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

May 12, 2016

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

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION – INTEGRATED INSPECTION REPORT 05000387/2016001 AND 05000388/2016001

Dear Mr. Rausch:

On March 31, 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 April 22, 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 two findings of very low safety significance (Green) in this report. Both of these findings involved violations of NRC requirements. Additionally, NRC inspectors documented one Severity Level IV violation with no associated finding. Further, inspectors documented two licensee-identified violations which were determined to be of very low safety significance in this report. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest the NCVs 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 website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

***/RA/***

Daniel L. Schroeder, Chief
Reactor Projects Branch 4
Division of Reactor Projects

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

Enclosure:
Inspection Report 05000387/2016001
and 05000388/2016001
w/Attachment: Supplementary Information

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**U.S. NUCLEAR REGULATORY COMMISSION**

REGION I

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| Docket Nos.: | 50-387 and 50-388 |
| License Nos.: | NPF-14 and NPF-22 |
| Report No.: | 05000387/2016001 and 05000388/2016001 |
| Licensee: | Susquehanna Nuclear, LLC (Susquehanna) |
| Facility: | Susquehanna Steam Electric Station, Units 1 and 2 |
| Location: | Berwick, Pennsylvania |
| Dates: | January 1, 2016 through March 31, 2016 |
| Inspectors: | J. Greives, Senior Resident Inspector<br>T. Daun, Resident Inspector<br>P. Meier, Project Engineer<br>N. Floyd, Reactor Inspector |
| Approved By: | Daniel L. Schroeder, Chief<br>Reactor Projects Branch 4<br>Division of Reactor Projects |

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## SUMMARY

IR 05000387/2016001, 05000388/2016001; January 1, 2016 to March 31, 2016; Susquehanna Steam Electric Station, Units 1 and 2; Follow-Up of Events and Notices of Enforcement Discretion, Maintenance Risk Assessments and Emergent Work Control, and Problem Identification and Resolution.

This report covered a three-month period of inspection by resident inspectors and announced baseline inspections performed by regional inspectors. The inspectors identified three 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,” 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 NUREG-1649, “Reactor Oversight Process,” Revision 5.

### Cornerstone: Initiating Events

- Green. A self-revealing finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” was identified for Susquehanna’s failure to establish measures to assure a condition adverse to quality was corrected. Specifically, vibration induced fatigue cracking on the Unit 1 ‘B’ reactor recirculation pump (RRP) lower seal cavity vent piping was not corrected in December 2014 after a reactor coolant pressure boundary leak had occurred. This resulted in another reactor coolant pressure boundary leakage at the same location with Unit 1 operating in Mode 1, a condition prohibited by technical specifications (TS) LCO 3.4.4. Susquehanna’s entered the issue into the corrective action program (CAP) as CR-2015-30901 and replaced and modified the union that included the weld.

  The finding was more than minor because it was associated with the Equipment Performance attribute of the Initiating Events cornerstone and affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors evaluated the finding in accordance with Exhibit 1 of IMC 0609, Appendix A, “The significance determination process (SDP) for Findings At-Power,” and determined the finding was of very low safety significance (Green) because the leakage would not have exceeded the reactor coolant system (RCS) leak rate for a small loss of coolant accident (LOCA) and it did not affect other systems used to mitigate a LOCA. This finding had a cross-cutting aspect in the area of Human Performance, Work Management, because Susquehanna did not implement a process of planning, controlling, and executing work activities such that nuclear safety is the overriding priority, in that Susquehanna did not adequately coordinate the work activities with different groups [H.5]. Specifically, welding engineers were not engaged in the decision making process during the December 2014 repair and consequently the repair was inadequate to ensure the entire crack had been removed. [4OA3]

**Cornerstone: Mitigating Systems**

- Green. The inspectors identified a Green NCV of 10 CFR 50.65(a)(4) because Susquehanna did not adequately assess the risk of performing maintenance in accordance with station procedures. Specifically, Susquehanna did not assess the risk of performing a standby liquid control (SLC) system flow surveillance in conjunction with having the 'D' emergency diesel generator (EDG) unavailable and therefore did not specify appropriate risk management actions (RMAs). Susquehanna entered the issue into the CAP as CR-2016-04137.

  The inspectors determined that this performance deficiency is more than minor because it was associated with the Human Performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Additionally, the finding is similar to example 7.e. in NRC IMC 0612 Appendix E, "Examples of Minor Issues." This example states, in part, that failure to perform an adequate risk assessment when required by 10 CFR 50.65 (a)(4) is not minor if the overall elevated plant risk would put the plant into a higher licensee-established risk category or would require, under plant procedures, RMAs or additional RMAs. In this case, the combination of the 'D' EDG maintenance and SLC flow surveillance resulted in changing risk to Yellow which required additional RMAs in accordance with station procedures. The inspectors evaluated the finding using IMC 0609 Appendix K, "Maintenance Risk Assessment and Risk Management SDP." The inspectors and the Region I senior resident analyst used Appendix K, Flowchart 1, "Assessment of Risk Deficit," and determined that the inadequate risk assessment was of very low safety significance (Green). The basis for this determination was that the short duration of the actual planned maintenance activities (3.5 hours) associated with the SLC unavailability results in less than E-9 calculated incremental core damage probability deficit (ICDPD) using Susquehanna's risk model. Since the resultant ICDPD is below 1 E-8 threshold, the finding was determined to be Green. This finding was determined to have a cross-cutting aspect in the area of Human Performance, Work Management in that Susquehanna did not appropriately incorporate insights from probabilistic risk assessments into the daily work activities [H.5]. Specifically, Susquehanna did not appropriately assess the risk of performing maintenance activities as specified in station procedures. [1R13]

**Cornerstone: Other**

- SLIV. Inspectors identified a Severity Level IV NCV of 10 CFR Part 50.73 (a)(2)(v) when Susquehanna did not submit a licensee event report (LER) within 60 days of identifying that both trains of the control room emergency outside air supply system (CREOASS) were rendered inoperable during surveillance testing, a condition that could have prevented fulfillment of a safety function. Susquehanna entered the issue into the CAP as CR-2016-03713 and reported the condition on May 5, 2016 in LER 50-388(387)/2015-015.

  Since the issue had the potential to affect the NRC's ability to perform its regulatory function, the inspectors evaluated this performance deficiency in accordance with the traditional enforcement process. Using example 6.9.d.9 from the NRC Enforcement Policy, the inspectors determined that it was a Severity Level IV violation. The significance of the associated performance deficiency was also screened against the reactor oversight process (ROP) per the guidance of IMC 0612, Appendix B, "Issue Screening." Because this violation

involves the traditional enforcement process and does not have an associated finding under the ROP, inspectors did not assign a cross-cutting aspect to this violation. [4OA2]

**Other Findings**

Violations of very low safety significance that were identified by Susquehanna were reviewed by the inspectors. Corrective actions taken or planned by Susquehanna have been entered into Susquehanna's CAP. These violations and corrective action tracking numbers are listed in Section 4OA7 of this report.

# REPORT DETAILS

Summary of Plant Status

Unit 1 began the inspection period at 100 percent rated thermal power (RTP). On January 8, 2016, operators reduced reactor power to approximately 75 percent RTP to perform a rod pattern adjustment and power was restored to 100 percent the following day. On January 16, 2016, operators reduced power to 85 percent to perform a planned rod pattern adjustment and power was restored to 100 percent on January 17, 2016. On January 24, 2016, operators reduced power to 80 percent to perform a planned rod pattern adjustment and power was restored to 100 percent on January 25, 2016. On January 30, 2016, operators commenced an end of cycle coast down for a planned refueling outage. During the coast down, operators reduced power from 89 percent to 67 percent to perform a planned rod pattern adjustment. Power was restored to approximately 87 percent and the coast down was resumed on March 3, 2016. On March 11, 2016, operators performed a shutdown for a planned refueling outage. Unit 1 remained shutdown in Mode 5 for the remainder of the inspection period.

Unit 2 began the inspection period at 100 percent power and operated at full power until January 15, 2016, when operators reduced reactor power to 71 percent RTP for a planned rod sequence exchange. Power was restored to 100 percent on January 16, 2016 and remained at or near 100 percent power for the remainder of the inspection period.

## 1. REACTOR SAFETY

**Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity**

1R04 Equipment Alignment

.1 Partial System Walkdowns (71111.04 – 4 samples)

a. Inspection Scope

The inspectors performed partial walkdowns of the following systems:

- Unit 1, ‘A’ core spray (CS) during ‘B’ CS system outage window (SOW) on January 13, 2016
- Common, restoration of ‘E’ EDG following mid-cycle overhaul on February 1, 2016 (vertical inspection)
- Unit 1, division II residual heat removal (RHR) during division I RHR SOW on February 10, 2016
- Unit 1, division I shutdown cooling on March 13, 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 UFSAR, TS, work orders, condition reports, 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.

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.

- Unit 1, general access area and pump room, elevation 779' (fire zone 1-6A) on February 23, 2016
- Unit 1, access corridor, elevation 749' (fire zone 1-5A-W) on February 23, 2016
- Unit 1, 4.16kv switchgear room division II, elevation 749' (fire zone 1-5F) on February 23, 2016
- Unit 1, condenser bay (fire zones 1-31D, 1-32D and 1-33C) on March 13, 2016
- Unit 1, drywell (fire zone 1-4F) on March 13, 2016

b. Findings

No findings were identified.

1R07 Heat Sink Performance (711111.07A – 1 sample)

a. Inspection Scope

The inspectors reviewed the Unit 1 'A' RHR heat exchanger readiness and availability to perform its safety functions. The inspectors reviewed the design basis for the component and verified Susquehanna's commitments to NRC Generic Letter 89-13, "Service Water System Requirements Affecting Safety-Related Equipment." The inspectors reviewed the results of previous inspections of the Unit 1 'A' RHR heat exchanger and similar heat exchangers. The inspectors discussed the results of the most recent inspection with engineering staff and reviewed pictures of the as-found and as-left conditions. The inspectors verified that Susquehanna initiated appropriate corrective actions for identified deficiencies. The inspectors also verified that the

number of tubes plugged within the heat exchanger did not exceed the maximum amount allowed.

b. Findings

No findings were identified.

1R08 In-service Inspection (71111.08 - 1 sample)

a. Inspection Scope

From March 21, 2015 to March 24, 2016, the inspectors conducted an inspection and review of in-service inspection (ISI) activities in order to assess the effectiveness of Susquehanna's program for monitoring degradation of the RCS boundary, risk-significant piping boundaries, and the containment system boundaries during the Susquehanna Steam Electric Station Unit 1 19th refueling outage.

Non-destructive Examination and Welding Activities (IMC Section 02.01)

The inspectors observed a sample of in-process non-destructive examinations (NDE), reviewed completed documentation, and interviewed Susquehanna personnel to verify that the NDE activities performed as part of the fourth interval, first period, of the Susquehanna Unit 1 ISI program were conducted in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 2007 Edition with 2008 Addenda. For augmented examinations, the inspectors verified that activities were performed in accordance with Susquehanna's augmented inspection program and procedures, and with any applicable industry guidance documents. The inspectors verified that indications and defects, if present, were dispositioned in accordance with the ASME Code or an NRC approved alternative, and verified that relevant indications were compared to previous examinations to determine if any changes had occurred.

Activities included a review of ultrasonic testing (UT), liquid penetrant testing (LPT), and visual testing (VT). The inspectors reviewed certifications of the NDE technicians performing the examinations and verified that the inspections were performed in accordance with qualified NDE procedures and industry guidance. For those UT activities observed, the inspectors also verified the calibration of equipment used to perform the examinations. The inspectors verified that the test results were reviewed and evaluated by certified Level III NDE personnel and that the parameters used in the test were in accordance with the limitations, precautions, and prerequisites specified in the test procedure.

ASME Code Required Examinations:

- Direct observation of the manual UT of the N1B nozzle-to-vessel weld and the nozzle inner radius on the recirculation system suction.

- Documentation review of the automated phased array UT of the N2J nozzle-to-safe end weld on the recirculation system discharge.

- Direct observation of the LPT of two socket welds (FW-10 and -14 in SPDCA110-6), performed as part of a modification activity in the residual heat removal system.

- Documentation review of the VT of the drywell (i.e. containment) and suppression pool interior penetrations and surfaces. The inspectors independently examined the condition of the drywell liner surfaces at accessible floor elevations and compared those to the inspector walkdowns.

Other Augmented, License Renewal or Industry Initiative Examinations:

- Direct observation of the remote enhanced VT records of the reactor vessel internals during in-vessel visual inspection activities in accordance with BWRVIP-03. Specifically, the inspectors reviewed the jet pump components including the main wedges, slip joints, set screw auxiliary wedges, and the anti-vibration system modification.

Review of Previous Indications

The inspectors reviewed the volumetric examination of the N2J nozzle-to-safe end weld that had a previously recorded intergranular stress corrosion cracking indication, which subsequently received a weld overlay in 2004. The inspectors directly observed the encoded UT data from 2004 and 2008 with the NDE Level III analyst and reviewed the completed UT report from this outage. The inspectors verified the original flaw did not change compared to the previous examinations and was acceptable for continued service.

Welding on Pressure Boundary Systems

The inspectors reviewed the pressure boundary risk-significant welding activity, including the associated NDE, of two socket welds (FW-10 and -14 in SPDCA110-6) as part of a modification on the RHR system. Specifically, the scope of the modification was to install a valve cross-tie line in order to reduce the vibrations on the unit 1 RHR testable check valves. The inspectors performed a direct observation of the welding activities to verify that the welding, LPT/VT examinations, and final acceptance were performed in accordance with the ASME Code requirements. The inspectors reviewed the weld procedure specification to ensure it contained the required essential and supplemental essential weld variables and that those variables were within the ranges demonstrated by the supporting qualification record. The modification was performed under work order 1891735.

Identification and Resolution of Problems (IMC Section 02.05)

The inspectors reviewed a sample of Susquehanna Steam Electric Station Unit 1 corrective action reports, which identified NDE indications, deficiencies, and other non-conforming conditions since the previous refueling outage and during the current outage. The inspectors verified that non-conforming conditions were properly identified, characterized, evaluated, and that corrective actions were identified and entered into the CAP for resolution.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program and Licensed Operator Performance (71111.11Q – 2 samples)

.1 Quarterly Review of Licensed Operator Requalification Testing and Training

a. Inspection Scope

The inspectors observed licensed operator simulator training on February 12, 2016, which included small break LOCA coincident with a loss of offsite power (LOOP) and the failure of the high pressure coolant injection system to automatically start as required. 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

a. Inspection Scope

On March 12, 2016, inspectors observed the control room operators perform a planned reactor shutdown for the Unit 1 refueling outage. 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 crew during the evolution 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 – 2 samples)

a. Inspection Scope

The inspectors reviewed the samples listed below to assess the effectiveness of maintenance activities on structure, system, and component (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.

- Primary containment isolation valve (PCIV) quarterly valve stroke timing on January 26, 2016
- Review of maintenance performed during ‘E’ EDG mid-cycle overhaul from January 25, 2016 through February 8, 2016 (vertical inspection)

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 – 5 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 2, battery charger 2D643 emergent maintenance on January 7, 2016
- Unit 1, elevated risk during ‘B’ CS SOW on January 14, 2016
- Common, elevated risk during ‘E’ EDG mid-cycle overhaul on January 28, 2016
- Unit 1, elevated risk during division I RHR SOW on February 9, 2016
- Unit 2, yellow risk during SLC flow surveillance on February 16, 2016

b. Findings

Introduction. The inspectors identified a Green NCV of 10 CFR 50.65(a)(4) because Susquehanna did not adequately assess the risk of performing maintenance in accordance with station procedures. Specifically, Susquehanna did not assess the risk of performing a SLC system flow surveillance in conjunction with having the ‘D’ EDG unavailable and therefore did not specify appropriate RMAs.

Description. 10 CFR 50.65(a)(4) requires licensees to assess and manage the risk that may result from a proposed maintenance activity prior to performing the maintenance. Susquehanna implements this requirement with NDAP-QA-1902, “Integrated Risk Management,” Revision 22, which states that its purpose is to provide “administrative controls to assess and manage all aspects of nuclear safety risk associated with performance of work activities.”

On February 18, 2016, inspectors reviewed the online risk assessments for maintenance that was performed on February 17 on Unit 2. Susquehanna communicated the online risk as Green while performing a flow surveillance on the SLC system in conjunction with the ‘D’ EDG supply fan being out-of-service for a routine inspection. Inspectors questioned the work control center about the status of the ‘D’ EDG during the maintenance because their risk assessment did not include any impairment. Through consultation with operations and engineering personnel, Susquehanna determined that the ‘D’ EDG was in fact unavailable and should have been entered into the station’s equipment out of service (EOOS) software tool as such. The combination of the ‘D’ EDG and the SLC system unavailability resulted in a Yellow on-line risk configuration in EOOS. Inspectors reviewed Attachment F of NDAP-QA-1902 which is the schedule risk assessment worksheet for operating units. Attachment F states:

- Are there any concurrent risk significant activities scheduled that will cause core damage frequency to be Yellow?
- If the answer is Yes, then concurrent risk significant activities in these areas should be rescheduled to prevent the occurrence of yellow risk.

Inspectors discovered that on February 16, the Operations Shift Manager questioned why the ‘D’ EDG was not included in the EOOS profile and reached out to the Plant Analysis group for clarification. The Plant Analysis group supervisor contacted the shift manager and informed him that the ‘D’ EDG would be unavailable during the supply fan maintenance and that the appropriate code should be entered in EOOS. The Shift Manager directed the Shift Technical Advisor to update EOOS and station risk remained Green at the time. The shift technical advisor did not run the risk profile for the next shift since he assumed the work week manager would update EOOS when he arrived in the morning. On February 17, the SLC flow surveillance was authorized without validating that EOOS correctly reflected the condition of the plant. In this case, the ‘D’ EDG was still unavailable since maintenance on the supply fan was still in progress. The combination of the ‘D’ EDG and SLC being unavailable resulted in a Yellow risk configuration.

The station performed a prompt human performance investigation and discovered that the EOOS code had been changed from the ‘D’ EDG (0G501D) to the ‘D’ EDG supply fan (0V512D) in preparation for an upcoming revision to the EOOS model. Since

0V512D was not yet in the EOOS model, when the schedule was imported into EOOS an unresolved item was generated. This item was not resolved prior to performing maintenance as directed by the procedure.

Inspectors determined that Susquehanna should have considered rescheduling the SLC flow surveillance until after the 'D' EDG was returned to service or managed the increased risk by specifying additional RMAs in accordance with station procedures. Ultimately, the inspectors determined that the risk assessment was inadequate because, if done correctly, it would it would have resulted in a higher risk category and would have required additional RMAs in accordance with station procedures. Susquehanna entered the issue into their CAP as CR-2016-04137.

Analysis. The inspectors determined that Susquehanna's inadequate assessment of the risk of performing maintenance activities, to include specifying appropriate RMAs, was a performance deficiency that was within Susquehanna's ability to foresee and correct, and should have been prevented. The inspectors determined that this performance deficiency is more than minor because it was associated with the Human Performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Additionally, the finding is similar to example 7.e. in NRC IMC 0612 Appendix E, "Examples of Minor Issues," dated August 11, 2009. This example states, in part, that failure to perform an adequate risk assessment when required by 10 CFR 50.65 (a)(4) is not minor if the overall elevated plant risk would put the plant into a higher licensee-established risk category or would require, under plant procedures, RMAs or additional RMAs. In this case, the combination of the 'D' EDG maintenance and SLC flow surveillance resulted in changing risk to Yellow, which required additional RMAs in accordance with station procedures.

The inspectors evaluated the finding using IMC 0612 Appendix K, "Maintenance Risk Assessment and Risk Management SDP," dated March 19, 2005. The inspectors and the Region I senior risk analyst used Appendix K, Flowchart 1, "Assessment of Risk Deficit," and determined that the inadequate risk assessment was of very low safety significance (Green). The basis for this determination was that the short duration of the actual planned maintenance activities (3.5 hours) associated with the SLC unavailability results in less than E-9 calculated ICDPD using Susquehanna's risk model. Since the resultant ICDPD is below 1 E-8 threshold, the finding was determined to be Green. This finding was determined to have a cross-cutting aspect in the area of Human Performance, Work Management in that Susquehanna did not appropriately incorporate insights from probabilistic risk assessments into the daily work activities. Specifically, Susquehanna did not appropriately assess the risk of performing maintenance activities as specified in station procedures [H.5].

Enforcement. 10 CFR 50.65 (a)(4) states, in part, prior to performing maintenance activities, a licensee shall assess and manage the increase in risk that may result from the proposed maintenance activity. Contrary to the above, on February 17, 2016, Susquehanna performed a SLC system flow surveillance in conjunction with having the 'D' EDG unavailable without adequately assessing and managing the increase in risk. Specifically, the risk assessment for the station did not consider the unavailability of the 'D' EDG when the SLC system flow surveillance was performed resulting in an unrecognized yellow risk condition. When the violation was identified by inspectors, both maintenance activities had been completed; therefore, no immediate corrective actions

were required.  Because this violation was of very low safety significance (Green), and Susquehanna entered this issue into their CAP as CR-2016-04137, this violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. **(NCV 05000388/2016001-01, Failure to Assess and Manage Risk of Maintenance Activities for a SLC System Flow Surveillance)**

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 2, 2 ‘A’ emergency switchgear room chiller not maintaining refrigerant pressure on January 19, 2016
- Common, oil leak on the ‘A’ control structure chiller on February 4, 2016
- Unit 1, automatic depressurization system (ADS) solenoid continuity lights on February 5, 2016
- Unit 2, incorrect grease added to the division II Swing Bus motor generator set on February 5, 2016
- Unit 1, failure of division I swing bus automatic transfer switch on March 5, 2016
- Common, water in the ‘E’ EDG diesel fuel oil day tank on March 15, 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.  The inspectors verified that Susquehanna identified OWAs at an appropriate threshold and addressed them in a manner that effectively managed OWA-related adverse effects on operators and SSCs.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18 – 1 sample)

.1 Permanent Modifications

a. Inspection Scope

The inspectors evaluated a modification to the station black-out diesel generator (Blue Max) implemented by engineering change package 1915914, “Add a Fuel Oil Heater and Thermostat to 0G503.”  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.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19 – 5 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 point were performed and checked, and that results adequately demonstrated restoration of the affected safety functions.

- Unit 1, 'B' CS following SOW on January 14, 2016
- Common, post mid-cycle operability run of the 'E' EDG on February 1, 2016
- Unit 1, division II RHR after SOW on February 10, 2016
- Unit 1, 1ATS229 transfer switch on March 8, 2016
- Unit 1, 1D620 and 1D610 battery replacement on March 17, 2016 and March 26, 2016, respectively

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 SSCs to assess whether test results satisfied TS, 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, quarterly calibration of reactor pressure vessel pressure channels (CS and low-pressure coolant injection permissive) on January 20, 2016
- Unit 1, CS pump B and D suction header relief valve (PSV152F012B) and discharge header relief valve (PSV152F032B) lift pressure setpoint and seat leakage testing on January 25, 2016 (IST)
- Unit 1, high-pressure coolant injection functional testing (SE-152-002) on March 1, 2016
- Unit 1, LOCA/LOOP testing on March 15, 2016 (division II) and March 26, 2016 (division I)
- Unit 1, 'B' and 'D' MSIV as-found local leak rate testing on March 17, 2016 (PCIV)

b. Findings

No findings were identified.

**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 February 16, 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, technical support center, 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

No findings were identified.

**4. OTHER ACTIVITIES**

4OA1 Performance Indicator Verification (71151)

.1 Unplanned Scrams, Unplanned Power Changes, and Unplanned Scrams with Complications (6 samples)

a. Inspection Scope

The inspectors reviewed Susquehanna submittals for the following Initiating Events Cornerstone performance indicators for the period of January 1, 2015, through December 31, 2015.

- Units 1 and 2, Unplanned Scrams
- Units 1 and 2, Unplanned Power Changes
- Units 1 and 2, Unplanned Scrams with Complications

To determine the accuracy of the performance indicator data reported during those periods, inspectors used definitions and guidance contained in Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7. The inspectors reviewed Susquehanna's operator narrative logs, maintenance planning schedules, condition reports, event reports, and NRC integrated inspection reports to validate the accuracy of the submittals.

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152 – 1 sample)

.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 condition report 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 Annual Sample: Compliance with Plant TSs during Instrumentation Testing

a. Inspection Scope

The inspectors performed an in-depth review of Susquehanna's evaluation and corrective actions associated with condition reports CR-2015-26455, which was written to address a loss of safety function which occurred during instrument testing and the resultant NCV documented in IR 05000387;388/2015004. Specifically, because Susquehanna's original evaluation of the issue was inadequate, as documented in the subject NCV, this in-depth review focused on revision to the evaluation, with particular emphasis on evaluation of the extent of condition.

The inspectors assessed Susquehanna's problem identification threshold, cause analyses, extent of condition reviews, compensatory actions, and the prioritization and timeliness of Susquehanna's corrective actions to determine whether Susquehanna was

appropriately identifying, characterizing, and correcting problems associated with this issue and whether the planned or completed corrective actions were appropriate. The inspectors compared the actions taken to the requirements of Susquehanna's CAP and 10 CFR 50, Appendix B, Criterion XVI.

b. Observations

On September 29, 2015, a loss of safety function of CREOASS and the standby gas treatment (SBGT) system occurred when Susquehanna performed maintenance on redundant trains concurrently. Susquehanna entered the issue into the CAP as CRs 2015-26442, 2015-26455 and 2015-26475 and determined that the losses of safety function (SBGT and CREOASS) occurred when one of the tested instruments unexpectedly failed, concurrent with a planned inoperability of the other train of SBGT and CREOASS during a separate maintenance activity. Therefore, Susquehanna identified the cause of the event was that the risk was not considered when scheduling surveillance tests because personnel were accustomed to successful outcomes and therefore had allowed surveillances from multiple divisions to be performed concurrently.

As documented in NCV 05000388/2015004-03, inspectors reviewed the apparent cause evaluation and associated corrective actions and determined that Susquehanna had not identified the appropriate cause of the event and therefore had not identified reasonable corrective actions. Susquehanna entered the inspectors' observations into the CAP as CR-2015-31802 and took action to reevaluate the cause of the loss of safety function. In follow-up to Susquehanna's actions to address the deficient evaluation, inspectors noted:

- Susquehanna's actions to address the extent of condition remained inadequate following reevaluation. Specifically, despite inspectors identifying and providing details of a separate reportable condition that occurred for the same cause, this event was not included in the revision to the evaluation that was being presented to the corrective action review board. As discussed in the below NCV, Susquehanna failed to take action to report this condition as required by 10 CFR 50.73. Following identification of this deficiency, Susquehanna made an additional revision to the evaluation and generated a separate condition report (CR-2016-02823) to review and identify any other occurrences of this condition. Inspectors note that this extent of condition review is currently not due until August 5, 2016, almost one year from original identification of the issue.

- Inspectors identified that Susquehanna had not assessed instrument surveillance correctly in accordance with the maintenance rule. Specifically, 10 CFR 50.65(a)(4) requires licensees to assess and manage the increase in risk prior to performance maintenance activities. Susquehanna uses the EOOS program, in conjunction with their probabilistic risk analysis model, to quantitatively assess the increase of risk of performing maintenance and implements RMAs based on comparison of this risk assessment to pre-established thresholds. Inspectors identified that for instrumentation surveillances that test all four channels of instrumentation, only the first channel that is tested during the procedure is included in the risk assessment. Inspectors noted this was incorrect and could provide significantly different results if the testing was performed concurrently with other work that rendered redundant trains of equipment unavailable. To assess the significance of this violation, inspectors reviewed scheduled work and risk assessments for a sampling of

instrumentation tests that had been performed over the past several years. Because inspectors did not identify any instances where the failure to adequately assess risk would have resulted in a higher risk color and required additional RMAs, inspectors determined this violation was of minor safety significance. Susquehanna entered the issue into the CAP as CR-2016-04173 and took action to ensure instrument testing was properly assessed for risk.

- Inspectors reviewed the CR screening of CR-2015-26475, which was assigned a level 2 apparent cause evaluation based on being assessed a consequence of “critical” and probability of “remote.” CR risk significance level is assigned using Attachment C of LS-120, “Issue Identification and Screening Process.” Inspectors reviewed the screening of the CR by Screening Team and Management Review Committee and determined that the risk significance level of the issue was likely underestimated due to a mischaracterization of the probability. LS-120 defines remote probability as “unlikely, but possible to occur sometime in the life of an individual item or system, or can reasonably be expected to occur in the life of a large number of similar components.” At the time of the screening of the CR in October 2015, there was only one known occurrence of this condition and Susquehanna did not recognize the cause of the issue or extent of condition. However, following identification of another loss of safety function and programmatic cause related to application of plant TSs, as documented in NCV 388/2015004-03, the issue was not appropriately represented as a probability of “remote.” A change in the probability screening, in combination with appropriately screening the CR with high uncertainty as described in Table D.1, “Uncertainty Assessment Table,” of LS-120, would likely have resulted in the CR being assigned a level 1 root cause evaluation type. Inspectors determined that this level of evaluation would have been appropriate and likely would have avoided the deficiencies in causal analysis and extent of condition that was identified by inspectors. Inspectors determined that the failure to appropriately screen and evaluate the issue was causal to NCV 388/2015004-03 as well as the NCV discussed below and did not constitute a separate finding.

c. Findings

Introduction. Inspectors identified a Severity Level IV NCV of 10 CFR Part 50.73 (a)(2)(v) when Susquehanna did not submit a LER within 60 days of identifying that both trains of the CREOASS were rendered inoperable during surveillance testing, a condition that could have prevented fulfillment of a safety function.

Description. As documented in NCV 05000387/2015004-02, Loss of Safety Function of SBGT and CREOASS due to Concurrently Performing Maintenance on Redundant Trains, inspectors identified that Susquehanna had misapplied a note in the TS surveillance requirement which allows delaying entry into the action statement if an inoperability is solely due to testing. Specifically, Susquehanna took into account that the note allowed considering the instrument operable for the delay period. By interpreting the TSs in this manner, operators allowed performance of surveillances on redundant trains of the CREOASS and SBGT system, resulting in a loss of the safety function of both systems. In addition to the loss of safety function which occurred on September 29, 2015 and was reported in LER 50-387/2015-006, inspectors identified a similar loss of safety function occurred on October 3, 2014 when surveillances which affected redundant trains of CREOASS were performed simultaneously. Inspectors

communicated this to Susquehanna management at a plant operational review committee meeting on November 24, 2015. Inspectors considered this as the time of discovery for the reportable condition because sufficient details were communicated, including work order numbers, to plant personnel who have knowledge of and responsibility for making reports under 10 CFR 50.73. Despite the fact that inspectors continued to communicate the condition to Susquehanna management, including a detailed description at the exit meeting on January 19, 2016, the condition was not reported to NRC within 60 days after the time of discovery. Susquehanna entered the issue into the CAP as CR-2016-03713 and reported the condition on May 5, 2016 in LER 50-388(387)/2015-015.

Analysis. The inspectors determined that Susquehanna's failure to report an event that could have prevented the fulfillment of a safety function was a performance deficiency that was reasonably within the Susquehanna's ability to foresee and correct and therefore, should have been prevented. Since the issue had the potential to affect the NRC's ability to perform its regulatory function, the inspectors evaluated this performance deficiency in accordance with the traditional enforcement process. Using example 6.9.d.9 from the NRC Enforcement Policy, the inspectors determined that it was a Severity Level IV violation. The significance of the associated performance deficiency was also screened against the ROP per the guidance of IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012. Because this violation involves the traditional enforcement process and does not have an associated finding under the ROP, inspectors did not assign a cross-cutting aspect to this violation.

Enforcement. 10 CFR 50.73(a)(2)(v) requires, in part, that an event or condition that could have prevented fulfillment of a safety function be reported to the NRC within 60 days from the time of discovery. Contrary to the above, Susquehanna did not submit a LER within 60 days of November 24, 2015, after discovering that both trains of CREOASS were rendered inoperable by performing surveillances on redundant trains concurrently, a condition that could have prevented the fulfillment of a safety function. After inspectors identified the missed report, Susquehanna reported the condition on May 5, 2016 in LER 50-388(387)/2015-015. Since this violation was of very low safety significance, was not repetitive or willful, and was entered into Susquehanna's CAP (CR-2016-03713), this violation is being treated as an NCV consistent with section 2.3.2 of the Enforcement Policy. (**NCV 05000387/2016001-02, Failure to Report Loss of Safety Function as Required by 10 CFR 50.73(a)(2)(v))**

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153 – 5 samples)

.1 (Closed) Licensee Event Report (LER) 05000387/2015-008-00: Unit 1 'A' Reactor Protection System Electrical Protection Assembly Breaker Underfrequency Trip Setpoint out of TS 3.3.8.2.2 Surveillance Allowable Value for Longer than Allowed by Technical Specifications

On November 10, 2015, during the completion of an evaluation for a failed surveillance test that occurred on September 22, 2015, a condition was identified that in 2014, a degraded reactor protection system (RPS) electrical protection assembly (EPA) breaker logic card had been installed in the Unit 1 RPS, resulting in an unplanned inoperability of the 'A' train of the Unit 1 RPS. On September 22, 2015, during the performance of routine surveillance testing of the Unit 1 RPS, the as-found underfrequency (UF) trip setpoint for the EPA breaker in the Unit 1 'A' train of the Alternate RPS power supply,

could not be determined because the function did not operate at as low as approximately 40 hertz (Hz). At approximately 40Hz, the undervoltage characteristic took over and initiated the trip. The TS 3.3.8.2.2 surveillance requirement UF allowable value is > 57 Hz. The UF trip setpoint test was subsequently attempted multiple times with the same result. Because the UF trip setpoint was not achieved, TS 3.3.8.2.2 was not met. The licensee concluded with firm evidence that the condition existed prior to discovery on September 22, 2015 based on the occurrence of a prior failure on September 3, 2000 with the same logic card and the same symptoms as the failure on September 22, 2015. Specifically, the condition existed from the time the logic card was installed in the Unit 1 'A' train of RPS on September 25, 2014. Because this condition existed for a period longer than allowed by TS, this condition was reported in accordance with 10 CFR 50.73(a)(2)(i)(B) as a condition prohibited by TS 3.3.8.2.2. The direct cause of the failure of the 1A Alternate RPS breaker was due to an intermittently degraded circuit logic card. The apparent cause was due to the installation of a poor quality spare electrical protection logic card in 2014. The failed logic card was subsequently replaced and the Unit 1 Alternate RPS EPA breaker was declared operable on September 22, 2015. The enforcement aspects of this finding are discussed in Section 4OA7. The inspectors did not identify any new issues during the review of the LER. This LER is closed.

2. <u>(Closed) Licensee Event Reports (LERs) 05000388/2015-001-00 and 05000388/2015-001-01</u>: Condition Prohibited by Technical Specifications Due to Drift of Reactor Pressure Steam Dome – Low Switches

On January 21, 2015 during quarterly calibration of the Reactor Steam Dome Pressure - Low switches, PIS-B21-2N021A was found outside of the TS allowable value. PIS-B21-2N021A was restored to within the allowable value. Subsequent evaluation of the switch's history showed that it had drifted outside of its TS allowable value six times during the last three years. The historical search also identified other related Unit 1 and Unit 2 switches had drifted outside of the allowable value. In addition, on July 26, 2013, two Unit 2 channels (A and B) were found outside of the TS allowable values. With both the A and B channels inoperable at the same time, the safety functions of both CS and Low Pressure Coolant Injection (LPCI) were impacted; however, CS and LPCI remained capable of performing their design function as analyzed in the FSAR accident analysis.

Based on the historical information, Susquehanna considered firm evidence to exist to indicate that the condition existed prior to the time of discovery. Based on this conclusion, the condition was considered to be a condition prohibited by TSs (10 CFR 50.72(a)(2)(i)(B)), a common cause inoperability of independent trains or channels (10 CFR 50.73(a)(2)(vii)), a condition that could have prevented fulfillment of a safety function (10 CFR 50.73(a)(2)(v)(D)), and a single cause that could have prevented fulfillment of safety functions of trains or channels in different systems (10 CFR 50.73(a)(2)(ix)(A)).

The apparent cause was determined to be less than adequate design in that the design failed to consider the effects of mechanical hysteresis of the subject devices when operated above their normal operating range and did not recognize the seasonal temperature effects to the set points. Key planned corrective actions include replacing the affected switches with switches that are better suited for the application. As an interim action, the calibration frequency for the switches has been increased from

quarterly to every 45 days. The enforcement aspects of this finding are discussed in Section 4OA7. The inspectors did not identify any new issues during the review of the LER. These LERs are closed.

3. (Closed) Licensee Event Report (LER) 05000387/2015-007-00: Unit 1 'B' Inboard Main Steam Isolation Valve, HV141F022B Closed During Surveillance Test Which Caused a SCRAM on Unit 1

On November 12, 2015 at 1132 hours, the Unit 1 'B' Inboard Main Steam Isolation Valve (MSIV), HV141F022B, closed during the performance of SI-183-207, Quarterly Functional Test of Main Steam Line 'C' Flow Channels FIS-B21-1 N008A&B and Main Steam Line 'D' Flow Channels FIS-B21-1N009A&B. This resulted in an automatic scram of Unit 1 on high reactor pressure.

This event was reported under 10 CFR 50.72(b)(2)(iv)(B) and 10CFR 50.72(b)(3)(iv)(A) per the guidance of NUREG 1022, Revision 3, Section 3.2.6.

Susquehanna determined that station personnel did not evaluate recommendations made in 2011 by the Boiling Water Reactor Owners Group Instrument and Controls (BWROG I&C) Maintenance Committee to mitigate Primary Containment Isolation System (PCIS) Group 1 Surveillance Testing Risk. The licensee also determined that a lack of specific guidance in SI-183-306 (24 Month Calibration Main Steam C/D) led to an inaccurate field decision when determining the cause of an extinguished pilot valve continuity light.

Planned corrective actions include evaluating the BWROG I&C Maintenance Committee Recommendation to mitigate PCIS Group 1 Surveillance Testing Risk and then designing and implementing the proposed modification. Additionally, I&C personnel are performing a review of other BWROG recommendations that were issued to ensure they were evaluated.

The inspectors reviewed this LER, Susquehanna's evaluation, and associated corrective actions. The enforcement aspects associated with this event are documented in IR 05000387;388/2015004. The inspectors did not identify any new issues during the review of the LER. This LER is closed.

4. (Closed) Licensee Event Report (LER) 05000387/2015-009: “Pressure Boundary Leakage from an Inadequate Weld Repair in Small Bore Pump Seal Vent Piping” on the Unit 1 ‘B’ Reactor Recirculation Pump Lower Seal Cavity Vent Piping

a. Inspection Scope

On November 13, 2015, during a drywell entry following an unplanned Unit 1 scram due to the ‘B’ inboard MSIV closure (LER 05000387/2015-007), Susquehanna identified a leak on the Unit 1 ‘B’ RRP lower seal cavity vent piping. Susquehanna determined the location of the leak was part of the reactor coolant pressure boundary. Susquehanna determined that this leakage constituted a violation of the Unit 1 TS LCO 3.4.4,"RCS," which requires RCS leakage to be limited to no pressure boundary leakage. The affected piping had been in service for approximately 11 months following a previous repair of the weld at this location in December 2014 (ref. LER 05000387/2014-011).

The pressure boundary leakage was reported in accordance with 10 CFR 50.72(b)(3)(ii)(A). Susquehanna performed a root cause analysis of the November 2015 pressure boundary leakage event (CR-2015-30901) and determined the previous weld repair from in December 2014 did not fully excavate the deficient weld so as to eliminate the presence of a crack. The LER and associated evaluation was reviewed for accuracy, the appropriateness of corrective actions, violations of requirements, and potential generic issues. This LER is closed.

b. Findings

Introduction. A self-revealing finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," was identified for Susquehanna's failure to establish measures to assure that a condition adverse to quality was adequately corrected. Specifically, vibration induced fatigue cracking on the Unit 1 'B' RRP lower seal cavity vent piping was not corrected in December 2014 after a reactor coolant pressure boundary leak had occurred. This resulted in another reactor coolant pressure boundary leak at the same location with Unit 1 operating in Mode 1, a condition prohibited by TS LCO 3.4.4.

Description. On November 13, 2015, Susquehanna identified a leak on the Unit 1 'B' RRP lower seal cavity vent piping. Since Unit 1 was in Mode 1 when the elevated drywell leakage occurred, Susquehanna determined that this constituted a violation of the Unit 1 TS LCO 3.4.4, "RCS" which requires, in part, RCS leakage to be limited to no pressure boundary leakage. Susquehanna entered the leak into the CAP as CR-2015-30901 and performed an evaluation to identify the cause of the leakage and specify corrective actions. Susquehanna determined that a leak at the same location was identified on December 13, 2014 (LER 05000387/2015-011). That leak was repaired on December 15, 2014 (CR-2014-37848) by excavating the through-wall portion of the flaw and reworking that portion of the weld. Susquehanna performed a start-up of Unit 1 on December 17, 2014 following the repair and entered Mode 1 on December 18, 2014. On December 28, 2014, Unit 1 drywell leakage took a step increase from 0.1 gpm to 0.2 gpm (CR-2014-38904) and gradually increased to approximately 0.6 gpm for the next three months where it stabilized until the shutdown in November 2015.

Susquehanna attributed the cause of the leak in November 2015 to the decision to not fully excavate the weld when the repair was made in December 2014. Specifically, Susquehanna noted that weld engineers, who believed there was a possibility that a subsurface defect could be present after excavation, were not consulted on the repair plan. Additionally, NDAP-QA-1208, "Control of Welding," provides guidance to ensure that a flaw/defect be fully characterized prior to any metal removal to ensure the extent of the flaw/defect is understood and requires socket welds which contain defects to be removed prior to rework. Inspectors determined that since a volumetric examination could not be performed based on the location of the flaw, the socket should have been removed and reworked. Ultimately, inspectors determined that Susquehanna's corrective actions were inadequate to address the fatigue cracking identified in December 2014 which resulted in pressure boundary leakage recurring in November 2015. Susquehanna corrected the condition adverse to quality by replacing and modifying the union that included the weld.

Analysis. Failure to establish measures to assure a condition adverse to quality was corrected in accordance with 10 CFR Part 50, Appendix B, Criterion XVI, 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 Initiating Events cornerstone and affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors evaluated the finding in accordance with Exhibit 1 of IMC 0609, Appendix A, "The SDP for Findings At-Power," dated June 19, 2012 and determined the finding was of very low safety significance (Green) because the leakage would not have exceeded the RCS leak rate for a small LOCA and it did not affect other systems used to mitigate a LOCA.

This finding had a cross-cutting aspect in the area of Human Performance, Work Management, because Susquehanna did not implement a process of planning, controlling, and executing work activities such that nuclear safety is the overriding priority, in that Susquehanna did not adequately coordinate the work activities with different groups (H.5). Specifically, welding engineers were not engaged in the decision making process during the December 2014 repair.

Enforcement. 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, 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 requirement, despite identifying a condition adverse to quality in December 2014 associated with vibration-induces fatigue cracking on the Unit 1 'B' RRP lower seal cavity vent piping, implementation of the CAP did not assure that the condition adverse to quality was corrected prior to return to service. This resulted in RCS pressure boundary leakage that was identified on November 13, 2015.

Additionally, TS 3.4.4, "RCS," prohibits pressure boundary leakage in Mode 1. Contrary to this, Susquehanna operated in Mode 1 from December 2014 through November 2015 with reactor coolant pressure boundary leakage. Susquehanna's immediate corrective actions included correcting the condition adverse to quality by replacing and modifying the union that included the weld. Since the violation is of very low safety significance and has been entered into Susquehanna's CAP as CR-2015-30901, this violation is being treated as an NCV, consistent with section 2.3.2.a of the Enforcement Policy. **(NCV05000387/2016001-03, Failure to Correct Fatigue Related Cracking of the 'B' RRP Lower Seal Cavity Vent Line)**

4OA6 Meetings, Including Exit

On April 22, 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.

4OA7 Licensee-Identified Violations

The following violations of very low safety significance (Green) were identified by Susquehanna and are violations of NRC requirements which meets the criteria of the NRC Enforcement Policy for being dispositioned as NCVs.

- 10 CFR Part 50 Appendix B, Criterion XVI Corrective Action, requires, in part, that measures be established to assure conditions adverse to quality are promptly identified and corrected. Susquehanna Unit 1 TS 3.3.8.2 requires operability of two RPS electrical power monitoring assemblies for each inservice RPS alternative power supply. Contrary to this, a defective electrical power monitoring assembly was installed into the RPS alternative power supply in December 2014. Susquehanna identified that in September 2000, an electrical power monitoring assembly had failed to perform an UF trip. An evaluation of the failure was conducted and Susquehanna incorrectly concluded in 2001 that the electrical power monitoring assembly did not contribute to the failure and returned the electrical power monitoring assembly back to a spare parts status. In December 2014 the subject electrical power monitoring assembly was installed into the alternative RPS power supply and in September 2015 again failed to perform an UF trip. As reported in LER 387/2015-008 this resulted in an inoperable electrical power monitoring assembly in the inservice alternative RPS power supply for greater than the allowed outage time. This violation has been entered into Susquehanna's CAP as CR-2015-25881. The inspectors evaluated this finding using IMC 0609.04, "Initial Characterization of Findings," and IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." Because it was associated with the RPS, the inspectors screened the violation against the questions in IMC 0609, Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," Part C, "Reactivity Control Systems." The inspectors determined the finding was of very low safety significance (Green) because the second RPS electrical power monitoring assembly remained operable which would have generated an UF trip for the RPS alternative power supply if required, did not involve unintentional reactivity addition, and did not involve a mismanagement of reactivity by operators.

- 10 CFR Part 50 Appendix B, Criterion III Design Control, requires, in part, that measures be established for the selection and review for suitability of applicable equipment that is essential to the safety-related functions of systems. Susquehanna Unit 2 TS 3.3.5.1 requires operability of four reactor steam dome pressure – low channels to provide the injection permissive for CS (Function 1d) and LPCI system (Function 2d). Contrary to this requirement, Susquehanna did not consider the effects of mechanical hysteresis on the function of reactor steam dome pressure switches when operated above their normal operating range. These reactor steam dome pressure switches remain in an overpressure condition during normal operations and as reported in LER 388/2015-001 resulted in pressure switches drifting out of TS allowable values for emergency core cooling system (ECCS) injection valve permissive interlocks for greater than the TS allowed outage time. This violation has been entered into Susquehanna's CAP as CR-2015-06243.

In consultation with regional Senior Reactor Analysts, the inspectors determined the finding was of very low safety significance (Green) because the ability to open low pressure ECCS injection valves manually remained available and engineering analysis for the as-found condition of the switches determined that the resultant delay in automatic response would have a negligible increase in peak central temperature during a design basis accident.

ATTACHMENT: SUPPLEMENTARY INFORMATION

# SUPPLEMENTARY INFORMATION

## KEY POINTS OF CONTACT

Licensee Personnel

J. Franke, Site Vice President
B. Franssen, Plant Manager
M. Comstock, Welding Engineer
T. Creasy, System Engineering Manager
M. Dziedzic, Talen Level III
J. Hertzell, Manager- Plant Analysis
T. Jardine, Online Manager
J. Jennings, Manager- Nuclear Regulatory Affairs
P. Jones, Operations Manager
T. Kupetz, ISI Program Engineer
C. Manges, Regulatory Assurance
B. Payne, Work Week Manager
A. Schvad, Engineer
J. Setzer, GE Level III
R. Whitenight, FAC Engineer

## LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

Opened/Closed

| | | |
|---|---|---|
| 05000388/2016001-01 | NCV | Failure to Assess and Manage Risk of Maintenance Activities for a SLC System Flow Surveillance (Section 1R13) |
| 05000387/2016001-02 | NCV | Failure to Report Loss of Safety Function as Required by 10 CFR 50.73(a)(2)(v) (Section 4OA2) |
| 05000387/2016001-03 | NCV | Failure to Correct Fatigue Related Cracking of the ‘B’ RRP Lower Seal Cavity Vent Line (Section 4OA3) |

Closed

| | | |
|---|---|---|
| 05000387/2015-008-00 | LER | Unit 1 ‘A’ Reactor Protection System Electrical Protection Assembly Breaker Underfrequency Trip Setpoint out of TS 3.3.8.2.2 Surveillance Allowable Value for Longer than Allowed by Technical Specifications (Section 4OA3) |

| | | |
|---|---|---|
| 05000388/2015-001-00/01 | LER | Condition Prohibited by Technical Specifications Due to Drift of Reactor Pressure Steam Dome – Low Switches (Section 4OA3) |
| 05000387/2015-007-00 | LER | Unit 1 'B' Inboard Main Steam Isolation Valve, HV141F022B Closed During Surveillance Test Which Caused a SCRAM on Unit 1 (Section 4OA3) |
| 05000387/2015-009 | LER | "Pressure Boundary Leakage from an Inadequate Weld Repair in Small Bore Pump Seal Vent Piping" on the Unit 1 'B' Reactor Recirculation Pump Lower Seal Cavity Vent Piping (Section 4OA3) |

## LIST OF DOCUMENTS REVIEWED

### Section 1R04: Equipment Alignment

Procedures
OP-024-001, Diesel Generators, Revision 79

Condition Reports
CR-2016-03282

Drawings
M-152, Unit 1 P&ID CS, Sheet 1, Revision 40
M-151, Unit 1 P&ID RHR, Sheet 1, Revision 71
M-151, Unit 1 P&ID RHR, Sheet 2, Revision 53
M-151, Unit 1 P&ID RHR, Sheet 3, Revision 30
M-151, Unit 1 P&ID RHR, Sheet 4, Revision 20
M-151, Unit 1 P&ID RHR, Sheet 5, Revision 3
M-143, Unit 1 P&ID Reactor Recirculation, Sheet 1, Revision 49
M-151, Unit 1 P&ID RHR, Sheet 1, Revision 71
M-151, Unit 1 P&ID RHR, Sheet 2, Revision 53
M-151, Unit 1 P&ID RHR, Sheet 3, Revision 30

### Section 1R05: F ire Protection

Procedures
FP-113-125, Access Area (1-604) Adjoining Rooms (I-621, 620, 619, 606, 601) Fire Zones 1-6A, 1-61, 0-6G Elevation 779'-1", Revision 5
FP-113-123, Load Center Room (I-507) Load Center Room (I-510) Fire Zone 1-5F, 1-5G Elevation 749-1", Revision 4
FP-113-119, Circulation Space (I-500) and Adjacent Rooms (I-511, 517, 514, 508, 513) Fire Zones 1-5A-N, S, W; 1-5H Elevation 749'-1", Revision 6
FP-113-212, Condenser Area (1-36) RFPT Exhaust Areas (I-37), (I-38), (I-39) Fire Zone 1-31D Elevation 656', Revision 4

FP-113-225, Condenser Mezzanine (I-211) Fire Zone 1-3C Elevation 699’-0”, Revision 1
FP-113-291, Condenser Gallery (I-113) Fire Zone 1-32D Elevation 676’-0”, Revision 2
FP-113-100, Drywell (I-400, I-516, I-607) Fire Zone 1-4F Elevation 704’ through 807’, Revision 3

Condition Reports (*NRC identified)

| | | |
|---|---|---|
| CR-2016-04591* | CR-2016-04594* | CR-2016-06346* |

Drawings
C-1725, Unit 1 Reactor Building Fire Zone Plan Elevation 779’-1”, Sheet 1, Revision 9
C-1724, Unit 1 Reactor Building Fire Zone Plan Elevation 749’-1”, Sheet 1, Revision 11

**Section 1R07: Heat Sink Performance**

Procedures
MT-GM-025, Hear Exchanger- Cleaning and Inspection, Revision 22
MT-116-002, RHR Heat Exchanger Cleaning, Inspection and Repair, Revision 15

Maintenance Orders/Work Orders
1919140

Miscellaneous
Attachment 71111.07, Heat Sink Performance, 7/6/10

**Section 1R08: In-service Inspection**

Procedures
NDAP-QA-1208, Control of Welding, Revision 13
NDE-PT-001, Color Contrast Liquid Penetrant Examination, Revision 5
NDE-UT-027, Procedure for Manual Ultrasonic Examination of Nozzle Inner Radius, Bore, and Selected Nozzle Regions, Revision 8
NDE-UT-042, Procedure for Manual Examination of Reactor Vessel Assembly Welds in Accordance with PDI, Revision 5
NDE-UT-076, Automated Phased Array Ultrasonic Examination of Weld Overlaid Dissimilar Metal and Austenitic Welds, Revision 0
NDE-VT-005, Underwater Visual Examination of RPV Internals, Revision 10

Condition Reports (*NRC identified)

| | | | |
|---|---|---|---|
| CR-2016-06839 | CR-2016-07158 | CR-2016-07408 | CR-2016-07541 |
| CR-2016-07695 | CR-2016-07816* | CR-2016-07819* | |

Maintenance Orders/Work Orders
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Drawings
A296242, Sheet 5, Unit 1 Containment Liner Surface, Revision 0
SPDCA110-6, Sheet 1, Unit 1 Isometric Reactor Building RHR Crosstie Line, Revision 0

Miscellaneous
1-B3.100.0003, UT Data Report for N1B inner radius, dated March 25, 2016
1-B3.90.0002, UT Data Report for N1B nozzle to shell, dated March 25, 2016
1-R1.16.0010, UT Data Report for N2J nozzle to safe end overlay, dated March 26, 2008
1-R1.16.0010, UT Data Report for N2J nozzle to safe end overlay, dated March 30, 2016
624701, UT Data Report for N2J nozzle to safe end overlay, dated April 28, 2004
BOP-PT-16-194, PT Data Report for SPDCA110-6 FW-10 and FW-14, dated March 23, 2016
BOP-PT-16-199, PT Data Report for SPDCA110-6 FW-10 and FW-14, dated March 23, 2016
Indication Notification Report (INR) IVVI-16-05, Jet Pump 01-10 Set Screws, dated March 19, 2016
INR IVVI-16-06, Jet Pump 01-10 Slip Joint Clamp, dated March 19, 2016
INR IVVI-16-18, Jet Pump 11-20 Main Wedges, dated March 23, 2016
INR IVVI-16-19, Jet Pump 11-20 Aux Wedges, dated March 24, 2016
INR IVVI-16-20, Jet Pump 11-20 Set Screws, dated March 24, 2016
INR IVVI-16-21, Jet Pump 11-20 Slip Joint Clamp, dated March 24, 2016
Material Issue Record 150509-003, dated March 23, 2016
Owner's Activity Report for Unit 1 18th Refueling Outage, dated August 28, 2014
Owner's Activity Report for Unit 2 17th Refueling Outage, dated August 11, 2015
Susquehanna Fourth 10-Year Interval ISI Program Plan, Revision 1
VT-16-052 thru -057, VT Data Report for containment liner and penetrations, dated March 24, 2016
VT-16-059, VT Data Report for suppression chamber liner plate and penetrations interior, dated March 24, 2016
Welding Procedure Specification WSI WPS 08-08-TS-001, Revision 01
Weld Record 150509 for SPDCA110-6 FW-10 and FW-14, dated March 23, 2016

**Section 1R11: Licensed Operator Requalification Program**

Procedures
EO-000-103, Primary Containment Control, Revision 16
EO-000-102, RPV Control, Revision 14
ON-LOOP-101, Unit 1 Loss of All Offsite Power, Revision 1
GO-100-004, Plant Shutdown to Minimum Power, Revision 77
GO-100-005, Plant Shutdown to Hot/Cold Shutdown, Revision 68

Condition Reports (*NRC identified)
CR-2016-06255

**Section 1R12: Maintenance Effectiveness**

Procedures
SO-173-014, Containment Atmosphere Control Cold Shutdown Valve Exercising, Revision 13
SO-169-001, Quarterly Liquid Radwaste Valve Exercising, Revision 18
SO-034-001, Quarterly Zone III Isolation Damper Testing
SO-134-014, Two Year Reactor Building Chilled Water Valve Exercising, Revision 14
SO-269-001, Quarterly Liquid Radwaste Valve Exercising, Revision 18
SO-234-014, Two Year Reactor Building Chilled Water Valve Exercising, Revision 11

Condition Reports (*NRC identified)
CR-2016-02583 CR-2016-02590 CR-2016-02593

**Section 1R13: Maintenance Risk Assessments and Emergent Work Control**

Procedures
NDAP-QA-1900, Conduct of Work Management, Revision 10
NDAP-QA-1901, Susquehanna Station Work Management Process, Revision 21
NDAP-QA-1902, Integrated Rick management, Revision 22
NDAP-QA-0412, Leakage Rate Test Program, Revision 19
OI-013-002, Fire Risk Management, Revision 8
OI-013-002, Fire Risk Management, Revision 9
PSP-26, Online and Shutdown Nuclear Risk Assessment Program, Revision 15

Condition Reports (*NRC identified)

| | | | |
|---|---|---|---|
| CR-2016-00549 | CR-2016-00917 | CR-2016-00991 | CR-2016-00992 |
| CR-2016-01016 | CR-2016-01020 | CR-2016-01027 | CR-2016-01034 |
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Maintenance Orders/Work Orders
1929681

Drawings
E-26, Unit 2 Schematic Meter and Relay Diagram 125V DC System, Sheet 1B, Revision 10
E-11, Unit 2 Single Line Meter and Relay Diagram 125 & 250 VDC System, Sheet 2, Revision 28
M-182, Unit 1, Unit 2 & Common P&ID DG & ESSW Pump House Air Flow Diagram, Sheet 1, Revision 9

Miscellaneous
ANSI/IEEE Std 308-1980, IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
00-001, Div 2 Fire RMA/ATT C, Protected Equipment Clearance Order, February 8, 2016
49-001, Unit 1 Div 2 RHR, Protected Equipment Clearance, February 8, 2016
06-001, Unit 1 Div 2 Swing Bus, Protected Equipment Clearance, February 8, 2016
54-001, ESW Div 2, Protected Equipment Clearance, February 8, 2016
16-001, RHRSW Div 2, Protected Equipment Clearance, February 8, 2016
Unit 1 Risk Profile, February 5, 2016
Clearance: 52-001-HPCI, Protected Equipment
Clearance: 18-001D-INST Air Comp B, Protected Equipment
Unit 1 and 2 Risk Profile for Week of January 24, 2016
Unit 2 Control Room Logs, February 16, 2016
EC-028-0505, DG Building Heating and Ventilating System, Revision 0

**Section 1R15: Operability Determinations and Functionality Assessments**

Procedures
SO-234-005A, Quarterly Emergency Switchgear and Load Center Rooms Cooling Units Valve Exercising Loop A, Revision 6
AR-110-001, ADS and DRWL CLG 1C601, Revision 18
NDAP-QA-1201, Configuration Management Process and Program, Revision 12
NDAP-QA-1220, Engineering Change Process, Revision 10
TP-024-071, Factory Test of D/G ‘E’, Revision 1

Condition Reports (*NRC identified)

| | | | |
|---|---|---|---|
| CR-2016-01449 | CR-2016-02268 | CR-2016-03128 | CR-2016-03190 |
| CR-2016-03583 | CR-2016-03790 | CR-2016-04210* | CR-2016-05589 |
| CR-2016-06540 | | | |

Action Requests

AR-2016-05442    AR-2016-06584    AR-877268

Maintenance Orders/Work Orders

1937323    1964924    1978410

Drawings

E-324, Reactor Core Cooling System Benchboard-IC601 Unit 1, Sheet 12, Revision 13
M-141, SES Unit 1 P&ID Nuclear Boiler, Sheet 1, Revision 54
E-180, Block Diagram SRV Flow Monitoring System Unit 1, Sheet 7, Revision 9
M-134, SES Common P&ID 'E' Diesel Auxiliaries (Fuel Oil System, Lube Oil System and Air Intake &Exhaust System), Sheet 7, Revision 22
EC-1981690

Miscellaneous

IOM 662, Installation, Operations and Maintenance Instructions for Refrigeration System for Unit 2 Emergency Switchgear Room Cooling
J93-2102, Increase Set Point for PSH-27202A/B to Restore Design Basis Capacity to Unit 2 Eskige Cooling Units (Dx Units), Revision 0
SSES Significant Operating Occurrence Report 94113
Clearance: 06-001 -1ATS229 Troubleshoot
EC-Risk-1110, PSA-004.16-RHR PRA System Notebook, Revision 2
EC-Risk-1101, Susquehanna PRA Model: Containment Structural Analysis Notebook, Revision 2
ASTM D6185-11, Standard Practice for Evaluating Compatibility of binary Mixtures of Lubricating Greases
IOM 1139, KSV Operating and Maintenance Manual, Revision 6
Diesel Generators and Auxiliaries DBD013, Design Basis Document

**Section 1R18: Plant Modifications**

Procedures

NDAP-QA-0029, Procedure and Work Instruction Use and Adherence, Revision 26

Condition Reports (*NRC identified)

CR-2016-04721*    CR-2016-04877*    CR-2016-05303*

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1885732    1923699    1923700

Miscellaneous

EC-1915914

**Section 1R19: Post-Maintenance Testing**

Procedures
PSP-29, Post Maintenance Testing Matrix, Revision 20
NDAP-QA-0423, Station Pump and Valve Testing Program, Revision 33
SO-151-B02, Quarterly CS Flow Verification Division II, Revision 24
SO-151-15B, 24 Month CS Division II System Remote Position Indicator (RPI) Checks, Revision 0
SO-151-B04, Quarterly CS Valve Exercising Division II, Revision 13
SI-151-202B, Quarterly Functional Test of RCS Leakage High Pressure Monitor Channels PSH-E21-1N007B CS System Injection Valve, Revision 0
SM-151-002, 24 Month CS Pumps 1B 1P206B and 1D 1P206D Offsite Power Timer Relay Testing, Revision 10
SO-106-A01, Monthly ESS Division 1 Swing Bus, Revision 7
SO-149-A05, Quarterly RHR Loop A Valve Exercising, Revision 15
SO-149-A02, Quarterly RHR System Flow Verification Division 1, Revision 25
SM-102-B03, 24 Month Channel B 1D620 125VDC Battery Service Discharge Test and 1D623 Battery Charger Capability Test, Revision 26
SM-102-A03, 24 Month Channel A 1D610 125VDC Battery Service Discharge Test and 1D613 Battery Charger Capability Test, Revision 25
SO-106-B01, Monthly ESS Division 2 Swing Bus, Revision 8

Condition Reports (*NRC identified)

| | | | |
|---|---|---|---|
| CR-2016-02600 | CR-2016-02602 | CR-2016-02616 | CR-2016-03076 |
| CR-2016-03082 | CR-2016-03384 | CR-2016-03407 | CR-2016-05589 |
| CR-2016-06070 | | | |

Maintenance Orders/Work Orders

| | | | | | |
|---|---|---|---|---|---|
| 1728643 | 1779143 | 1802973 | 1805701 | 1805959 | 1808854 |
| 1835240 | 1939264 | 1939265 | 1945812 | 1945814 | 1954882 |
| 1959669 | 1967833 | 1975910 | | | |

Miscellaneous
M-1269, Motor Operated Valve Data Detail Drawing for: HV152F015B, Sheet 1. Revision 7

**Section 1R22: Surveillance Testing**

Procedures
SI-180-301, Quarterly Calibration of Reactor Vessel Pressure Channels PIS-B21-1N021A,B,C,D and PS-B21-1N021E,G (CS System and LPCI Permissive) Reactor Pressure Greater than Setting (420 PSIG), Revision 29
MT-GM-005, Safety/Relief Valve Setting, Revision 30
SE-152-002, Unit 1 HPCI Logic System Functional Test (On-line Version), Revision 6
NDAP-QA-0412, Leakage Rate Test Program, Revision 19
SE-159-022, LLRT of MSIVs Penetration Number X-7B, Revision 17
SE-124-107, Unit 1 Division I Diesel Generator LOCA/LOOP (Special, Test Infrequent or Complex Test/Evolution), Revision 28

Condition Reports (*NRC identified)

| | | | |
|---|---|---|---|
| CR-1549033 | CR-1550610 | CR-2015-01224 | CR-2015-01518 |
| CR-2015-17988 | CR-2015-28468 | CR-2016-01003 | CR-2016-01253 |
| CR-2016-01706 | CR-2016-01728 | CR-2016-01738 | CR-2016-05157 |
| CR-2016-06225 | CR-2016-06239 | CR-2016-07026 | CR-2016-07659 |
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Action Requests

1091220 1602188

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|---|---|---|---|---|---|
| 1467901 | 1556117 | 1571573 | 1571574 | 1810470 | 1810596 |
| 1928010 | 1928059 | 1953733 | 482620 | | |

Drawings

M334-35, SES Common Wiring Diagram Local Control Panel 0C877A, Sheet 1, Revision 18
E-105, SES Unit 1 Schematic Diagram 4.16K BUS 1A D/G Gen CKT BKR Control, Sheet 1, Revision 26
E-214, SES Common Schematic Diagram HVAC Control STRC Chilled Water System LOOP Reset, Sheet 9A, Revision 6
E-207, SES Unit 1 Schematic Diagram HVAC ESSW Pump House Vent Sys RHR SW PP Supply Fans, Sheet 3, Revision 20
M334-35, SES Common Wiring Diagram Local Control Panel 0C877A, Sheet 2, Revision 13
E-192, SES Unit 1 Schematic Diagram HVAC Reactor BLDG Vent System Emer. SWGR. RM. Unit Cooler 1V222A, Sheet 45, Revision 7

Miscellaneous

Unit 1 Control Room Logs, January 20, 2016, 07:55:00
IOM 29, LCT Series Relief Valves Manual, Revision 7
EC-051-0003, CS System Relief Valve Setpoints, Revision 1
NRC Information Notice 2015-13: MSIV Events
BWROG, MSIV Testing Position Paper: Appendix J Compliance, Document Number 04-A, June 2005
EPRI NP-2381, Measurements and Comparisons of Generic BWR MSIV, July 1982
ANSI/ANS- 56.8, Containment System Leakage Testing Requirements, 1994
DBD016, Main Steam System and Automatic Depressurization System, Revision 4
Unit 1 Control Room Logs dated March 26, 2016
Unit 1 Control Room Logs dated March 23, 2016
Clearance: 34-001 -1980204-0, 1V222A Load Sequence Timer Failed During LOCA LOOP Testing
EC-024-1029, Evaluation of EDG Frequency Response during Design Basis LOOP/LOCA Transient Loading with Replacement Woodward 2301A Governor, Revision 0

**Section 1EP6: Drill Evaluation**

Procedures

EP-PS-245, EOF Dose Assessment Supervisor, Revision 19
EP-PS-001, Emergency Planning Forms and Supplementary Instructions, Revision 9
EP-RM-005, SSES MIDAS-NU User Manual, Revision 3
EP-RM-004, EAL Classification Bases, Revision 7

Condition Reports (*NRC identified)

| | | | |
|---|---|---|---|
| CR-2016-03985 | CR-2016-03987 | CR-2016-03993 | CR-2016-04000 |
| CR-2016-04011 | CR-2016-04012 | CR-2016-04016 | CR-2016-04023 |
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| CR-2016-04341 | CR-2016-04483 | | |

Miscellaneous
2016 Drill Report, Emergency Preparedness Drill Blue Team, February 16, 2016

**Section 4OA1: Performance Indicator Verification**

Condition Reports (*NRC identified)
CR-2015-31534

Miscellaneous
DI-2015-11928

**Section 4OA2: Problem Identification and Resolution**

Procedures
SI-158-302, 24 Month Calibration of Scram Discharge Volume (SDV) High Water Level Channels LSH-C12-1N013 A, B, C, D, Revision 13
SI-180-305, Quarterly Calibration of Reactor Vessel Water Level Channels LIS-B21-1N024 A, B, C, D, Revision 35

Condition Reports (*NRC identified)

| | | | |
|---|---|---|---|
| CR-2015-26455 | CR-2016-02823* | CR-2016-03713* | CR-2016-03909* |
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Maintenance Orders/Work Orders
1594025

Miscellaneous
NUMARC 93-01, Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Revision 4A
Maintenance Rule Basis Document- System 58, Revision 5

**Section 4OA3: Follow-up of Events and Notices of Enforcement Discretion**

Procedures
SM-158-002, RPS ‘A’ Alternate EPA 24 Month Channel Calibration and Functional Test, Revision 15
SI-280-301, Quarterly Calibration of Reactor Vessel Pressure Channels PIS-B21-2N021 A, B, C, D, and PS-B21-2N021E,G (CS System and LPCI Permissive) Reactor Pressure Greater than Setting (420 PSIG), Revision 27

Condition Reports (*NRC identified)

| | | | |
|---|---|---|---|
| CR-2015-06243 | CR-2015-25881 | CR-2015-30721 | CR-333200 |
| CR-97-2013 | | | |

Maintenance Orders/Work Orders

1870358 278591 301379 333379

Drawings

M1-E21-35, Elem Diagram CS System, Sheet 3, Revision 23

Miscellaneous

EC-080-1007, Setpoint B21, E21: Core Spray RHR/LPCI Reactor Low Pressure Permissive Pressure Indicating Switches PISB212N021A, PISB212N021B, PISB212N021C, and PISB212N021D Setpoint Calculation, Revision 1

EC-049-1020, CS/LPCI Reactor Vessel Pressure Permissive Setpoint Evaluation, Revision 4

50.59 Safety Evaluation, U1/U2 CS/RHR/LPCI Reactor Low Pressure Permissive Pressure Switch Replacement, 97-9075/6

EC-080-1011, Allowable Value, TRM and Process Setpoint of PIS-B21-2N021B,D; PS-B21-2N021E,G Switch Contact #1, RPV Steam Dome Low Pressure Permissive for Recirc Discharge Valve Closure, Revision 1

## LIST OF ACRONYMS

| | |
|---|---|
| ADAMS | Agencywide Documents Access and Management System |
| ASME | American Society of Mechanical Engineers |
| BWROG | Boiling Water Reactor Owners Group |
| CAP | corrective action program |
| CFR | Code of Federal Regulations |
| CREOASS | control room emergency outside air supply system |
| CS | core spray |
| ECCS | emergency core cooling system |
| EDG | emergency diesel generator |
| EOOS | equipment out of service |
| EPA | electrical protection assembly |
| Hz | hertz |
| I&C | instrument and controls |
| ICDPD | incremental core damage probability deficit |
| IMC | Inspection Manual Chapter |
| ISI | in-service inspection |
| KV | kilovolt |
| LER | licensee event report |
| LOCA | loss of coolant accident |
| LOOP | loss of offsite power |
| LPCI | low pressure coolant injection |
| LPT | liquid penetrant testing |
| MSIV | main steam isolation valve |
| NCV | non-cited violation |
| NDE | non-destructive examination |
| NRC | Nuclear Regulatory Commission |
| OWA | operator workaround |
| PCIS | primary containment isolation system |
| RCS | reactor coolant system |
| RHR | residual heat removal |
| RMA | risk management action |
| ROP | reactor oversight process |
| RPS | reactor protection system |
| RRP | reactor recirculation pump |
| RTP | rated thermal power |
| SBGT | standby gas treatment |
| SDP | significance determination process |
| SLC | standby liquid control |
| SOW | system outage window |
| SSC | structure, system, and component |
| SSES | Susquehanna Steam Electric Station |
| TS | technical specification |
| UF | Underfrequency |
| UFSAR | Updated Final Safety Analysis Report |
| UT | ultrasonic testing |
| VT | visual testing |