueled and Refueling, Revision 57
GO-100-010, ECCS/Decay Heat Removal in Mode 4, 5 or Defueled, Revision 29
GO-100-002, Plant Startup, Heatup and Power Operation, Revision 104

Action Requests
AR-2016-13519

Condition Reports (*NRC identified)

CR-2016-05258	CR-2016-06114	CR-2016-06158	CR-2016-06162
CR-2016-06245	CR-2016-06276	CR-2016-06290	CR-2016-06320
CR-2016-06574	CR-2016-06649	CR-2016-06651	CR-2016-06667
CR-2016-06712*	CR-2016-06715*	CR-2016-06718*	CR-2016-06723
CR-2016-06728	CR-2016-06753	CR-2016-06875	CR-2016-06881
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Section 1R22: Surveillance Testing

Procedures

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SO-153-003A, 24 Monthly SBLC Operability (Loop A), Revision 5
SE-100-003, Primary Containment Integrated Leakage Rate Test (ILRT) (Special, Infrequent or Complex Test/Evolution), Revision 10
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SE-024-100, Unit 1 and Unit 2 Ten Year Simultaneous Start of Four Emergency Diesel Generators, Revision 5

Condition Reports (*NRC identified)

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Action Requests

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Section 1EP6: Drill Evaluation

Procedures
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EO-000-103, Primary Containment Control, Revision 16

Condition Reports (*NRC identified)

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Section 2RS1: Access Control to Radiologically Significant Areas

Procedures
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RP-122, Radiation Protection Stop Work Authority, Revision 0
HP-TP-500, Health Physics Radiological Survey Program, Revision 51
RP-180, Radiological Postings, Labelings, and Markings, Revision 3
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Condition Reports (*NRC identified)

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Section 2RS2: Occupational ALARA Planning and Controls

Procedures
HP-AL-400, RWP ALARA Reviews and Evaluations, Revision 21
NDAP-QA-1191, ALARA Program and Policy, Revision 23
HP-TP-103, Plant Radiation Profile, Revision 4
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Condition Reports (*NRC identified)
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Section 2RSO5: Radiation Monitoring Instrumentation

Procedures
HP-TP-047, HP Instrument Lab Work Activities, Revision 5
HP-TP-108, Calibration of the ASP-1/AC-3 (Alpha Meter), Revision 5
HP-TP-117, Calibration of the Eberline AMS-4, Revision 9
HP-TP-127, Calibration of the A<P 50-100-200 Area Radiation Monitors, Revision 12
HP-TP-133, RP Instrument Checks, Revision 45
HP-TP-134, Calibration of the Fluke 451B, Revision 5
HP-TP-147, Calibration of the Canberra GEM5 Portal Monitor, Revision 8
HP-TP-166, Calibration of the Canberra ARGOS 5AB, Revision 11
HP-TP-177, Calibration of Friskers, Revision 16
HP-TP-201, Operation of the Whole Body Counting System Using APEX-INVIVO Software, Revision 2
HP-TP-208, Performance Verification and Calibration of the Whole Body Counting System, Revision 14
HP-TP-249, Calibration and Testing of Health Physics Counting Equipment, Revision 26
HP-TP-443, Use of Radiation Detection Equipment, Revision 35
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SH-179-003, 24 Month Radiation Source check of the Containment Monitoring System Channels 15720A and 15720B, Revision 15
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Source Certificate of Calibration 2013-0014
Source Certificate of Calibration 2005-0077
Source Certificate of Calibration 2005-0079
Source Certificate of Calibration 2005-0080

Section 4OA1: Performance Indicator Verification

Procedures
SC-176-102, Unit 1 Primary Coolant Specific Activity- Dose Equivalent I-131, Revision 15
CH-SY-013, Station Sample Collection, Revision 10
OI-AD-094, NRC Performance Indicator Monthly Update Reactor Coolant System Total Leakage (RCSL), Revision 7

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Section 4OA2: Problem Identification and Resolution

Procedures
Appendix OPS-1P, Quality assurance for the Fire Protection Program and Related system, Revision 5
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LS-120, Issue Identification and Screening Process, Revision 6
LS-125, Corrective Action Program, Revision 4
NDAP-QA-0440, Control of Transient Combustible/Hazardous Materials, Revision 19
NADP-QA-0442, Control of Ignition Sources, Cutting, Welding, and Hot Work Permits, Revision 11
OI-013-001, Fire Protection Component Technical Data, Revision 33
OI-013-002, Fire Risk Management, Revision 9

Condition Reports (*NRC identified)

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CR-2015-05614	CR-2015-15635	CR-2015-16689	CR-2015-16958
CR-2015-17402	CR-2015-20254	CR-2015-22158	CR-2015-22971
CR-2015-26472	CR-2015-30935	CR-2015-32385	CR-2015-33747
CR-2015-34076	CR-2016-00747	CR-2016-00839	CR-2016-00851
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Drawings
E205951, Unit 1 Reactor Bldg. Fire Zone Plan Elevation 683'-0'', Sheet 1, Revision 13
E205952, Unit 1 Reactor Bldg. Fire Zone Plan Elevation 719'-1'', Sheet 1, Revision 12
E205953, Unit 1 Reactor Bldg. Fire Zone Plan Elevation 749'-1'', Sheet 1, Revision 11
E205954, Unit 1 Reactor Bldg. Fire Zone Plan Elevation 779'-1'', Sheet 1, Revision 9
E205955, Unit 1 Reactor Bldg. Fire Zone Plan Elevation 799'-1'', Sheet 1, Revision 12
E205986, Unit 1 & 2 Control Structure Fire Zone Plan Elevation 676'-0'', Sheet 1, Revision 11
E205986, Unit 1 & 2 Control Structure Fire door and Fire Dampers Elevation 676'-0'', Sheet 2, Revision 9
E205986, Unit 1 & 2 Control Structure Fire Protection Plan Elevation 676'-0'', Sheet 3, Revision 9
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027-16, Combustible or Hazardous Material Permit, dated 2/10/2016
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ZWO-1968479, Combustible Storage Per Permit 027-16, dated 2/10/2016
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Fire Protection Impairments (Non TRO) as of 2/22/2016
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EC-013-1040, Combustible Loading Data for input into to Crimp, Revision 5
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Section 4OA3: Follow-up of Events and Notices of Enforcement Discretion

Procedures
OP-252-001, HPCI System, Revision 59
GO-200-002, Plant Startup, Heatup and Power Operation, Revision 88
ON-YPNL-201, Loss of Instrument Bus, Revision 4
ON-4KV-201, Loss of 4KV Bus, Revision 1
OP-AD-300, Administration of Operations, Revision 4
ON-SCRAM-201, Reactor Scram, Revision 4

Condition Reports (*NRC identified)

CR-2015-26475	CR-2015-26590	CR-2016-03713	CR-2016-03909
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CR-2016-12636	CR-2016-12640	CR-2016-12645	CR-2016-12658
CR-2016-12659	CR-2016-12666	CR-2016-12680	CR-2016-12681
CR-2016-12684	CR-2016-12686	CR-2016-12692	CR-2016-12697
CR-2016-12703	CR-2016-12707	CR-2016-12708	CR-2016-12710
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Action Requests

AR-2015-01726	AR-2015-33631	AR-2016-12853

Miscellaneous

Hot Box 16-11
Operations Directive 16-01
Hot Box 16-12

LIST OF ACRONYMS

AC	alternating current
ACE	apparent cause evaluation
ADAMS	Agencywide Documents Access and Management System
ALARA	as low as is reasonably achievable
CAP	corrective action program
CFR	Code of Federal Regulations
CR	condition report
CREOASS	control room emergency outside air supply system
DEP	drill and exercise performance
ECCS	emergency core cooling system
EDG	emergency diesel generator
EGM	enforcement guidance memorandum
ERO	emergency response organization
FSAR	Final Safety Analysis Report
HPCI	high pressure coolant injection
HRA	high radiation area
IMC	inspection manual chapter
kV	kilovolt
LCO	limiting condition of operation
LER	licensee event report
LHRA	locked high radiation area
LOCA	loss of coolant accident
LOOP	loss of off-site power
LPCI	low-pressure coolant injection
LPRM	local power range monitor
MOC	mechanism-operated cell
NCV	non-cited violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NUREG	NRC technical report designation
OPDRV	operation with a potential for draining the reactor vessel
ORO	offsite response organization
OWA	operator workarounds
PAR	protective actions recommendation
PCIS	primary containment isolation system
PCIV	primary containment isolation valve
PI	performance indicator
PMT	post maintenance testing
PS	planning standard
RCIC	reactor core isolation cooling
RHR	residual heat removal
RP	radiation protection
RPV	reactor pressure vessel
RRP	reactor recirculation pump
RSPS	risk significant planning standard
RWCU	reactor water cleanup
RWP	radiation work permit
SCIS	secondary containment isolation system
SDP	significance determination process
SGTS	standby gas treatment systems

SPAR	standardized plant analysis risk
SR	surveillance requirement
SSC	structure, system, and component
SSES	Susquehanna Steam Electric Station
SSO	senior state official
TDM	technical decision making
TS	technical specification
UFSAR	Updated Final Safety Analysis Report
VHRA	very high radiation area