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

November 5, 2015

EA-15-161
EA-15-149
EA-15-189

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 05000387/2015003 AND 05000388/2015003 AND NOTICE OF ENFORCEMENT DISCRETION

Dear Mr. Rausch:

On September 30, 2015, 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 that were discussed on October 13, 2015, with Mr. Jon Franke, 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 three findings of very low safety significance (Green) in this report, all of that involved violations of NRC requirements. Further, inspectors documented a licensee-identified violation that was 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.

Separately, a violation involving a failure to maintain secondary containment during operations with the potential to drain the reactor vessel (OPDRVs) was identified during the Unit 2 refueling outage. Specifically, from April 14, 2015 to May 16, 2015, while all other Technical Specifications (TSs) were met, Susquehanna conducted several OPDRVs without establishing secondary containment operability, a condition that was a violation of TS 3.6.4.1, "Secondary Containment." NRC issued Enforcement Guidance Memorandum (EGM) 11-003, "Enforcement Guidance Memorandum on Dispositioning Boiling Water Reactor (BWR) Licensee Non-compliance with TS Containment Requirements during Operations with a Potential for Draining the Reactor Vessel," on October 4, 2011, allowing for the exercise of enforcement discretion for such OPDRV-related TS violations, when certain criteria are met. These criteria include, but are not limited to, the licensee taking measure to ensure adequate water inventory and makeup capability, and the licensee maintaining defense in depth against fission product release. The EGM most recently revised on December 13, 2013, also requires that, to be eligible for discretion, a licensee must submit a license amendment request (LAR) to accept the NRC's generic change to the Standard Technical Specifications (STS) that will allow a graded approach to OPDRV requirements. The LAR must be submitted within twelve months of NRC publication of the generic change in the Federal Register.

The NRC concluded that, for the specified periods, Susquehanna met the EGM criteria and has committed to submit the LAR, as required. Therefore, I have been authorized, after consultation with the Director, Office of Enforcement, and the Regional Administrator, to exercise enforcement discretion, in accordance with NRC Enforcement Policy Section 2.2.4, "Exceptions to Using Only the Operating Reactor Assessment Program," and Section 3.5 "Violations Involving Special Circumstances," and refrain from issuing enforcement for the violation, subject to a timely LAR being submitted.

Finally, the inspectors reviewed Licensee Event Reports (LERs) 50-387/2014-011-00 and 50-388/2015-004-00, both that described the details associated with reactor coolant system (RCS) pressure boundary leakage from reactor recirculation pump small bore pipe socket welds. Although these issues constitute violations of TSs involving the reactor coolant pressure boundary, the NRC concluded that these issues were not within Susquehanna's ability to foresee and correct, Susquehanna's actions did not contribute to the degraded condition, and actions taken were reasonable to address these matters. As a result, the NRC did not identify a performance deficiency. A risk evaluation was performed and the issues were determined to be of very low safety significance. Based on the results of the NRC's inspection and assessment, I have been authorized, after consultation with the Director, Office of Enforcement, and the Regional Administrator to exercise enforcement discretion in accordance with NRC Enforcement Policy Section 2.2.4, "Exceptions to Using Only the Operating Reactor Assessment Program," and Section 3.5, "Violations Involving Special Circumstances."

If you contest the non-cited violations in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at SSES. In addition, if you disagree with the cross-cutting aspect assigned to any finding, or a finding not associated with a regulatory requirement in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region I, and the NRC Resident Inspector at SSES.

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

Sincerely,

/RA/

Michael L. Scott, Director
Division of Reactor Projects

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

Enclosure:
Inspection Report 05000387/2015003
and 05000388/2015003
w/Attachment: Supplementary Information

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Sincerely,

/RA/

Michael L. Scott, Director
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos.:	50-387 and 50-388
License Nos.:	NPF-14 and NPF-22
Report No.:	05000387/2015003 and 05000388/2015003
Licensee:	Susquehanna Nuclear, LLC (Susquehanna)
Facility:	Susquehanna Steam Electric Station, Units 1 and 2
Location:	Berwick, Pennsylvania
Dates:	July 1, 2015 through September 30, 2015
Inspectors:	J. Greives, Senior Resident Inspector T. Daun, Resident Inspector N. Graneto, Operations Engineer C. Graves, Health Physicist E. H. Gray, Senior Reactor Inspector B. Smith, Resident Inspector
Approved By:	Fred L. Bower, III, Chief Reactor Projects Branch 4 Division of Reactor Projects

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SUMMARY

IR 05000387/2015003, 05000388/2015003; July 1, 2015 to September 30, 2015; Susquehanna Steam Electric Station, Units 1 and 2; Plant Modifications, Problem Identification and Resolution, and Follow-Up of Events and Notices of Enforcement Discretion.

This report covered a three-month period of inspection by resident inspectors and announced baseline inspections performed by regional inspectors. The inspectors identified three non-cited violations, all of that 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: Mitigating Systems

Green. Inspectors identified a finding of very low safety significance (Green) and associated NCV of SSES Unit 1 and 2 TS 5.4.1, “Procedures,” because Susquehanna did not maintain the procedure for operation of the residual heat removal (RHR) shutdown cooling (SDC) system consistent with the requirements in TS 3.4.8, “RHR Shutdown Cooling- Hot Shutdown.” As TS 3.4.8 requires two RHR SDC loops to be operable and, if no reactor recirculation pumps (RRPs) are running, one of the loops to be in-service in Mode 3 below the RHR cut in permissive pressure (98 psig), inspectors determined that OP-1(2)49-002, “RHR Shutdown Cooling,” was not maintained appropriately because a change to the procedure precluded operation of the system between 40 psig and the RHR cut in permissive pressure (98 psig). Susquehanna entered the issue into the corrective action program (CAP) as CR-2015-22882 and CR-2015-24137 and revised the procedure to remove the requirement that precluded operation of the SDC system between 40 psig and the RHR cut in permissive pressure.

This finding is more than minor because it is associated with the procedure quality attribute of the Mitigating Systems cornerstone and adversely affected its objective to ensure the availability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the 40 psig procedural limit impacted the availability and capability of RHR to be placed in SDC between 98 psi, the cut-in permissive for the system, and 40 psig. In accordance with Exhibit 2 of IMC 0609, Appendix A, “The SDP for Findings At-Power,” the inspectors determined that this finding is of very low safety significance (Green) because the performance deficiency was not a design or qualification deficiency, did not involve an actual loss of safety function, did not represent actual loss of a safety function of a single train for greater than its TS allowed outage time, and did not screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event. This finding had a cross-cutting aspect in the area of Human Performance, Change Management because Susquehanna did not use a systematic process for evaluating and implementing change so that nuclear safety remains the overriding priority (H.3). Specifically, implementation of Susquehanna’s procedure change process did not ensure that the RHR SDC procedure was maintained consistent with the requirements of plant TSs.

Green. A self-revealing finding of very low safety significance (Green) and associated NCV of 10 CFR 50 Appendix B, Criterion V, “Instructions, Procedures and Drawings,” was identified when Susquehanna inadvertently operated the ‘C’ emergency diesel generator (EDG) mode switch during the performance of a job performance measure (JPM). Specifically, the student performing the JPM operated plant equipment that was contrary to the quality assurance program requirement to only simulate equipment operation. Susquehanna entered the issue into the CAP as CR-2015-19578, the ‘C’ EDG mode switch was restored to the ‘Remote’ position, and the operating crew performed a walk-down of the ‘C’ EDG to confirm proper standby alignment, restoring operability of the EDG.

Inspectors determined that the finding was more than minor because it was associated with the Human Performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the reliability of systems that respond to initiating events to prevent undesirable consequences. Specifically, improper manipulation of the ‘C’ EDG mode switch while simulating a task resulted in an inoperable condition since the EDG would not have auto started, if required. In accordance with Exhibit 2 of IMC 0609, Appendix A, “The SDP for Findings At-Power,” dated June 19, 2012, the inspectors determined that this finding is of very low safety significance (Green) because the performance deficiency was not a design or qualification deficiency, did not involve an actual loss of safety function, did not represent actual loss of a safety function of a single train for greater than its TS allowed outage time, and did not screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event. This finding had a cross-cutting aspect in the area of Human Performance, Avoid Complacency because Susquehanna did not implement appropriate error reduction tools (H.12). Specifically, personnel did not implement appropriate human error prevention tools (e.g. self-check, stop-think-act-review) in accordance with station processes.

Cornerstone: Barrier Integrity

Green. A self-revealing finding of very low safety significance (Green) and associated NCV of SSES Unit 1 and 2 TS 5.4.1, “Procedures,” was identified because Susquehanna incorrectly implemented procedures for maintaining secondary containment integrity. Specifically, on July 27, 2015, maintenance technicians rendered secondary containment for both units inoperable for approximately 44 minutes when a secondary containment boundary door was opened to access the reactor building roof. Susquehanna entered the issue into the CAP as CR-2015-20857 and CR-2015-24442, restored the boundary, and verified the integrity of secondary containment.

The finding was more than minor because it was associated with the Human Performance (Routine OPS/Maintenance Performance) attribute of the Barrier Integrity cornerstone, and affected the cornerstone objective of providing reasonable assurance that physical design barriers (Secondary Containment) protect the public from radionuclide releases caused by accidents or events. Specifically, opening the secondary containment barrier did not maintain reasonable assurance that the secondary containment would be capable of performing its safety function in the event of a reactor accident. The inspectors evaluated the finding in accordance with IMC 0609, Appendix A, "The SDP for Findings At-Power," Exhibit 3, for the Barrier Integrity cornerstone, dated June 19, 2012. The inspectors determined the finding was of very low safety significance (Green) because only represented a degradation of the radiological barrier function of secondary containment provided by the standby gas treatment (SBGT) system. This finding had a cross-cutting aspect in the area of Human Performance, Teamwork because Susquehanna did not effectively communicate and coordinate their activities within and across organizational boundaries to ensure nuclear safety was maintained (H.4). Specifically, when the work plan was changed to accessing the reactor building roof through secondary containment, the change was not effectively communicated to operations department personnel to ensure the secondary containment impairment was appropriately controlled.

Other Findings

A violation of very low safety significance that was identified by Susquehanna was reviewed by the inspectors. Corrective actions taken or planned by Susquehanna have been entered into Susquehanna's CAP. This violation and corrective action tracking number are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 began the inspection period at 61 percent power following an unexpected isolation of extraction steam to the '5C' feedwater heater on June 30, 2015. Following repair of a failed level switch, operators commenced power ascension and achieved 100 percent power on July 1, 2015. On September 11, 2015, operators reduced power to approximately 60 percent to perform a planned rod sequence exchange and control rod scram time testing. Following the maintenance, operators returned the unit to 100 percent on September 16, 2015. On September 18, 2015, operators reduced power to 71 percent to perform a planned rod pattern adjustment and power was restored to 100 percent on September 20, 2015. The unit remained at or near 100 percent power for the remainder of the inspection period.

Unit 2 began the inspection period at 100 percent power and operated at full power until August 14, 2015, when operators reduced reactor power to 66 percent power for a planned rod sequence exchange. Power was restored to 100 percent on August 15, 2015, 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 – 2 samples)

a. Inspection Scope

The inspectors performed partial walkdowns of the following systems:

- Unit 2, 'B' emergency switchgear room cooler while 'A' cooler out of service for maintenance on July 29, 2015
- Unit 1, 'A' loop of RHR while 'D' RHR pump was out of service for maintenance on August 4, 2015

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

b. Findings

No findings were identified.

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 2, emergency switchgear rooms, elevations 719' and 749' (fire zones 2-4C, 2-4D, 2-5F, 2-5G) on July 29, 2015
- Unit 2, general access and remote shutdown panel room, elevation 670' (fire zones 2-2A and 2-2B) on August 11, 2015
- Common, control room and technical support center (fire zones 0-26A, 0-26E, 0-26F, 0-26G, 0-26H, 0-26I, 0-26J, 0-26K, and 0-26L) on August 17, 2015
- Unit 2, battery rooms (fire zones 0-28C, 0-28D, 0-28E, 0-28F, 0-28G and 0-28T) on September 2, 2015
- Unit 1, battery rooms (fire zones 0-28I, 0-28J, 0-28K, 0-28L, 0-28M and 0-28N) on September 2, 2015

b. Findings

A licensee-identified violation is documented in Section 4OA7 of this report.

1R06 Flood Protection Measures (71111.06 – 1 sample)

.1 Annual Review of Cables Located in Underground Bunkers/Manholes

a. Inspection Scope

The inspectors conducted an inspection of underground bunkers/manholes subject to flooding that contain cables whose failure could affect risk-significant equipment. The inspectors performed walkdowns of risk-significant areas, including MH016, MH027 and MH028, containing safety-related medium voltage power cables for both divisions of emergency service water and RHR service water pumps, to verify that the cables were not submerged in water, that cables and/or splices appeared intact, and to observe the condition of cable support structures. The inspectors also ensured that Susquehanna's strategy for dewatering the manholes provided reasonable assurance that they cables would not become submerged.

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 July 14, 2015, that included an earthquake induced steam line rupture with resultant reactor scram. 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 September 11, 2015, inspectors observed the control room operators perform a down power and scram time testing of control rods on Unit 1. The inspectors observed pre-shift briefings and reactivity control briefings to verify that the briefings 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 – 3 samples)

a. Inspection Scope

The inspectors reviewed the samples listed below to assess the effectiveness of maintenance activities on structure, system, and component 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 structure, system, or component 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 structures, systems, and components classified as (a)(1), the inspectors assessed the adequacy of goals and corrective actions to return these structures, systems, and components to (a)(2). Additionally, the inspectors ensured that Susquehanna staff was identifying and addressing common cause failures that occurred within and across maintenance rule system boundaries.

- Unit 1, repetitive failures of reactor water cleanup (RWCU) differential flow isolation function
- Common, 'E' EDG unavailability during diesel maintenance
- Unit 2, excessive unavailability for 'B' RHR suppression pool cooling

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 1, 'D' RHR pump out of service for planned maintenance on August 3, 2015
- Unit 2, division II RHR and swing bus motor-generator set out of service for planned maintenance on August 6, 2015
- Common, yellow risk during planned ultimate heat sink valve maintenance on August 19, 2015
- Common, yellow risk during planned testing of division II ESW system on August 21, 2015
- Common, management of refuel floor loading during dry fuel storage campaign on August 27, 2015

b. Findings

No findings were identified.

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

a. Inspection Scope

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

- Unit 1, failure of steam leak detection for reactor core isolation cooling (RCIC) system on May 7, 2015
- Unit 2, failed as-left calibration of local leak rate test equipment on July 1, 2015
- Unit 2, emergency core cooling systems (ECCS) keepfill tank isolation valve left closed on July 16, 2015
- Unit 1, RWCU differential flow indicator downscale indication on July 27, 2015
- Unit 2, missing insulation on ‘B’ emergency switchgear room cooler on August 3, 2015
- Common, operator workaround (OWA) management on August 21, 2015

The inspectors evaluated the technical adequacy of the operability determinations to assess whether TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TSs and UFSAR to 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 change to the procedure for placing RHR SDC in service implemented by actions 1747541 and ACT-10-CR-1746612. The inspectors verified that the design bases, licensing bases, and performance capability of the affected systems were not degraded by the procedure change. In addition, the inspectors reviewed modification documents associated with the change.

b. Findings

Introduction. Inspectors identified a finding of very low safety significance (Green) and associated NCV of SSES Unit 1 and 2 TS 5.4.1, “Procedures,” because Susquehanna did not maintain the procedure for operation of the RHR SDC system consistent with the requirements in TS 3.4.8, “RHR Shutdown Cooling- Hot Shutdown.” Specifically, OP-1(2)49-002, “RHR Shutdown Cooling,” was not maintained consistent with TS 3.4.8 because it precluded operation of the system between 40 psig and the RHR cut in permissive pressure (98 psig).

Description. While placing a loop of RHR in SDC in September 2013, a thermal relief valve on the discharge line lifted, resulting in elevated room temperatures and fire detection alarms. During their investigation, Susquehanna determined that the margin between the relief valve setpoint (450 psig) and the pressure spike from RHR pump starts were insufficient to ensure the relief valve remains closed. As a corrective action, Susquehanna revised OP-1(2)49-002, “RHR Shutdown Cooling,” to delay placing SDC in service until RPV pressure had been lowered to below 40 psig to provide additional margin to the relief valve setpoint. Specifically, Revision 54 (September 2013) required RPV pressure to be lowered to 50 psig prior to placing SDC in service. This was changed to 40 psig in Revision 58 (March 2014). During the technical review of both revisions required by NDAP-QA-0004, “Procedure Change Process,” Susquehanna determined the procedure changes were acceptable and, in part, that the limits were consistent with plant TSs. Susquehanna assessed 50.59 applicability per NDAP-QA-0726 “10 CFR 50.59 and 10 CFR 72.48 Implementation,” Revision 13, and determined that the changes to the procedures did not require a 50.59 evaluation. Prior to this procedure change SDC could be placed in service as soon as the RHR cut-in permissive had cleared at 98 psig. OP-1(2)49-002 is used not only during a routine shutdown with all systems available, but also as directed by EOPs in response to plant transients to reach cold shutdown conditions.

Inspectors questioned the acceptability of including a step that limits the availability of RHR SDC and the results of the procedure change review process. Specifically, Unit 1 and 2 TS 3.4.8, “RHR Shutdown Cooling- Hot Shutdown,” requires two RHR SDC loops to be operable and, if no RRPs are running, one of the loops to be in-service. This TS limiting condition for operation (LCO) is applicable in Mode 3 with reactor steam dome pressure less than the RHR cut in permissive pressure (98 psig). There is a note that allows a delay of up to 2 hours for placing a loop in service prior to declaring the LCO not met to allow time to prepare RHR for the SDC mode. There were no precautions or warnings in the procedure to note any of the requirements in the TS LCO. Inspectors determined that the procedure change, as written, was inconsistent with the TS requirements because it would have precluded operators from placing either loop of SDC in service between the RHR cut-in permissive pressure (98 psig) and the 40 psig procedural limitation, making the SDC mode of RHR unavailable in mode 3 for the specified pressure band possibly for greater than the two hours allowed by TS 3.4.8. Consequently, inspectors determined that one of the questions on the NDAP-QA-0004 procedure change technical review checklist, which asks in part, if all acceptance criteria, limits and specifications in the procedure are consistent with TSs, should have been answered “No.” Answering this question “no” would have identified that the procedure change should not have been made as written and would have prompted further review and assessment.

Analysis. Failure to have an adequate procedure for placing RHR SDC in service was a performance deficiency that was within Susquehanna's ability to foresee and correct and should have been prevented. Specifically, Susquehanna's technical review of procedure changes that limited the availability of RHR SDC to below 40 psig did not identify that the limit was inconsistent with the plant TSs and would not have provided reasonable assurance that both loops of SDC remained operable as required by TS LCO 3.4.8. This finding is more than minor because it is associated with the procedure quality attribute of the Mitigating Systems cornerstone and adversely affected its objective to ensure the availability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the 40 psig procedural limit impacted the availability and capability of RHR to be placed in SDC between 98 psi, the cut-in permissive for the system, and 40 psig.

In accordance with Exhibit 2 of IMC 0609, Appendix A, "The SDP for Findings At-Power," dated June 19, 2012, the inspectors determined that this finding is of very low safety significance (Green) because the performance deficiency was not a design or qualification deficiency, did not involve an actual loss of safety function, did not represent actual loss of a safety function of a single train for greater than its TS allowed outage time, and did not screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event.

This finding had a cross-cutting aspect in the area of Human Performance, Change Management, because Susquehanna did not use a systematic process for evaluating and implementing change so that nuclear safety remains the overriding priority (H.3). Specifically, implementation of Susquehanna's procedure change process did not ensure that the RHR SDC procedure was maintained consistent with the requirements of plant TSs.

Enforcement. SSES Unit 1 and 2 TSs 5.4.1, "Procedures," requires that written procedures be maintained for activities recommended in Regulatory Guide (RG) 1.33, "Quality Assurance Program Requirements (Operation)," Appendix A, Revision 2. RG 1.33 requires procedures for operation of the SDC system. TS LCO 3.4.8, "RHR Shutdown Cooling- Hot Shutdown," requires two RHR SDC loops to be operable and, if no RRPs are running, one of the loops to be in-service in Mode 3 with reactor steam dome pressure less than the RHR cut in permissive pressure (98 psig). Contrary to the above, from September 2013 until September 2015, Susquehanna maintained OP-1(2)49-002 "RHR Shutdown Cooling," inconsistent with plant TS 3.4.8 when a change was made to the procedure that precluded the operation of RHR SDC in Mode 3 between 40 psig and 98 psig without a time limit. To restore compliance, Susquehanna revised the procedure to remove the requirement that precluded operation of the SDC system between 40 psig and the RHR cut in permissive pressure. Because this violation was of very low safety significance, was not repetitive or willful, and was entered into Susquehanna's CAP as CR-2015-22882 and CR-2015-24137, this violation is being treated as a NCV consistent with Section 2.3.2.a of the Enforcement Policy. **(NCV 05000387;388/2015003-01, RHR Shutdown Cooling Procedure Not Maintained Consistent with Technical Specification Requirements)**

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

a. Inspection Scope

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

- Unit 1, planned maintenance on the ‘A’ standby liquid control (SLC) pump on July 2, 2015
- Common, ‘D’ EDG cylinder head inspection on July 17, 2015
- Unit 1, ‘D’ RHR pump motor oil cooler emergency service water check valve maintenance on August 5, 2015
- Unit 1, ‘D’ main steam line high flow switch replacement on August 12, 2015
- Common, ‘B’ EDG Work on September 1, 2015
- Unit 1, ‘B’ alternate reactor protection system (RPS) electrical protection assembly (EPA) breaker card replacement on September 18, 2015

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22 – 3 samples)

a. Inspection Scope

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

- Unit 1, daily and shiftly surveillances on July 23, 2015
- Unit 2, SLC system quarterly flow surveillance on July 31, 2015 (IST)
- Unit 1, 24-month calibration of main steam line flow channels August 12, 2015

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06 – 2 samples)

.1 Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of two routine Susquehanna emergency drills on July 14, 2015, and September 1, 2015, to identify any weaknesses and deficiencies in the classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the simulator and technical support center 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.

2. RADIATION SAFETY

Cornerstone: Occupational/Public Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01 – 1 sample)

a. Inspection Scope

During August 17-20, 2015, the inspectors reviewed Susquehanna's performance in assessing and controlling radiological hazards in the workplace. The inspectors used the requirements contained in 10 CFR 20, TSs, applicable Regulatory Guides (RGs), and the procedures required by TSs as criteria for determining compliance.

Inspection Planning

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

Radiological Hazard Assessment

The inspectors reviewed recent plant radiation surveys and any changes to plant operations since the last inspection to identify any new radiological hazards for onsite workers or members of the public.

Instructions to Workers

The inspectors observed several containers of radioactive materials and assessed whether the containers were labeled and controlled in accordance with requirements.

The inspectors reviewed several occurrences where a worker's electronic personal dosimeter alarmed. The inspectors reviewed the Susquehanna's evaluation of the incidents, documentation in the CAP, and whether compensatory dose evaluations were conducted when appropriate.

Contamination and Radioactive Material Control

The inspectors observed the monitoring of potentially contaminated material leaving the radiological control area and inspected the methods and radiation monitoring instrumentation used for control, survey, and release of that material.

Radiological Hazards Control and Work Coverage

The inspectors evaluated in-plant radiological conditions and performed independent radiation measurements during facility walk-downs and observation of radiological work activities. The inspectors assessed whether posted surveys, radiation work permits (RWPs), worker radiological briefings, the use of continuous air monitoring and dosimetry monitoring were consistent with current plant conditions.

Risk-Significant High Radiation Areas (HRA) and Very High Radiation Area (VHRA) Controls

The inspectors reviewed the controls and procedures for HRAs, VHRAs, and radiological transient areas in the plant.

Problem Identification and Resolution

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

b. Findings

No findings were identified

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

a. Inspection Scope

During August 17-20, 2015, the inspectors reviewed the control of in-plant airborne radioactivity and the use of respiratory protection devices for radiological protection. The inspectors used the requirements in 10 CFR 20, RG 8.15, RG 8.25, NUREG/CR-0041, TS, and procedures required by TS as criteria for determining compliance.

Inspection Planning

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

Use of Respiratory Protection Devices

The inspectors reviewed the adequacy of Susquehanna's use of respiratory protection devices in the plant to include applicable As Low As is Reasonably Achievable (ALARA) evaluations, respiratory protection device certification, respiratory equipment storage, air quality testing records, and individual qualification records.

Self-Contained Breathing Apparatus (SCBA) for Emergency Use

The inspectors reviewed the following: the status and surveillance records for three SCBAs staged in-plant for use during emergencies; SCBA procedures and maintenance and test records; the refilling and transporting of SCBA air bottles; SCBA mask size availability; and the qualifications of personnel performing service and repair of this equipment.

Problem Identification and Resolution

The inspectors evaluated whether problems associated with the control and mitigation of in-plant airborne radioactivity were identified at an appropriate threshold and addressed by the Susquehanna's CAP.

b. Findings

No findings were identified

2RS4 Occupational Dose Assessment (71124.04 – 1 sample)

a. Inspection Scope

During August 17-20, 2015, the inspectors reviewed the monitoring, assessment, and reporting of occupational dose. The inspectors used the requirements in 10 CFR 20, Regulatory Guides, TSs, and procedures required by TSs as criteria for determining compliance.

Inspection Planning

The inspectors reviewed: radiation protection program audits; National Voluntary Laboratory Accreditation Program (NVLAP) dosimetry testing reports; and procedures associated with dosimetry operations.

External Dosimetry

The inspectors reviewed: dosimetry NVLAP accreditation; onsite storage of dosimeters; the use of “correction factors” to align electronic personal dosimeter (EPD) results with NVLAP dosimetry results; dosimetry occurrence reports; and CAP documents for adverse trends related to external dosimetry.

Internal Dosimetry

The inspectors reviewed: internal dosimetry procedures; whole body counter measurement sensitivity and use; adequacy of the program for whole body count monitoring of plant radionuclides; adequacy of the program for dose assessments based on air sample monitoring and the use of respiratory protection; and internal dose assessments for any actual internal exposures.

Special Dosimetric Situations

The inspectors reviewed: worker notification of the risks of radiation exposure to the embryo/fetus; the dosimetry monitoring program for declared pregnant workers; external dose monitoring of workers in large dose rate gradient environments; and dose assessments performed since the last inspection that used multi-badging, skin dose or neutron dose assessments.

Problem Identification and Resolution

The inspectors evaluated whether problems associated with occupational dose assessment were identified at an appropriate threshold and properly addressed in the CAP.

b. Findings

No findings were identified

Cornerstone: Public Radiation Safety (PS)

2RS7 Radiological Environmental Monitoring Program (71124.07 - 1 sample)

a. Inspection Scope

The inspectors reviewed the Radiological Environmental Monitoring Program (REMP) to validate the effectiveness of the radioactive gaseous and liquid effluent release program. The inspectors used the requirements in 10 CFR 20; 40 CFR 190; 10 CFR 50 Appendix I; and the site’s TSs, Offsite Dose Calculation Manual (ODCM), and procedures required by TSs as criteria for determining compliance.

Inspection Planning

The inspectors reviewed the Susquehanna’s 2013 and 2014 annual radiological environmental and effluent monitoring reports; REMP program audits; ODCM changes; land use census; and inter-laboratory comparison program results.

Onsite Inspection

The inspectors reviewed and/or observed the following items:

- Sample collection, monitoring, and dose measurement stations (e.g., thermoluminescent dosimeter, air monitoring, vegetation, milk)
- Calibration and maintenance records for air sample and dosimetry measurement equipment
- Environmental sampling of the effluent release pathways specified in the ODCM
- Meteorological tower and meteorological data readouts
- Meteorological instrument operability status and calibration results
- Missed and anomalous environmental samples identified, resolved, and reported in the annual radioactive environmental monitoring report
- Positive environmental sample assessment results
- The groundwater monitoring program as it applies to selected potential leaking structures, systems, or components (SSCs)
- 10 CFR 50.75(g) records of leaks, spills, and remediation since the previous inspection
- Changes to the ODCM due to changes to the land use census, long-term meteorological conditions, and/or modifications to the environmental sample stations
- Environmental sample laboratory analysis results, and measurement detection sensitivities
- Results of the laboratory quality control program audit, and the inter-and intra-laboratory comparison program results

Identification and Resolution of Problems

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

b. Findings

No findings were identified

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Radiological Effluent TS/ODCM Radiological Effluent Occurrences (1 sample)

a. Inspection Scope

The inspectors reviewed Susquehanna's submittals for the radiological effluent TS/ODCM radiological effluent occurrence Performance Indicator (PI) for the 1st quarter 2014 through the 4th quarter 2014. The inspectors used PI definitions and guidance contained in the Nuclear Energy Institute Document 99-02, Revision 7, to determine if the PI data was reported properly. The inspectors reviewed the public dose assessments for the PI for public radiation safety to determine if related data was accurately calculated and reported.

The inspectors reviewed the CAP database to identify any potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose. The inspectors reviewed gaseous and liquid effluent summary data and the results of associated offsite dose calculations to determine if indicator results were accurately reported.

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index (6 samples)

a. Inspection Scope

The inspectors reviewed Susquehanna's submittal of the Mitigating Systems Performance Index for the following systems for the period of July 1, 2014, through June 30, 2015:

- Units 1 and 2, high pressure injection system
- Units 1 and 2, emergency alternating current power system
- Units 1 and 2, heat removal systems

To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7. The inspectors also reviewed Susquehanna's operator narrative logs, CRs, mitigating systems performance index derivation 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 – 2 samples)

.1 Routine Review of Problem Identification and Resolution Activities

a. Inspection Scope

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

b. Findings

Introduction. A self-revealing finding of very low safety significance (Green) and associated NCV of 10 CFR 50 Appendix B, Criterion V, “Instructions, Procedures and Drawings,” was identified when Susquehanna inadvertently operated the ‘C’ EDG mode switch during the performance of a JPM. Specifically, the student performing the JPM operated plant equipment that was contrary to the quality assurance program requirement to only simulate equipment operation.

Description. On July 14, 2015, a licensed operator class student performing an in-plant JPM that inadvertently repositioned the ‘C’ EDG mode switch from the ‘Remote’ to the ‘Local’ position. This placed the ‘C’ EDG in an inoperable condition since the EDG was prevented from auto starting and performing its safety-related function. TQ-203, “On-the-Job Training and Task Performance Evaluation,” Revision 3, step 5.4.1 states, in part, that actual plant equipment not be manipulated during “simulate” tasks. Upon realization of the error, the ‘C’ EDG mode switch was placed back to the ‘Remote’ position. The operating crew performed a walk-down of the ‘C’ EDG to confirm proper standby alignment, restoring system operability. During this period, Units 1 and 2 were placed into a yellow Equipment Out-of-Service risk configuration.

Analysis. The inspectors determined that operation of plant equipment during the performance of a JPM was a performance deficiency that was within Susquehanna’s ability to foresee and correct and should have been prevented. Inspectors determined that the finding was more than minor because it was associated with the Human Performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the reliability of systems that respond to initiating events to prevent undesirable consequences. Specifically, improper manipulation of the ‘C’ EDG mode switch while simulating a task resulted in an inoperable condition since the EDG would not have auto started, if required. In accordance with Exhibit 2 of IMC 0609, Appendix A, “The SDP for Findings At-Power,” dated June 19, 2012, the inspectors determined that this finding is of very low safety significance (Green) because the performance deficiency was not a design or qualification deficiency, did not involve an actual loss of safety function, did not represent actual loss of a safety function of a single train for greater than its TS allowed outage time, and did not screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event.

This finding had a cross-cutting aspect in the area of Human Performance, Avoid Complacency because Susquehanna did not implement appropriate error reduction tools (H.12). Specifically, personnel did not implement appropriate human error prevention tools (e.g. self-check, stop-think-act-review) in accordance with station processes.

Enforcement. 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures and Drawings,” requires, in part, that activities affecting quality be prescribed by procedures, and be accomplished in accordance with those procedures. Training program quality assurance procedure (TQ)-203, “On-The-Job Training and Task Performance Evaluation,” Revision 3, is a quality assurance program procedure that, in part, describes the requirements and expectations for implementing simulated in-plant training. Step 5.4.1 of TQ-203 states, in part, that actual plant equipment not be manipulated during “simulate” tasks.

Contrary to the above, on July 14, 2015, a training activity affecting quality was not accomplished in accordance with the prescribed TQ-203 procedure requirement that prohibits manipulating actual plant equipment during "simulate" tasks. Specifically, while performing a "simulate" task during an in-plant training JPM, a student repositioned the 'C' EDG mode switch from the 'Remote' to the 'Local' position and rendered the 'C' EDG inoperable. Upon realization of the error, corrective action was taken to restore the 'C' EDG mode switch to the 'Remote' position. The on-shift operating crew walked down the 'C' EDG to confirm proper standby alignment to restore operability. Because this violation was of very low safety significance (Green), and this performance deficiency was entered into the CAP as CR-2015-19578, this finding is being treated as an NCV in accordance with Section 2.3.2.a of the NRC Enforcement Policy. **(NCV 05000387(388)/2015003-02, 'C' EDG Rendered Inoperable by Switch Manipulation during Training Simulation)**

.2 Annual Sample: Recirculation System Vibrations

a. Inspection Scope

Coincident with the review of LERs 387/2014-011 and 388/2015-014, during July 20-21, 2015 an inspection was conducted of the work completed by the Susquehanna staff on prevention, identification and resolution of fatigue initiated degradation in small diameter piping, socket welds and jet pump vibration. Susquehanna's actions were compared to NRC requirements and applicable guidance in Electrical Power Research Institute (EPRI) and equipment supplier documents.

b. Findings and Observations

No findings were identified.

The inspector determined that the plant staff had conducted an initiative, starting in 2005, to identify the number and location of recirculation system socket welds and apply a fatigue mitigation plan of ultrasonic test (UT) examination, and addition of a "2 to 1" weld configuration, to a selected portion of the socket welds. In response to questions raised by the inspectors, two CRs, 2015-20466 and 20467, were initiated for the Susquehanna engineering staff to evaluate the applicability of the 2005 program to the current plant conditions. Additionally, the jet pump vibration effects were examined by Susquehanna's visual observation during previous refuel outages, with the implementation of mitigation modifications and adjustments. The inspector determined that additional work was in progress to develop and install jet pump vibration reduction modifications in future outages. The inspectors concluded that Susquehanna's planned actions were reasonable.

.3 Annual Sample: Emergency Service Water (ESW) Supply lines to the 1C, 1D, 2C, and 2D Residual Heat Removal (RHR) pump motor oil coolers fatigue mitigation

a. Inspection Scope

The inspectors reviewed the SSES analysis initiated in CR-2014-20129, and their follow-up Engineering evaluation EWR AR-2014-20505. This CR described vibration observed in the ESW piping to the 1C, 1D, 2C, and 2D RHR pump motor oil coolers when a residual heat removal service water (RHRSW) pump is operating and ESW is not operating. The concern involved the potential for the vibration to cause a through-wall pipe leak as a result of mechanical fatigue. The inspector also reviewed CR 2014-21114 R1, and its apparent cause evaluation that provided additional detail regarding the ESW piping vibration condition causes and corrective actions as related to the various operational line up combinations of ESW, RHR motor coolers and the RHRSW system. The identification of this issue by SSES staff was previously reviewed by NRC inspectors and the subject of a violation documented in NRC Inspection Report 05000387(388)/2014009 dated August 1, 2014. (ADAMS Accession No. ML14216A216)

These CRs and related evaluations were reviewed with a focus on the problem evaluation scope, extent of condition, and adequacy and timing of corrective actions. The inspector determined whether these activities were completed in accordance with the guidance in the SSES CAP procedures and 10 CFR 50 Appendix B requirements related to corrective actions.

The inspector completed document reviews and discussed the issue with responsible SSES staff. The inspector also walked down the affected systems. While the affected piping was in the 1C, 1D, 2C, and 2D loops, the inspector included the 1A, 1B, 2A, and 2B loops in the walk down scope for the conditions of the ESW system not running, and with and without the Unit 2, division 2 RHR pump running.

b. Findings

No findings were identified.

For the ESW, RHR pump motor oil coolers vibrations identified when an RHRSW pump is operating and ESW is not operating, the inspector found an identified problem, cause analysis and implementation of corrective actions, with planning in place to confirm effectiveness of the corrective actions. The inspector reviewed the fatigue calculation in EWR AR-2014-20505 and determined that, considering the number of expected cycles per year and the event fatigue stress levels, the estimate of piping acceptability exceeded 15 years. Since this vibration condition existed intermittently since 2008, the inspector concluded the piping remained within fatigue limits. The preventive measure to eliminate the fatigue driver was to increase the spring capability in selected ESW system check valves. This work was partly completed at the time of inspection, September 1, 2015, and was scheduled for final completion during the following weeks. The inspector concluded the timing of the corrective actions to change out springs in ESW check valves was appropriate to maintain system operability.

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153 – 11 samples)

.1 Plant Events

a. Inspection Scope

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

- Unplanned Unit 1 downpower to 61 percent due to isolation of extraction steam to the ‘5C’ feedwater heater on June 30, 2015

b. Findings

No findings were identified.

.2 LERs associated with Secondary Containment

a. Inspection Scope

The following LERs and associated evaluations were reviewed for accuracy, the appropriateness of corrective actions, violations of requirements, and potential generic issues. These LERs are closed.

(Closed) LER 05000387; 388/2015-002-00: Secondary Containment Inoperability Due to Failure to Meet Technical Specification Surveillance Requirement 3.6.4.1.1

On April 11, 2015, secondary containment differential pressure dropped below the TS limit of 0.25 inches water column due to a failure of the non-safety related normal reactor building ventilation system. During their investigation, Susquehanna identified an air solenoid valve that controls the running fan’s discharge damper to be porting air, such that both exhaust fans did not continue to run. The failed solenoid valve was isolated and replaced. This equipment failure did not impact the availability of the safety-related SBGT system to maintain a vacuum in secondary containment in an emergency. This was documented as CR-2015-09893.

(Closed) LER 05000387; 388/2015-003-00: Secondary Containment Inoperability due to Failure to Meet Technical Specification Surveillance Requirement 3.6.4.1.1

On April 21, 2015, secondary containment differential pressure dropped below the TS limit of 0.25 inches water column due to a failure of the non-safety related normal reactor building ventilation system. Susquehanna’s investigation concluded that a wind gust likely resulted in reaching a differential pressure limit for tripping of the running fan. Normally this would result in starting of the standby fan, but did not due to an abnormal switch lineup for an in-progress surveillance test. The short duration loss of secondary containment differential pressure did not impact the availability of the safety-related SBGT system to maintain a vacuum in secondary containment in an emergency. This was documented in CR-2015-11377.

(Closed) LER 05000387; 388/2015-004-00: Secondary Containment Inoperable due to Secondary Containment Boundary Door Found Ajar

On April 27, 2015, secondary containment was declared inoperable when a plant operator found a secondary containment boundary door unlatched and protruding approximately 0.25 inches from the door frame. TS 3.6.4.1. “Secondary Containment,” surveillance requirement 3.6.4.1.3, requires each secondary containment boundary door to be closed. Susquehanna performed an engineering analysis and determined that the condition of the door would not have affected the ability of the safety-related system to draw and maintain a vacuum in secondary containment in an emergency. Additionally, Susquehanna's analysis concluded that differential pressure in an emergency would apply pressure on the door in the closed direction, and that air in-leakage past the 0.25 inch gap would not have exceeded the allowable in-leakage per the TS. This was documented in CR-2015-12048.

<u>(Closed) LER 05000387; 388/2015-005-00</u>: Loss of Secondary Containment Differential Pressure during Ventilation Damper Testing

On May 4, 2015, secondary containment differential pressure dropped below the TS limit of 0.25 inches water column due to a failure of the non-safety related normal reactor building ventilation system, rendering secondary containment inoperable due to failure to meet TS 3.6.4.1. "Secondary Containment," surveillance requirement 3.6.4.1.1, that requires secondary containment differential pressure to be maintained above 0.25 inches water column. Susquehanna's investigation determined that two work activities on the non-safety related ventilation system were not coordinated appropriately such that an open pathway from secondary containment to the environment was established inadvertently. Susquehanna determined that the condition did not impact the ability to isolate secondary containment and draw and maintain differential pressure with the safety-related SBGT system. This was documented in CR-2015-13198.

<u>(Closed) LER 05000388/2015-002-00</u>: Secondary Containment Inoperability due to Failure to Meet Technical Specification Surveillance Requirement 3.6.4.1.1

On March 13, 2015, Unit 2 secondary containment differential pressure dropped below the TS limit of 0.25 inches water column due the non-safety related normal reactor building ventilation system being unable to maintain adequate differential pressure. Susquehanna's investigation determined maintenance technicians had propped open a ventilation plenum boundary that created an opening in the secondary containment boundary. Susquehanna determined that the condition did not impact the ability to isolate secondary containment and draw and maintain differential pressure with the safety-related SBGT system. This was documented in CR-2015-07319.

<u>(Closed) LER 05000387; 388/2015-006-00</u>: Secondary Containment Declared Inoperable due to Secondary Containment Boundary Door 104-R Breached

On July 27, 2015, operators received alarms for low reactor building differential pressure and observed that Unit 1 reactor building differential pressure was less than the 0.25 inches water column required by TS 3.6.4.1, "Secondary Containment." As a result, secondary containment was declared inoperable and both Units 1 and 2 entered TS 3.6.4.1. Susquehanna's investigation identified that electricians had accessed the reactor building roof to perform work on a security camera and opened a pathway from secondary containment to the environment. Though they had reported to the control room that they would be accessing the reactor building roof, breakdowns in communication between the technicians and operators resulted in a failure to recognize that the access path chosen rendered secondary containment inoperable. Specifically, operators assumed that the technicians would be accessing the roof via the preferred access point, that maintains secondary containment operability and did not verify the requirements of opening the door in accordance with NDAP-QA-0409, "Door, Floor Plug and Hatch Control," Revision 14. This was documented as CR-2015-09893.

a. Findings

Introduction. A self-revealing finding of very low safety significance (Green) and associated NCV of SSES Unit 1 and 2 TS 5.4.1, "Procedures," was identified because Susquehanna incorrectly implemented procedures for maintaining secondary containment integrity. Specifically, on July 27, 2015, maintenance technicians rendered secondary containment inoperable for both units for approximately 44 minutes when a secondary containment boundary door was opened to access the reactor building roof.

Description. On July 27, 2015, operators received alarms for secondary containment low differential pressure and entered TS LCO 3.6.4.1, "Secondary Containment," for both Units 1 and 2 due to secondary containment being inoperable. TS 3.6.4.1 requires that secondary containment be operable in Modes 1, 2, and 3 as well as during OPDRVs, core alterations, and movement of irradiated fuel in secondary containment. At the time the alarms were received, both units were in Mode 1 at 100 percent rated thermal power. Operators identified that the low differential pressure occurred when maintenance technicians accessed the reactor building roof by opening door 104R, that serves as boundary between secondary containment and the environment. Operators instructed the technicians to exit the roof area and secure the boundary. The door was maintained opened for approximately 44 minutes. Shift supervision was unaware that the technicians had planned to access the reactor building roof by opening the secondary containment boundary door that resulted in the low differential pressure condition. Susquehanna subsequently reported the condition in EN51269 in accordance with 10 CFR 50.72(b)(3)(v)(C).

Historically, the reactor building roof had been accessed via a door in the control structure that is not a secondary containment barrier. Susquehanna performed a prompt human performance investigation for the event and determined that during this evolution radiation protection personnel requested that the roof be accessed from the reactor building in an effort to minimize dose. This change was not communicated to operations department or security personnel. Upon reaching door 104R, the technicians encountered a sign that identified the door as a secondary containment boundary and cautioned that shift supervision permission was required for entry. Maintenance personnel contacted the control room and informed them that they would be accessing the reactor building roof via door 104R, but did not state that the door was a secondary containment barrier. Control room personnel assumed that the door was the one that was historically used without verifying the door number in NDAP-QA-0409, "Door, Floor Plug and Hatch Control," and gave permission to proceed. NDAP-QA-0409 identifies door 104R as a secondary containment boundary and requires that secondary containment be declared inoperable any time a secondary containment boundary is opened for any reason, unless the door is part of an airlock or is being used for ingress/egress and is between two operable secondary containment zones. In this case, door 104R is not part of an airlock and is not located between two secondary containment zones. The boundary remained open for the duration of the roof access evolution.

Analysis. Failure to implement the procedure for control of secondary containment was a performance deficiency that was within Susquehanna's ability to foresee and correct and should have been prevented. The finding was more than minor because it was associated with the Human Performance (Routine Operations/Maintenance Performance) attribute of the Barrier Integrity cornerstone and affected the cornerstone objective of providing reasonable assurance that physical design barriers (Secondary

Containment) protect the public from radionuclide releases caused by accidents or events. Specifically, opening the secondary containment barrier did not maintain reasonable assurance that the secondary containment would be capable of performing its safety function in the event of a reactor accident. The inspectors evaluated the finding in accordance with IMC 0609, Appendix A, "The SDP for Findings At-Power," Exhibit 3, for the Barrier Integrity cornerstone, dated June 19, 2012. The inspectors determined the finding was of very low safety significance (Green) because only represented a degradation of the radiological barrier function of secondary containment provided by the SBGT system.

This finding had a cross-cutting aspect in the area of Human Performance, Teamwork because Susquehanna did not effectively communicate and coordinate their activities within and across organizational boundaries to ensure nuclear safety was maintained (H.4). Specifically, when the work plan was changed to accessing the reactor building roof through secondary containment, the change was not effectively communicated to operations department personnel to ensure the secondary containment impairment was appropriately controlled.

Enforcement. SSES Unit 1 and 2 TSs 5.4.1, “Procedures,” requires that written procedures be implemented for activities recommended in Regulatory Guide 1.33, “Quality Assurance Program Requirements (Operation),” Appendix A, Revision 2. RG 1.33 requires, in part, implementing procedures for maintaining the integrity of secondary containment. NDAP-QA-0409, “Door, Floor Plug and Hatch Control,” requires that secondary containment be declared inoperable any time a secondary containment boundary is opened for any reason, unless the door is part of an airlock or is being used for ingress/egress and is between two operable secondary containment zones.

Contrary to the above, on July 27, 2015, Susquehanna did not declare secondary containment inoperable and take the required TS actions when maintenance technicians opened a non-airlock door (104R) for 44 minutes to access the reactor building roof. To restore compliance, Susquehanna secured the open boundary and verified the integrity of secondary containment. Because this violation was of very low safety significance (Green), and Susquehanna entered this performance deficiency into the CAP as CR-2015-20857 and CR-2015-24442, this finding is being treated as an NCV in accordance with Section 2.3.2.a of the NRC Enforcement Policy. (**NCV 05000387;388/2015003-03, Secondary Containment Inoperability due to Improperly Controlled Access to the Reactor Building Roof)**

.3 (Closed) LER 05000387/2013-009-01: RPS Electrical Protection Assembly (EPA) Logic Card Under-Frequency Trip Setpoints Out of Calibration

During routine surveillance testing on October 30, 2013, the two RPS EPAs in the 'A' train alternate RPS power supply were found to have under frequency trip set points outside of the TS allowable value, thus not meeting TS surveillance requirement 3.3.8.2.2. The affected EPAs were not in service during the testing, and are only required to be operable when the associated alternate RPS power supply is in service. The LER was submitted in accordance with 10 CFR 50.73(a)(2)(v) as a condition that could have prevented the fulfillment of the safety function of the RPS electric power monitoring system because the failures involved the redundant, series connected EPAs in an RPS power supply. In addition, this event was also determined to be a common cause inoperability of independent channels in a single system reportable under 10 CFR 50.73(a)(2)(vii), and a condition prohibited by TSs reportable under 10 CFR 50.73(a)(2)(i)(B).

The NRC reviewed this LER and closed it in inspection report 05000387/2014005 with a Green NCV for not establishing design control measures that provide for verifying or checking the adequacy of design and translating the design basis requirements into allowable values and trip set points. Revision 1 to this LER was submitted in June 2015 with results of a revised apparent cause evaluation. The inspectors reviewed the revision to the LER, Susquehanna's revised apparent cause evaluation, and associated corrective actions. This LER is closed.

.4 (Closed) LER 05000388/2015-005-00: Implementation of Enforcement Guidance Memorandum (EGM) 11-003, Revision 2

From April 14, 2015 to May 16, 2015, Susquehanna performed operations with the potential to drain the reactor vessel (OPDRVs) without establishing secondary containment integrity. An OPDRV is an activity that could result in the draining or siphoning of the reactor pressure vessel (RPV) water level below the top of fuel, without crediting the use of mitigating measures to terminate the uncovering of fuel. TS 3.6.4.1, “Secondary Containment” requires that secondary containment be operable and is applicable during OPDRVs. The required action for this specification, if secondary containment is inoperable in this condition of applicability, is to initiate actions to suspend OPDRVs immediately. Therefore, failing to maintain secondary containment operability during OPDRVs without initiating actions to suspend the operation was considered a condition prohibited by TSs as defined by 10 CFR 50.73(a)(2)(i)(B). As reported in LER 05000388/2015-005, Susquehanna conducted the following OPDRVs during the period of secondary containment inoperability:

- Recirculation system maintenance;
- Reactor Water Cleanup system maintenance;
- RHR system maintenance;
- Hydraulic Control Unit (HCU) maintenance;
- Local Power Range Monitor (LPRM) replacement; and,
- Control rod drive (CRD) mechanism replacements.

NRC EGM 11-03, “Enforcement Guidance Memorandum on Dispositioning BWR Licensee Noncompliance with TS Containment Requirements during Operations with a Potential for Draining the Reactor Vessel,” Revision 2, provides, in part, for the exercise of enforcement discretion only if the licensee demonstrates that it has met four specific criteria during an OPDRV activity. The inspectors’ assessments of Susquehanna’s implementation of these four criteria during the activities are described below:

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

2) The inspectors noted that the reactor vessel water level was maintained above the RHR high water level setpoint of 22 feet. The inspectors also noted that at least one safety-related pump was the standby source of makeup designated in the control room narrative logs for the evolutions. Susquehanna logged that the worst case

estimated time to drain the reactor cavity to the RPV flange was greater than the EGM criteria of 24 hours.

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

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

TS 3.6.4.1 is applicable during OPDRVs and requires that secondary containment be operable. TS 3.6.4.1, action C.3, requires operators to initiate actions to suspend OPDRVs immediately upon discovery that secondary containment is inoperable. Contrary to the above, from April 14, 2015 to May 16, 2015, Susquehanna did not maintain secondary containment operable while performing OPDRVs. Because the violation was identified during the discretion period described in EGM 11-003, the NRC is exercising enforcement discretion (EA-15-161) in accordance with Section 3.5, “Violations Involving Special Circumstances,” of the NRC Enforcement Policy and, therefore, will not issue enforcement action for this violation. In accordance with EGM 11-003, each licensee that receives discretion must submit a license amendment request within 12 months of the NRC staff’s publication in the Federal Register of the notice of availability for a generic change to the Standard TSs to provide more clarity to the term OPDRV. The inspectors observed that Susquehanna is tracking the need to submit a license amendment request in its CAP as CR 1707662. This LER is closed.

.5 <u>(Closed) Licensee Event Report (LER) 05000387/2014-011</u>: “Degraded Condition Due to Reactor Coolant Pressure Boundary Leakage Caused by an Inadequate Weld” on the Unit 1 Reactor Recirculation Pump (RRP) Lower Seal Cavity Vent Piping.

On December 13, 2014, after operators shutdown Unit 1 for a planned forced outage due to increased unidentified leakage, Susquehanna staff identified a small leak by observation on the Unit 1 "B" RRP Lower Seal Cavity Vent ¾” piping, a location that Susquehanna determined was part of the reactor coolant pressure boundary. Susquehanna determined that this leakage constituted a violation of the Unit 1 TS, Section 3.4.4 titled "Reactor Coolant System (RCS)" that requires RCS leakage to be limited to no pressure boundary leakage. Susquehanna staff performed a progressive non-destructive examination of the leak site to further characterize the flaw in the socket weld. The evaluation concluded the flaw was a linear defect approximately 0.74” in length through the center of the socket weld and that the apparent cause was a weld defect created during the vendor’s original manufacturing process. The defective area was removed by progressive grinding with confirmatory liquid penetrant test examinations to confirm the defect length. The weld was repaired and inspected satisfactorily prior to plant startup from the Unit 1 forced outage.

The inspector reviewed LER 2014-011 and CR-2014-37848, Revision 1 that documented the related apparent cause evaluation and failure modes analysis for this condition. The inspector also reviewed weld repair work package No. 140469 and the final nondestructive examination reports of the repair area. The inspector determined Susquehanna staff's leak review and analysis process concluded that the leakage, located in the center of the vendor supplied fillet weld was most likely the result of an original weld flaw. The leak area was removed by grinding and the area was repaired by welding to the required 2 to 1 dimensional fillet weld configuration. Although this leak did not appear to be fatigue or system vibration driven, the inspector noted Susquehanna staff's implementation of mitigation methods to reduce socket weld fatigue susceptibilities, including a 2 to 1 fillet weld configuration and instrumented hammer testing of the completed pipe assembly to confirm the absence of resonant frequency vibration modes, were applied.

LER 2014-011 documents the licensee's violation of TS 3.4.4, limiting reactor pressure boundary leakage during plant operations to zero. This TS violation occurred sometime between unit start-up in July 2014 and plant shutdown on December 13, 2014. The inspector determined that this violation of TS 3.4.4 was more than minor, but not the result of a performance deficiency. In accordance with the NRC Enforcement Policy guidance and IMC 0612, this violation is being treated under the traditional enforcement process and best characterized as a Severity Level IV (very low safety significance) violation, similar to example d.1 in NRC Enforcement Policy, Section 6.1, "Reactor Operations." In addition, using IMC 0609, Significance Determination Process, Appendix A, Exhibit 1, "Initiating Events Screening Questions," this TS violation screens to Green (very low safety significance) because the boundary leakage did not exceed the capacity of the control rod drive system or TS unidentified leakage (less than 5 gpm) and actual leakage did not adversely impact any LOCA mitigating systems or components.

Because this issue is of very low safety significance and it has been determined that it was not reasonable for Susquehanna staff to be able to foresee and prevent this leakage, and as such no performance deficiency exists, the NRC has decided to exercise enforcement discretion in accordance with Sections 2.2.4 and 3.5 of the NRC Enforcement Policy and refrain from issuing enforcement action for the violation of TS (EA-15-149). Further, because Susquehanna's actions did not contribute to this violation, it will not be considered in the assessment process or the NRC's Action Matrix. The inspectors did not identify any new issues during the review of the LER. This LER is closed.

.6 <u>(Closed) Licensee Event Report (LER) 05000388/2015-004</u>: "Degraded Condition Due to Reactor Coolant Pressure Boundary Leakage Caused by Vibration and Stiff Pipe Connection" on the Unit 2 RRP seal piping to the #2 seal flange weld.

On April 11, 2015, during the initial drywell walk down following shutdown for the spring 2015 Unit 2 refueling outage, Susquehanna staff identified a small leak on the Unit 2 "A" RRP seal piping at a seal flange weld associated with the pressure and vent piping to the upper seal chamber (Connection #2). Water was observed spraying out of the top of the pipe in a fan pattern. Susquehanna staff determined that this leakage constituted a violation of the Unit 2 TS, Section 3.4.4, titled "RCS" that requires RCS leakage be limited to no pressure boundary leakage. Based on the unidentified leakage rate of 0.25 gpm measured during plant operation and visual inspection of the leak area, the leak most likely existed during plant operation.

Susquehanna staff removed the seal flange and piping assembly containing the leak area and provided that material for metallurgical investigation. A qualified replacement assembly was installed with a modified pipe routing to provide for increased pipe flexibility and was inspected prior to and during startup from the Unit 2 refuel outage.

The inspector reviewed LER 2015-004, CR-2015-09907, the apparent cause evaluation completed for this leak, the metallurgical and structural analysis inputs, and CR-2015-009953 regarding the extent of condition evaluation scope. The leakage cause was attributed to the short rigid piping span between the pump and H5005 support with atypical vibration between the two. Using a modified pipe routing to increase pipe flexibility, the seal flange with pipe assembly was replaced. The modified pipe configuration was instrumented and hammer tested to confirm its vibrational frequency characteristics.

LER 2015-004 documents the licensee's violation of TS 3.4.4, limiting reactor pressure boundary leakage during plant operations to zero. This TS violation occurred sometime between Unit 2 start-up in 2013 and plant shutdown on April 10, 2015. The inspector determined that this violation of TS 3.4.4 was more than minor, but not the result of a performance deficiency. In accordance with the NRC Enforcement Policy guidance and IMC 0612, this violation is being treated under the traditional enforcement process and best characterized as a Severity Level IV (very low safety significance) violation, similar to example d.1 in NRC Enforcement Policy, Section 6.1, "Reactor Operations." In addition, using IMC 0609, Significance Determination Process, Appendix A, Exhibit 1, "Initiating Events Screening Questions," this TS violation screens to Green (very low safety significance) because the boundary leakage did not exceed the capacity of the control rod drive system or TS unidentified leakage (less than 5 gpm) and actual leakage did not adversely impact any loss-of-coolant accident mitigating systems or components.

Because this issue is of very low safety significance and it has been determined that it was not reasonable for Susquehanna staff to be able to foresee and prevent this leakage, and as such no performance deficiency exists, the NRC has decided to exercise enforcement discretion in accordance with Sections 2.2.4 and 3.5 of the NRC Enforcement Policy and refrain from issuing enforcement action for the violation of TS (EA-15-189). Further, because Susquehanna's actions did not contribute to this violation, it will not be considered in the assessment process or the NRC's Action Matrix. The inspectors did not identify any new issues during the review of the LER. This LER is closed.

4OA6 Meetings, Including Exit

On October 13, 2015, the inspectors presented the inspection results Mr. J. Franke, Site Vice President, and other members of the Susquehanna's staff. The inspectors verified that no proprietary information was retained by the inspectors or documented in this report.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by Susquehanna and is a violation of NRC requirements that meets the criteria of the NRC Enforcement Policy for being dispositioned as an NCV.

On May 28, 2015, patrols by Susquehanna personnel identified that the panel doors on 2C664, located in the Unit 2 upper relay room, were left open and unattended following troubleshooting by station personnel. Susquehanna's Unit 2 operating license condition 2.C.(6), requires, in part, that Susquehanna implement and maintain in effect all provisions of the approved fire protection program as described in the fire protection review report for the facility. Unit 2 Technical Requirements Manual (TRM) directs compensatory measures as required by the fire protection program for fire suppression system impairments. TRM bases for 3.7.3.4, halon systems, states the opening of any relay room panel door causes the affected Halon system to become inoperable if the panel door is unattended. TRM 3.7.3.4 requires a continuous fire watch with backup fire suppression equipment be established for inoperable halon systems. Contrary to the above, panel doors were left open and unattended without a continuous fire watch for a period of five hours and 35 minutes. Susquehanna entered this issue into the CAP as CR-2015-15709.

The inspectors determined that the finding was more than minor because it was associated with the protection against external factors attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The inspectors determined through a review of IMC 0609.04, "Initial Characterization of Findings," and IMC 0609, Appendix F, "Fire Protection Significance Determination Process," issued September 20, 2013, the finding to be of very low safety significance (Green) based on the reactor maintaining the ability to reach and maintain safe shutdown condition. Specifically, safe shutdown path 3 systems and components would be available for safe shutdown in the event of a fire in the Unit 2 upper relay room.

ATTACHMENT: SUPPLEMENTARY INFORMATION

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

J. Franke, Site Vice President
B. Franssen, Plant Manager
K. Cimorelli, General Manager- Operations
B. O'Rourke, Licensing Engineer
R. Rodrigues-Gilroy, Acting Radiation Protection Manager
E. O'Truba, Radiation Operations Supervisor
J. Barnhardt, Dosimetry Supervisor
N. Cottington, Licensing
M. Dziedzic, Site Level III and IWE/IWL Program Owner
J. Jennings, Regulatory Assurance Manager
D. Kostelnik, Engineering Supervisor
R. Perry, Engineer
D. Przjemski, Engineer
R. Vasquez, Engineer
P. Scanlan, Station Engineer Manager
T. Creasy, Assistant Operations Manager
D. Jones, Operations Manager
H. Strahley, Assistant Operations Manager
S. Muntzenberger, Branch Manager Engineer
N. Grusto, RCIC Engineer
A. May-Allen, EDG Engineer
B. Drysdale, General Manager, Training
A. Kuklis, System Engineer
C. Mangus, Regulatory Assurance
T. Terryah, ISI Programs Manager

LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

Opened/Closed

05000387;388/2015003-01	NCV	RHR Shutdown Cooling Procedure Not Maintained Consistent with Technical Specification Requirements (Section 1R18)
05000387;388/2015003-02	NCV	'C' EDG Rendered Inoperable by Switch Manipulation during Training Simulation (Section 4OA2)

05000387;388/2015003-03	NCV	Secondary Containment Inoperability due to Improperly Controlled Access to the Reactor Building Roof (Section 4OA3)

<u>Closed</u>

05000387;388/2015-002-00	LER	Secondary Containment Inoperability due to Failure to Meet Technical Specification Surveillance Requirement 3.6.4.1.1 (Section 4OA3)
05000387;388/2015-003-00	LER	Secondary Containment Inoperability due to Failure to Meet Technical Specification Surveillance Requirement 3.6.4.1.1 (Section 4OA3)
05000387;388/2015-004-00	LER	Secondary Containment Inoperable due to Secondary Containment Boundary Door Found Ajar (Section 4OA3)
05000387;388/2015-005-00	LER	Loss of Secondary Containment Differential Pressure during Ventilation Damper Testing (Section 4OA3)
05000388/2015-002-00	LER	Secondary Containment Inoperability due to Failure to Meet Technical Specification Surveillance Requirement 3.6.4.1.1 (Section 4OA3)
05000388;387/2015-006-00	LER	Secondary Containment Declared Inoperable due to Secondary Containment Boundary Door 104-R Breached (Section 4OA3)
05000387/2013-009-01	LER	Reactor Protection System Electrical Protection Assembly Logic Card Under Frequency Trip Setpoints Out of Calibration (Section 4OA3)
05000388/2015-005-00	LER	Implementation of Enforcement Guidance Memorandum (Section 4OA3)
05000387/2014-011	LER	Technical Specification Prohibited Condition, Reactor Coolant Pressure Boundary Leakage (Section 4OA3)
05000388/2015-004	LER	Technical Specification Prohibited Condition, Reactor Coolant Pressure Boundary Leakage (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

Procedures
OP-149-001, RHR System, Revision 44

Condition Reports (*NRC identified)
CR-2015-21169*

Drawings
M-2172, Unit 2 Emergency Switchgear Room Cooling, Sheet 1, Revision 22
M-151, Unit 1 P&ID Residual Heat Removal, Sheet 1
M-151, Unit 1 P&ID Residual Heat Removal, Sheet 3

Miscellaneous
TM-OP-034-ST, Secondary Containment, Revision 8
PPL Revision 2, Emergency Switchgear Room Cooling, 3.8.6
TS-Operations, Tag Report 34-001A-ESS SWGR Room Cooler B
SSES, Unit 1, Clearance Order 49-001-1669516-0, August 5, 2015

Section 1R05: Fire Protection

Procedures
FP-013-152, North, Center, and South Cable Chases, Storage Room, Kitchen, Janitor Closet, and Locker Room Fire Zones 0-26A through 0-26E Elevation 729'-0", Revision 5
FP-013-154, Office (C-401), Vestibule (C-404), Fire Zones 0-26F, 0-26G Elevation 729'-1", Revision 5
FP-013-155, Control Room (C-409) & SOFFITS Fire Zones 0-26H, 0-26N, & 0-26P Elevation 729'-1", Revision 7
FP-013-156, TSC and SOFFITS (C-410, 411, 412, 413, 414, 416) Fire Zones 0-26K, 0-26L, 0-26M, 0-26R, Revision 5
FP-013-157, Shift Office (C-402), Vestibule (C-403), Fire Zones 0-26I, 0-26J Elevation 729'-1", Revision 5
FP-013-169, Equipment and Battery Rooms Unit 1 East Side (C-604, 602, 603, 608) Fire Zones 0-28B-I, 0-28M, 0-28N, 0-28J Elevation 771'-0", Revision 4
FP-013-171, Equipment and Battery Rooms Unit 2 East Side (C-613, 609, 614, 615) Fire Zones 0-28A-1, 0-28G, 0-28E, 0-28C Elevation 771'0", Revision 4
FP-213-243, Access Area (II-105) Remote Shutdown Panel Area (II-109) Truck Airlock (II-100) Fire Zones 2-2A, 2-2C Elevation 670'0", Revision 7
FP-213-244, Equipment Access Area (II-102) Fire Zone 2-2B Elevation 670'0", Revision 6
FP-213-250, Switchgear Rooms (II-406, 11-407), Fire Zones 2-4C, 2-4D Elevation 719'-1", Revision 6
FP-213-258, Load Center Room (II-510), Load Center Room (II-507), Fire Zone 2-5F, 2-5G Elevation 749'-1", Revision 5
NDAP-QA-0040, Control of Transient Combustible/Hazardous Materials, Revision 19
NDAP-QA-0449, Fire Protection Program, Revision 12
ON-013-001, Response to Fire, Revision 43
ON-SRV-201, Stuck Open SRV, Revision 0

Condition Reports (*NRC identified)

CR-2015-15709 CR-2015-15793* CR-2015-20451* CR-2015-21168*
CR-2015-24567 CR-67288

Maintenance Orders/Work Orders

1780574 1898416 1900659

Drawings

C-1731, Unit 2 Reactor Building Fire Zone Plan Elevation 719'-1", Sheet 1, Revision 15
C-1731, Unit 2 Reactor Building Fire Doors and Fire Dampers Elevation 719'-1", Sheet 2, Revision 12
C-1731, Unit 2 Reactor Building Fire Protection Plan Elevation 719'-1", Sheet 3, Revision 8
C-1731, Unit 2 Reactor Building Fire Detector Location Plan Elevation 719'-1" to 749'-1", Sheet 4, Revision 10
C-1732, Unit 2 Reactor Building Fire Zone Plan Elevation 749'-1", Sheet 1, Revision 15
C-1732, Unit 2 Reactor Building Fire Doors and Fire Dampers Elevation 749'-1", Sheet 2, Revision 10
C-1732, Unit 2 Reactor Building Fire Protection Plan Elevation 749'-1", Sheet 3, Revision 9
C-1732, Unit 2 Reactor Building Fire Detector Location Plan Elevation 749'-1" to 779'-1", Sheet 4, Revision 8
C-1729, Unit 2 Reactor Building Fire Protection Plan Elevation 670'-0", Sheet 3, Revision 6
C-1729, Unit 2 Reactor Building Fire Doors and Fire Dampers Elevation 670'-0", Sheet 2, Revision 8
C-1729, Unit 2 Reactor Building Fire Zone Plan Elevation 670'-0", Sheet 1, Revision 10
C-1729, Unit 2 Reactor Building Fire Detector Location Plan Elevation 670'-0" to 683'-0", Sheet 4, Revision 5
C-1751, Units 1 & 2, Control Structure Fire Zone Plan Elevation 729'-1", Sheet 1, Revision 9
C-1751, Units 1 & 2, Control Structure Fire Doors and Fire Dampers Elevation 729'-1", Sheet 2, Revision 7
C-1751, Units 1 & 2, Control Structure Fire Protection Plan Elevation 729'-1", Sheet 3, Revision 7
C-1751, Units 1 & 2, Control Structure Fire Detector Location Plan Elevation 729'-1" to 741'-0", Sheet 4, Revision 6
C-1752, Units 1 & 2 Control Structure Fire Zone Plan Elevation 741'-1", Sheet 1, Revision 9
C-1752, Units 1 & 2 Control Structure Fire Doors and Fire Dampers Elevation 741'-1", Sheet 2, Revision 7
C-1752, Units 1 & 2 Control Structure Fire Protection Plan Elevation 741'-1", Sheet 3, Revision 7
C-1752, Units 1 & 2 Control Structure Fire Detector Location Plan Elevation 741'-0" to 754'-0", Sheet 4, Revision 9
C-1753, Units 1 & 2 Control Structure Heat & Ionization Detector, Upper Relay Room, Plan & Elevation 754'-0", Sheet 4A, Revision 0
C-1753, Units 1 & 2 Control Structure Fire Protection Plan Elevation 754'-0", Sheet 3, Revision 4
C-1754, Units 1 & 2 Control Structure Fire Detector Location Plan Elevation 771'-0" to 783'-0", Sheet 4, Revision 5

Miscellaneous

SSES Fire Protection Review Report, Revision 18

Section 1R06: Flood Protection Measures

Procedures
NDAP-QA-1163, Structural Monitoring Program, Revision 3

Condition Reports
CR-2015-26244*

Drawings
E-413, Units 1 & 2 Manholes and Duct Banks, Sheet 1, Revision 36

Section 1R11: Licensed Operator Requalification Program

Procedures
EO-100-102, RPV Control, Revision 8
EO-100-103, Primary Containment Control, Revision 9
EO-100-104, Secondary Containment Control, Revision 9
ON-SCRAM-101, Reactor Scram, Revision 0
OP-AD-002, Standards for Shift Operations, Revision 57
OP-AD-338, Reactivity Manipulations Standards and Communications Requirements, Revision 31
NDAP-QA-0338, Reactivity Management and Controls Program, Revision 24
RE-0TP-103, Guidelines for Planned Power Maneuvers, Revision 13
SR-155-004, Scram Time Measurement of Control Rods, Revision 10

Condition Reports

CR-2015-19828	CR-2015-19830	CR-2015-24907	CR-2015-24910
CR-2015-24928			

Maintenance Orders/Work Orders
1837430

Section 1R12: Maintenance Effectiveness

Procedures
NSEP-AD-0413D, Maintenance Rule-Performance Monitoring, Revision 2
NSEP-AD-0413E, Maintenance Rule-Dispositioning Between (A)(1) and (A)(2), Revision 1
NDAP-QA-0413, Implementation of the Maintenance Rule, Revision 13
SI-161-302, 24 Month Calibration of RWCU System High Differential Flow Channels FDSH-G33-1N603A&B, Revision 25

Condition Reports (*NRC identified)

CR-1164978	CR-1165246	CR-1594943	CR-1600373
CR-1710737	CR-2015-07952	CR-2015-18305	CR-2015-20050*

Action Requests

AR-1619873	AR-2014-26619	AR-2015-18586

Maintenance Orders/Work Orders

1890824	1596487	1912747	1822990
1165736			

Engineering Calculations
EC-RISK-1162, Performance Criteria for EDG E Ability to Substitute, Revision 0

Miscellaneous
Maintenance Rule Basis Document- System 24, 24 Diesel Generators

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures
NDAP-QA-1902, Integrated Risk Management, Revision 20
NDAP-QA-0440, Control of Transient Combustible/Hazardous Materials, Revision 19
OI-013-002, Fire Risk Management, Revision 4
PSP-26, Online and Shutdown Nuclear Risk Assessment Program, Revision 15

Condition Reports (*NRC identified)

CR-2015-21529	CR-2015-21750*	CR-2015-21896	CR-2015-23102
CR-2015-23109	CR-2015-231101	CR-2015-23561	CR-2015-23587

Maintenance Orders/Work Orders
1911131 1652796 1876722

Drawings
C-1807, Refueling Floor Non Outage Laydown Plan Spent Fuel Dry Storage Process Components, Sheet 5, Revision 5

Engineering Calculations
EC-012-6032, Independent Spent Fuel Storage Installation and OEM Steam Dryer Storage Installation- Laydown and Transport Analysis, Revision 7
EC-012-6038, NUHOMS Transfer Cask/ Dry Shielded Canister Tip Over Analysis, Revision 2

Miscellaneous
Protected Equipment Clearance Order 00-001A-OI-013-002 ATT D, August 3, 2015
Protected Equipment Clearance Order 49-001-UI B RHR PUMP, August 3, 2015
Protected Equipment Clearance Order 06-001-2B229 Swing Bus Alt, August 3, 2015
Protected Equipment Clearance Order 49-002-U2 RHR DIV 1, August 6, 2015
Protected Equipment Clearance Order 49-001-U2 B RHR PUMP, August 3, 2015
Protected Equipment Clearance Order 54-001 Division 1, ESW August 17, 2015
Protected Equipment Clearance Order 49-001 Unit 1 RHR Division 1, August 17, 2015
Protected Equipment Clearance Order 49-001 Unit 2 RHR Division 1, August 17, 2015
Protected Equipment Clearance Order 16-001 Division 1 RHRSW, August 17, 2015

Section 1R15: Operability Determinations and Functionality Assessments

Procedures
OI-AD-096, Operator Burdens, Revision 13
ON-100-009, Control Room Evacuation, Revision 36
SE-234-006, 24 Month ESGR Cooler B Chiller Performance Test, Revision 4
SO-249-001, RHR Monthly Alignment Check, Revision 26
SO-251-001A, Monthly Core Spray A Loop Discharge Line Filled & Valve Alignment Verification, Revision 1
SO-251-001B, Monthly Core Spray B Loop Discharge Line Filled & Valve Alignment Verification, Revision 1

Condition Reports

CR-2014-28100	CR-2015-02969	CR-2015-13489	CR-2015-17509
CR-2015-18004	CR-2015-18016	CR-2015-18108	CR-2015-18123
CR-2015-18305	CR-2015-18586	CR-2015-19812	CR-2015-20775
CR-2015-21858	CR-2015-21627	CR-2015-21627*	CR-2015-21169
CR-2015-22197	CR-2015-22586	CR-2015-24137	CR-474309

Action Requests

AR-1656682	AR-2014-01791	AR-2014-32778	AR-2014-38293
AR-2015-14502	AR-2015-18045	AR-2015-21369	AR-2015-21656
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C-1815, Units 1 & 2 Reactor Building Elevation 719'-0" Equipment Qualification Harsh Environment Zones, Sheet 8, Revision 11
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LIST OF ACRONYMS

ADAMS	Agencywide Documents Access and Management System
BWR	boiling-water reactor
CAP	corrective action program
CFR	Code of Federal Regulations
CR	condition report
CRD	control rod drive
ECCS	emergency core cooling systems
EDG	emergency diesel generator
EGM	enforcement guidance memorandum
EPA	electrical protection assembly
ESW	emergency service water
HCU	hydraulic control unit
HRA	high radiation area
IMC	Inspection Manual chapter
JPM	job performance measure
LAR	license amendment request
LCO	limiting condition for operation
LPRM	local power range monitor
NCV	non-cited violation
NRC	Nuclear Regulatory Commission
NVLAP	National Voluntary Laboratory Accreditation Program
ODCM	offsite dose calculation manual
OPDRV	operations with the potential to drain the reactor vessel
OWA	operator workaround
PI	performance indicator
RCIC	reactor core isolation cooling
RCS	reactor coolant system
REMP	radiological environmental monitoring program
RG	regulatory guide
RHR	residual heat removal
RHRSW	residual heat removal service water
RPM	radiation protection manager
RPS	reactor protection system
RPV	reactor pressure vessel
RRP	reactor recirculation pump
RWCU	reactor water cleanup
RWP	radiation work permit
SBGT	standby gas treatment
SCBA	self-contained breathing apparatus
SDC	shutdown cooling

SLC	standby liquid control
SSES	Susquehanna Steam Electric Station
TRM	Technical Requirements Manual
TS	technical specification
UFSAR	Updated Final Safety Analysis Report
VHRA	very high radiation area