UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
245 PEACHTREE CENTER AVENUE NE, SUITE 1200
ATLANTA, GEORGIA 30303-1257

July 28, 2016

Mr. Joseph W. Shea
Vice President, Nuclear Licensing
Tennessee Valley Authority
1101 Market Street, LP 3D-C
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT – NRC DESIGN BASES INSPECTION (TEAM) REPORT 05000327/2016007 AND 05000328/2016007

Dear Mr. Shea:

On, June 17, 2016, the U. S. Nuclear Regulatory Commission (NRC) completed an inspection at Sequoyah Nuclear Plant, Units 1 and 2, and discussed the results of this inspection with Mr. Schwarz and other members of your staff. Additional inspection results were discussed with Mr. Schwarz and other members of your staff on July 26, 2016. Inspectors documented the results of this inspection in the enclosed inspection report.

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

If you contest the violations or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Sequoyah Nuclear Plant.

If you disagree with a cross-cutting aspect assignment or a finding not associated with a regulatory requirement in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region II; and the NRC Resident Inspector at the Sequoyah Nuclear Plant.

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

Sincerely,

/RA/

Jonathan H. Bartley, Chief
Engineering Branch 1
Division of Reactor Safety

Docket Nos. 05000327, 05000328
License Nos. DPR-77, DPR-79

Enclosure: Inspection Report 05000327/2016007
and 05000328/2016007 w/Attachment:
Supplementary Information

cc: Distribution via ListServ

accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

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ADAMS: ☒ Yes ACCESSION NUMBER:_______________ ☒ SUNSI REVIEW COMPLETE ☒ FORM 665 ATTACHED

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SIGNATURE	**RNP1**	**GRO via Email**	**MAR1 via Email**	**BDB3 via Email**	**MXY4 via Email**	**OSM via Email**	**JHB1**
NAME	R. Patterson	G. Ottenberg	M. Riley	B. Bishop	M. Yeminy	O. Mazzoni	J. Bartley
DATE	7 / 27 / 16	7 / 25 / 16	7 / 23 / 16	7 / 25 / 16	7 / 25 / 16	7 / 24 / 16	7 /28 / 16
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-327, 50-328
License Nos.:	DPR-77, DPR-79
Report Nos.:	05000327/2016007, 05000328/2016007
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Sequoyah Nuclear Plant, Units 1 and 2
Location:	Soddy-Daisy, TN 37379
Dates:	May 9, 2016, to June 17, 2016
Inspectors:	R. Patterson, Reactor Inspector (Team Leader) M. Riley, Reactor Inspector G. Ottenberg, Sr. Reactor Inspector B. Bishop, Project Engineer O. Mazzoni, Contractor M. Yeminy, Contractor
Approved by:	Jonathan H. Bartley, Chief Engineering Branch 1 Division of Reactor Safety

SUMMARY

IR 05000327/2016-007 and 05000328/2016-007; 05/09/2016 – 06/17/2016; Sequoyah Nuclear Plant, Units 1 and 2; Design Bases Inspection (Team).

This inspection was conducted by a team of four Nuclear Regulatory Commission (NRC) inspectors from Region II and two NRC contract personnel. The significance of inspection findings are indicated by their color (i.e., greater than Green, or Green, White, Yellow, or Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," (SDP) dated April 29, 2015. Cross-cutting aspects are determined using IMC 0310, "Aspects Within the Cross-Cutting Areas," dated December 4, 2014. All violations of NRC requirements were 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.

<u>NRC-Identified and Self-Revealing Findings</u>

Cornerstone: Mitigating Systems

- <u>Green</u>: The NRC identified a non-cited violation of Title 10 Code of Federal Regulations (CFR) Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," for the licensee's failure to use the design change process to make modifications to the Emergency Diesel Generator EDG room inlet dampers as required by NPG-SPP-9.3, "Plant Modifications and Engineering Change Control." The licensee entered the issue into the corrective action program and implemented compensatory measures, while implementing plans to modify each of the affected inlet and exhaust fire dampers.

  This performance deficiency was determined to be more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee modified the dampers to include the wrong brackets, which could adversely affect the dampers ability to remain open to provide cooling during EDG operation and support EDG reliability and availability. The team determined the finding to be of very low safety significance (Green) because the finding was a deficiency affecting the design of a mitigating structure, system, or component (SSC), and the SSC maintained its operability or functionality. This finding was not assigned a cross-cutting aspect because the issue did not reflect current licensee performance. (Section 1R21.2.b.01)

  <u>Green</u>: The NRC identified a non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to install emergency diesel generator components that could withstand the effects of a design basis tornado as required by Section 3.1.2 of the Update Final Safety Analysis Report (UFSAR). The licensee entered the issue into the corrective action program and implemented compensatory measures to protect the affected components.

  This performance deficiency was determined to be more than minor because it was associated with the Design Control attribute of the Mitigating Systems cornerstone and adversely affects the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable

consequences. Specifically, the capability of the equipment to withstand the effects of a tornado was not ensured. The team determined the finding to be of very low safety significance (Green) because of the low frequency of tornados/high winds and the potential for recovery by the operators on site. This finding was not assigned a cross-cutting aspect because the issue did not reflect present licensee performance. (Section 1R21.2.b.02)

<u>Green</u>: The NRC identified a non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," for the licensee's failure to have documented procedures in place to ensure effective monitoring of the 480V Shutdown Transformers as required by Section 5.3.2.(4) of IEEE 308-1971. The licensee entered the issue into the corrective action program and planned to put additional transformer testing/monitoring in place to detect degradation prior to equipment failure.

This performance deficiency was determined to be more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to perform adequate maintenance on the shutdown transformer, which could result in the inability to detect the deterioration of the shutdown transformer toward an unacceptable condition. The team determined the finding to be of very low safety significance (Green) because the finding was a deficiency affecting the design of a mitigating structure, system, or component (SSC), and the SSC maintained its operability or functionality. This finding was not assigned a cross-cutting aspect because the issue did not reflect present licensee performance. (Section 1R21.2.b.03)

Cornerstone: Barrier Integrity

- <u>Green</u>: The NRC identified a finding (FIN) for the licensee's failure to meet their docketed commitment to revise the back-up generators to include supplying one train of containment hydrogen igniters per unit in response to Generic Safety Issue 189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident." The licensee entered this issue into their corrective action program and completed immediate corrective actions to revise procedure FSI-5.01, "Initial Assessment and Flex Equipment Deployment," Rev. 0, to ensure the hydrogen igniters would be energized during an extended station blackout (SBO) event.

  The performance deficiency was determined to be more than minor because it was associated with the Procedure Quality attribute of the Barrier Integrity Cornerstone and adversely affected the cornerstone objective of providing reasonable assurance that physical barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to energize the hydrogen igniters during an extended SBO event could result in containment failure. The team determined the finding to be of very low safety significance (Green) because the risk was mitigated by the low frequency of SBO conditions and the high likelihood of operator recovery given the obvious diagnosis of the performance deficiency. The team determined the finding was indicative of present licensee performance and was associated with the cross-cutting aspect of operating experience (OE), in the area of Problem Identification and

Resolution, because the licensee failed to effectively collect, evaluate, and implement relevant internal OE before implementing their new station procedures to use the FLEX diesels as the power supply to the hydrogen igniters. [P.5] (Section 1R21.2.b.04)

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R21 Design Bases Inspection (Team) (71111.21)

.1 Inspection Sample Selection Process

The team selected risk-significant components and related operator actions for review using information contained in the licensee's probabilistic risk assessment. In general, this included risk significant structures, systems, and components (SSCs) that had a risk achievement worth factor greater than 1.3 or Birnbaum value greater than 1E-6. The sample included 13 SSCs, 1 SSC associated with containment large early release frequency (LERF), and 4 operating experience (OE) items.

The team performed a margin assessment and a detailed review of the selected risk-significant components and associated operator actions to verify that the design bases had been correctly implemented and maintained. Where possible, this margin was determined by the review of the design basis and Updated Final Safety Analysis Report (UFSAR). This margin assessment also considered original design issues, margin reductions due to modifications, or margin reductions identified as a result of material condition issues. Equipment reliability issues were also considered in the selection of components for a detailed review. These reliability issues included items related to failed performance test results, significant corrective action, repeated maintenance, maintenance rule status, Inspection Manual Chapter 0326 conditions, NRC Resident Inspector input regarding problem equipment, system health reports, industry OE, and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, OE, and the available defense-in-depth margins. An overall summary of the reviews performed and the specific inspection findings identified is included in the following sections of the report.

.2 Component Reviews

a. Inspection Scope

SSCs

- Unit 1 Containment Sump Valve (1-FCV-63-072)
- Unit 1 / Unit 2 EDG Ventilation System
- Unit 1 / Unit 2 Essential Raw Cooling Water (ERCW) Pump (M-B)
- Unit 2 Residual Heat Removal (RHR) Pump 2A-A
- Unit 2 RHR Suction Isolation Valve (2-FCV-74-003)
- Unit 2 Turbine Driven Auxiliary Feedwater (TDAFW) Pump
- Unit 2 2A Emergency Diesel Generator (EDG)
- Unit 2 2A-A EDG Close Breaker
- Unit 2 EDG 2A/2B to ERCW Pump CABLING
- Unit 2 480 Shutdown Board (2A1-A)
- Unit 2 6.9kV Shutdown Transformer (2A1-A)
- Unit 2 Vital Battery Board-1/BRK 321
- Unit 2 TDAFW Turbine Normal Feeder Breaker (1-BKRC-3-KG/321-A)

Components with LERF Implications

- Unit 1 / Unit 2 Hydrogen Igniters (LERF)

For the 14 components listed above, the team reviewed the plant technical specifications (TS), UFSAR, design bases documents, and drawings to establish an overall understanding of the design bases of the components. Design calculations and procedures were reviewed to verify that the design and licensing bases had been appropriately translated into these documents and that the most limiting parameters and equipment line-ups were used. Logic and wiring diagrams were also reviewed to verify that operation of electrical components conformed to design requirements. Test procedures and recent test results were reviewed against design bases documents to verify the adequacy of test methods and that acceptance criteria for tested parameters were supported by calculations or other engineering documents, and that individual tests and analyses served to validate component operation under accident conditions. Maintenance procedures were reviewed to ensure components were appropriately included in the licensee's preventive maintenance program. System modifications, vendor documentation, system health reports, preventive and corrective maintenance history, and corrective action program documents were reviewed (as applicable) in order to verify that the performance capability of the component was not negatively impacted, and that potential degradation was monitored or prevented. Maintenance Rule information was reviewed to verify that the component was properly scoped, and that appropriate preventive maintenance was being performed to justify current Maintenance Rule status. Component walk-downs and interviews were conducted to verify that the installed configurations would support their design and licensing bases functions under accident conditions, and had been maintained to be consistent with design assumptions.

Additionally, the team performed the following specific reviews:

- The team reviewed the following time critical operator actions:
    1) Isolating a Steam Generator Tube Rupture within 40 mins
    2) Stopping Reactor Coolant Pumps during a small break Loss of Coolant Accident (LOCA) within 5 min
    3) Cooling and restoring one train of RHR following a Mode 4 LOCA
- The team conducted a walk-down of the 6.9kv and 480V FLEX Diesel Generators to verify the observable condition of the back-up power supplies to the hydrogen igniters.
- The team reviewed the licensee's thermal overload setting determination and methodology to verify the setpoints for the thermal overloads associated with the SQN-1-FCV-63-72 and the SQN-2-FCV-74-3 motor operated valves were established in favor of completion of the safety related functions of the equipment in accordance with NRC Regulatory Guide 1.106.
- The team observed a steam generator tube rupture scenario in the simulator. The team verified that operator actions could be accomplished within the required time and that the actions could be accomplished in accordance with approved licensee procedures.

- The team performed table-top reviews of applicable Abnormal Operating and Emergency Operating Procedures with a licensed senior reactor operator to better understand actions to be taken to:
    1) mitigate the effects of a steam generator tube leak which escalates to a rupture
    2) cooldown and depressurize the reactor coolant system in order to minimize break flow
    3) isolate the affected steam generator within the required time
- The team conducted in-field walk-downs of plant procedures to verify the actions could be accomplished within the assumed timeframe and that there was sufficient guidance in the procedures to properly complete the tasks.

b. Findings

.01 Failure to Implement the Design Change Process when Modifying the Safety-Related Fire Dampers

Introduction: The NRC identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures and Drawings,” for the licensee’s failure to implement the design change process prescribed by plant procedures.

Description: In 2008, SQN identified a trend of spurious, partial closures of the EDG room inlet fire dampers due to pressure surges at the start of two diesel building exhaust fans. SQN modified the EDG rooms’ inlet fire dampers, applying “jumper brackets” to connect the four quadrants of each damper to address this condition. These design changes were made by revising the vendor manual for the fire dampers, thereby circumventing the design change process. Procedure NPG-SPP-9.3, “Plant Modifications and Engineering Change Control,” Revision 16, Appendix K, specified that for structures, systems, or components (SSCs) under configuration control, changes that are not temporary, not replacement parts, with a required physical change, and which meet the definition of a design change, the use of a DCN was required.

As a result of modifying the inlet fire dampers outside of the design change process, the level of review and understanding of potential impacts to the plant was not performed and the licensee installed brackets that were not qualified for the application. Furthermore, the vendor manual change process did not require the licensee to evaluate the need to modify the exhaust dampers, which incorporate the identical design. As a result, the exhaust fire dampers experienced similar failures later. The licensee entered the issue into the corrective action program (CAP) as condition reports 1177144 and 1180402 and implemented compensatory measures, while implementing plans to modify each of the affected inlet and exhaust fire dampers.

Analysis: The licensee’s failure to use a DCN to make modifications to the EDG room inlet dampers as required by NPG-SPP-9.3, “Plant Modifications and Engineering Change Control,” was a performance deficiency. This performance deficiency was determined to be more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee modified the dampers to include the wrong brackets, which could adversely affect the dampers ability to remain open to provide cooling during EDG operation and support EDG reliability and availability.

The team screened this finding in accordance with IMC 0609, “Significance Determination Process for findings at power,” issued June 19, 2012, Attachment 4, “Phase 1 - Initial Screening and Characterization of Findings,” and determined the finding to be of very low safety significance (Green) because the finding was a deficiency affecting the design of a mitigating SSC, and the SSC maintained its operability or functionality. This finding was not assigned a cross-cutting aspect because the modifications made to the inlet fire dampers were made in 2008 and issue did not reflect present licensee performance.

Enforcement: Title 10 of CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” required in part, that “activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.” Contrary to the above, the modification of the safety-related inlet fire dampers (activity effecting quality), was not accomplished in accordance with procedure NPG-SPP-09.3 when they were installed in 2008. This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee’s CAP as condition report (CR) 1177144 and CR 1180402. (NCV 05000327, 328/2016007-01, “Failure to Implement the Design Change Process when Modifying the Safety-Related Fire Dampers.”)

.02 Failure to Install Safety-Related Components that are Designed to Withstand the Effects of a Design Basis Tornado

Introduction: The NRC identified a Green NCV of 10 CFR 50, Appendix B, Criterion III, “Design Control,” for the licensee’s failure to install components important to safety that can withstand the effects of a design basis tornado.

Description: The Sequoyah UFSAR, Section 3.3.2.1, stated, in part, that “the design tornado is characterized by an "eye" with reduced atmospheric pressure of 3 lb/in^2 below ambient and by a "funnel" having a maximum rotational velocity of 300 mph.” A tornado could cause a differential pressure across the east and west sides of the Diesel Generator Building, thereby creating a large rate of air flow across the inlet and outlet openings of the ventilation system. The intake and exhaust fire dampers at each EDG bay, which are required to isolate the space for CO2 fire suppression, have not been analyzed to withstand the high rate of air flows and the turbulence resulting from a passing tornado, and therefore could fail closed and block the required airflow needed for EDG room cooling. The following ventilation system components are also not rated nor evaluated for the effects of a design basis tornado:

- EDG exhaust fans
- EDG exhaust dampers for each EDG Bay exhaust fan
- EDG exhaust fan ductwork and fan housing.

Additionally, the EDG crankcase over-pressure trip device could lock out the EDG start logic upon depressurization by a tornado and maintain the start logic locked out during an emergency start because it is designed to actuate at 1 inch W.G. Damage to these safety-related ventilation system components could either lock up the start logic of each

EDG or stop the cooling air to the EDGs and to the electrical equipment cabinets located in the room.

Section 3.1.2 of the UFSAR, stated in part, that, “all structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena, including tornadoes.” In addition, on December 6, 2006, the NRC published Regulatory Issue Summary 2006-23, “Post-Tornado Operability Of Ventilating And Air-Conditioning Systems Housed In Emergency Diesel Generator Rooms,” which stated, in part, that “the NRC expects licensees to consider natural hazards during the design of systems and components housed inside safety-related structures if these systems and components may be exposed to the outside environment and if their malfunction or loss may prevent or impact the operability of safety-related systems and components.

In response to the team’s questioning, the licensee prepared a prompt determination of operability, and implemented the following compensatory measures:

(1) SQN manufactured and staged 32 blocking devices (four per fire damper) to be attached to the grid located in each EDG room above the inlet and outlet fire dampers, protruding into each fire damper, thereby preventing its undesirable closure. These blocking devices are to be installed at a time of a tornado watch in accordance with a specially prepared procedure.
(2) Open an access hatch at the outlet side of the exhaust fan in each EDG room upon a notification of a tornado watch to minimize the pressure differential across the duct and fan exhaust damper.
(3) Start all the EDGs upon a notification of a tornado warning or whenever a tornado is seen in the vicinity of the plant.

<u>Analysis</u>: The licensee’s failure to install components important to safety that can withstand the effects of a design basis tornado as required by Section 3.1.2 of the UFSAR, was a performance deficiency. This performance deficiency was determined to be more than minor because it is associated with the Design Control attribute of the Mitigating Systems cornerstone and adversely affects the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the capability of the EDGs to withstand the effects of a tornado was not ensured.

This finding was assessed using IMC 0609, Attachment 4, “Initial Characterization of Findings,” issued June 19, 2012, for the Mitigating Systems cornerstone. In accordance with IMC 0609, Appendix A, “The Significance Determination Process for Findings At-Power,” issued June 19, 2012, this finding was evaluated using IMC 0609, Appendix A, Exhibit 2, section B, “External Event Mitigation Systems,” since the finding involved a loss or degradation of equipment or function specifically designed to mitigate a severe weather initiating event. In accordance with IMC 0609, Appendix A, Exhibit 4, it was determined that a detailed risk evaluation was warranted because the loss of this equipment or function by itself during the external initiating event it was intended to mitigate: would degrade two or more trains of a multi-train system or function with an exposure time of greater than 30 days. A regional senior reactor analyst performed a detailed risk evaluation and determined the finding to be of very low safety significance (Green). The dominant accident sequence in which the procedure would be used was a weather-related non-recoverable loss of offsite power that simultaneously affects all four of the EDGs on site. The risk of this performance deficiency was low due to the low frequency of tornados/high winds and the potential for recovery by the operators on site.

This finding was not assigned a cross-cutting aspect because the issue did not reflect current licensee performance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," states, in part, that, "Measures shall also be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components." Contrary to the above, since original design, the licensee did not review for suitability the application of equipment essential to the safety-related function of the EDGs. Specifically, ventilation system dampers, fans, and ductwork as well as the effect of non-emergency trips were not reviewed or analyzed for their suitability to withstand the effects of a design basis tornado. The licensee implemented compensatory measures, which included staging damper blocking devices, starting all four EDGs upon receipt of a tornado warning, and establishing alternate airflow paths within the EDG rooms. This violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. This violation was entered into the licensee's CAP as CRs 1181710 and 1182240. (NCV 05000327, 328/2016007-02, "Failure to Install Safety-Related Components that are Designed to Withstand the Effects of a Design Basis Tornado.")

.03 Inadequate Monitoring of the 480V Shutdown Transformers

Introduction: The NRC identified a Green NCV of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to effectively monitor the 480V Shutdown Transformers in accordance with written procedures appropriate to the circumstances.

Description: At Sequoyah Nuclear Plant, each of the two 480V Class 1E switchgear serving a power train has an alternate supply from its parent 6.9kV Class 1E switchgear. Each supply is through a separate 6.9kV to 480V transformer. The function of the 480V Shutdown Transformers is to stepdown the voltage fed from either offsite or emergency power from 6.9kV to 480V, then provide power to the associated safety-related loads during both normal and emergency operation. To ensure that each Class 1E 480V Shutdown Transformer is maintained ready to perform its safety function, each transformer is required to be periodically monitored for degradation.

In UFSAR, Section 8.1.5.1, the licensee committed to IEEE 308-1971, "IEEE Standard Criteria for Class IE Electric Systems for Nuclear Power Generating Stations." Standard IEEE 308-1971, section 5.3.2.(4), required, in part, "that the distribution system shall be monitored to the extent that it is shown to be ready to perform its intended function." In addition, IEEE 308-1971, section 6.3, required, in part, that "tests shall be performed at scheduled intervals to detect the deterioration of the system toward an unacceptable condition."

Prior to 2007, the licensee met this standard by performing the vendor recommended oil testing to demonstrate transformer readiness and detect degradation. The licensee later discontinued the vendor recommended oil testing due to difficulty finding vendors to test oil containing polychlorinated biphenyls (PCBs). Following the discontinuation of oil testing, the licensee relied on adequacy of output voltage of the transformer, thermography scans, and operator's round verification of pressure and oil temperature as sole indicators of the unit's condition.

The team reviewed the following procedures used by the licensee to meet IEEE 308-1971, and found them to be inadequate to detect transformer degradation because of the following:

- 0-GO-14-5, AUO Operator Rounds, rev. 43
    - Procedure lacked acceptance criteria and did not require temperature and pressure readings to be recorded or evaluated
    - Gauges were not frequently calibrated
- 0-TI-PDM-000-003.0, Infrared Thermography Inspections, rev. 2
    - Due to the design of transformer, the scans were limited to the outer shell of the transformer and would not provide sufficient insight on the condition of the transformer internals
- 2SI-OPS-082-026.A DG2A, Loss of Offsite PWR with SI, dated 11/16/2015
    - Limited to confirming the transformers could supply the necessary voltages to the safety buses but did not detect degradation toward an unacceptable condition

As a result of the team's questions, the licensee planned to put additional transformer testing/monitoring in place to comply with IEEE-308.

Analysis: The licensee's failure to have documented procedures in place to ensure effective monitoring of the 480V Shutdown Transformers as required by Section 5.3.2.(4) of IEEE 308-1971, was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to monitor the shutdown transformers could result in the undetected deterioration of the shutdown transformer to an unacceptable condition. The team used IMC 0609, Att. 4, "Initial Characterization of Findings," issued June 19, 2012, for Mitigating Systems, and IMC 0609, App. A, "The Significance Determination Process (SDP) for Findings At-Power," issued June 19, 2012, and determined the finding to be of very low safety significance (Green) because the finding was a deficiency affecting the design of a mitigating SSC, and the SSC maintained its operability or functionality. This finding was not assigned a cross-cutting aspect because the issue did not reflect present licensee performance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," required, in part, that "activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings." IEEE 308-1971, section 5.3.2.(4) required, in part, "that the distribution system shall be monitored to the extent that it is shown to be ready to perform its intended function." In addition, IEEE 308-1971, section 6.3, required, in part, that "tests shall be performed at scheduled intervals to detect the deterioration of the system toward an unacceptable condition." Contrary to the above requirements, since 2008, an activity affecting quality, monitoring the 480V Shutdown Transformers for deterioration, was not prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances. The licensee did not have adequate procedures to monitor the 480V shutdown transformers to detect deterioration as required by IEEE 308-1971. The licensee planned to put additional transformer testing/ monitoring in place to comply with IEEE 308. This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee's CAP as CR 1182472. (NCV 05000327, 328/2016007-03, "Inadequate Monitoring of the 480V Shutdown Transformers."

.04 Failure to Energize Hydrogen Igniters during Extended Station Blackout

Introduction: The NRC identified a Green Finding (FIN) for the licensee's failure to meet their docketed commitment to revise the back-up generators to include supplying one train of containment hydrogen igniters per unit in response to Generic Safety Issue (GSI) 189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident."

Description: The NRC opened GSI-189 to document that, following a severe core damage event associated with station blackout (SBO) or other events affecting AC power distribution, pressurized water reactor (PWR) ice condenser containments, such as those at Sequoyah Nuclear Plant, had the potential to fail as a result of large hydrogen detonations. Sequoyah Nuclear Plant uses hydrogen igniters, which induce periodic burns at moderate energy addition rates to mitigate the effects of hydrogen releases in containment and to mitigate the potential failure of containment due to large hydrogen concentrations. To address GSI-189, the licensee made a commitment in a docketed letter (ADAMS Accession No. ML070670149) to the NRC, dated March 6, 2007, to revise the back-up generator procedures to include supplying one train of containment hydrogen igniters per unit, and to train personnel to the procedure revision, by December 31, 2007.

During a loss of all AC power, the licensee enters procedure ECA-0.0, "Loss of All AC Power," Rev. 28. This procedure stated that if offsite power or EDG power could not be restored within four hours (i.e., Extended Station Blackout), enter procedures FSI-4, "DC Bus Management and 480V FLEX D/G Alignment/Loading," Rev.1, and FSI-5.01, "Initial Assessment and Flex Equipment Deployment," Rev. 0, to tie on the 480V and 6.9kv FLEX diesels, respectively. The team noted that procedure FSI-4 did not provide steps to energize the hydrogen igniters and that procedure FSI-5.01 provided steps to open the breakers which fed the hydrogen igniters, but did not include steps to close the breakers to ensure the hydrogen igniters would be energized when needed. The team also noted that there was no procedural guidance provided which would inform the operators on what actions to complete if the hydrogen igniters were not energized. The team determined that the failure of both procedures to ensure the hydrogen igniters would be energized during an extended SBO event did not meet the licensee's commitment, which was made to address GSI-189 and would adversely affect the licensee's ability to prevent containment failure during such an event.

The team noted that this issue was similar to a finding the licensee received in 2009 for failing to ensure procedure 0-MA-REM-000-001.0, "Extended Station Blackout," Rev. 3, energized the hydrogen igniters. The licensee entered that issue into their CAP and resolved the lack of procedural guidance in that procedure. However, on December 7, 2015, the licensee implemented a new strategy to rely on the FLEX diesels to provide the backup power to the hydrogen igniters and reintroduced the deficiency from 2009 into procedures FSI-4 and FSI-5.01. The licensee entered this issue into their CAP as CRs 1170662, 1172319, and 1174214, and completed immediate corrective actions to revise procedure FSI-5.01 to ensure the hydrogen igniters would be energized during an extended SBO. The licensee also determined that it would have been likely that an operator would have determined that the hydrogen igniters were not energized and would have taken the appropriate actions to ensure they were energized during an extended SBO event.

Analysis: The licensee's failure to meet their docketed commitment to revise the back-up generators to include supplying one train of containment hydrogen igniters per unit in response to GSI-189 was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Procedure Quality attribute of the Barrier Integrity Cornerstone and adversely affected the cornerstone objective of providing reasonable assurance that physical barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to energize the hydrogen igniters during an extended SBO event could result in containment failure.

This finding was assessed using IMC 0609, Attachment 4, "Initial Characterization of Findings," issued June 19, 2012, for the Barriers Cornerstone. In accordance with IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power," issued June 19, 2012, this finding was evaluated using IMC 0609, Appendix H, "Containment Integrity Significance Determination Process," issued May 6, 2004, since it involved an actual reduction in function of the hydrogen igniters in the reactor containment. The deficiency was a Type B finding since it was related to a degraded condition that had potentially important implications for the integrity of the containment, without affecting the likelihood of core damage. Since the performance deficiency affected the igniters of a PWR with an ice condenser containment, Table 6.1 of IMC 0609, Appendix H, "Phase 1 Screening – Type B Findings at Full Power," indicated that a Phase 2 was warranted. A Phase 2 screening in accordance with Table 6.2 of IMC 0609, Appendix H, "Phase 2 Risk Significance – Type B Findings at Full Power," indicated that a failure of multiple igniters such that coverage is lost in two adjacent compartments for a PWR Ice Condenser containment with an exposure time of greater than 30 days, was greater than Green and warranted a detailed risk evaluation. A regional senior reactor analyst performed a detailed risk evaluation in accordance with NRC IMC 0609 Appendix A using the NRC Sequoyah SPAR model. The performance deficiency was modeled as a non-recoverable station blackout requiring use of FLEX equipment. The major analysis assumptions included an exposure period of 165 days and operator recovery credit to close the hydrogen igniter breakers. The dominant sequence was extended station blackout leading to core damage and failure of the operator to energize the igniters leading to containment failure and large early release. The risk was mitigated by the low frequency of SBO conditions and the high likelihood of operator recovery given the obvious diagnosis of the performance deficiency. The detailed risk evaluation determined the finding to be of very low safety significance (Green) for delta CDF and LERF.

Since this issue was introduced on December 7, 2015, the team determined the finding was indicative of present licensee performance and was associated with the cross-cutting aspect of operating experience (OE), in the area of Problem Identification and Resolution, because the licensee failed to effectively collect, evaluate, and implement relevant internal OE before implementing their new station procedures to use the FLEX diesels as the power supply to the hydrogen igniters. Specifically, the licensee failed to evaluate a similar NRC-identified finding issued in 2009 for applicability to their new procedures. [P.5]

Enforcement: The team did not identify a violation of regulatory requirements associated with this finding. Because this finding does not involve a violation and is of very low safety significance, it is identified as a FIN. (FIN 05000327, 328/2016007-04, “Failure to Energize Hydrogen Igniters during Extended Station Blackout”)

.3 Operating Experience

a. Inspection Scope

The team reviewed four operating experience issues for applicability at the Sequoyah Nuclear Plant. The team performed an independent review for these issues and, where applicable, assessed the licensee’s evaluation and disposition of each item. The issues that received a detailed review by the team included:

- NRC Information Notice 2012-11, “Age Related Capacitor Degradation”
- NRC Generic Letter 2007-01, “Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients.”
- Part 21 report 2013-09-00 (ADAMS Accession No. ML13064A012), Flowserve Part 21 dated 2/25/13, “Subject: Wedge Pin Failure of an Anchor/Darling Double- Disc Gate Valve at Browns Ferry Nuclear Plant Unit 1”
- NRC Information Notice No. 90-61: “Potential for Residual Heat Removal Pump Damage Caused by Parallel Pump Interaction”

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA6 Meetings, Including Exit

On June 17, 2016, the team presented the inspection results to Mr. Christopher Schwarz and other members of the licensee’s staff. On July 26, 2016, a telephone re-exit was conducted to present the final inspection results to Mr. Schwarz and other members of the licensee’s staff. Proprietary information that was reviewed during the inspection was returned to the licensee or destroyed in accordance with prescribed controls.

ATTACHMENT: SUPPLEMENTARY INFORMATION

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:
C. Schwarz, SQN Vice President
P. Pratt, SQN Plant Manager
W. Pierce, Director of Engineering
K. Smith, Director of Training
T. Marshall, Director of Operations
M. Rasmussen, Director of Maintenance
M. McBrearty, Licensing Manager
R. Travis, Licensing Engineer
J. Campbell, Electrical I&C Principal Engineer
B. Thombs, Electrical I&C Design Manager
C. Johnson, Sr. Manager Projects (acting)
C. Reneau, Sr. Manager Design Engineering
J. Hodge, Sr. Manager Systems Engineering
B. Simril, Sr. Manager Corporate Design Engineering
P. Wilson, Corporate Licensing
D. Porter, Operation Procedures Senior Reactor Operator
E. Craig, Design Engineering
J. Nicely, MOV Program Manager

NRC personnel:
J. Bartley, Chief, Engineering Branch 1, Division of Reactor Safety, Region II
R. Bernhard, Senior Reactor Analyst, Division of Reactor Projects, Region II
G. Smith, Senior Resident Inspector, Division of Reactor Projects, Sequoyah Resident Office
W. Deschaine, Resident Inspector, Division of Reactor Projects, Sequoyah Resident Office

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened & Closed

05000327, 328/2016007-01	NCV	Failure to Implement the Design Change Process when Modifying the Safety-Related Fire Dampers (Section 1R21.2.b.01)
05000327, 328/2016007-02	NCV	Failure to Install Safety-Related Components that are Designed to Withstand the Effects of a Design Basis Tornado (Section 1R21.2.b.02)
05000327, 328/2016007-03	NCV	Inadequate Monitoring of the 480V Shutdown Transformers (Section 1R21.2.b.03)
05000327, 328/2016007-04	FIN	Failure to Energize Hydrogen Igniters during Extended Station Blackout (Section 1R21.2.b.04)

LIST OF DOCUMENTS REVIEWED

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0-MA-REM-000-001.0, Extended Station Blackout, Rev. 3
0-MA-REM-000-001.0, Extended Station Blackout, Rev. 7
0-MI-EBR-317-010.C, Testing of Molded Case Circuit Bkr’s not located on Vital Instrument, Vital Battery, or Perm H2 Mitigation PNL Boards, Rev. 13
ECA-0.0, Loss of all AC Power, Rev. 28
FSI-4, DC Bus Management and 480V FLEX D/G Alignment/Loading, Rev.1
FSI-5.01, Initial Assessment and Flex Equipment Deployment, Rev. 0
FSI-5.02, 6900V Flex D/G Startup and Alignment, Rev. 0
NPG-SPP-09.18.6, Lifecycle Management Practice for Printed Circuit Boards, Rev. 1
NPG-SPP-22.303, CR Actions, Closures, and Approvals, Rev. 8
0-MI-EMV-317-144.0, Procedure for Testing Motor Operated Valves Using MOVATS Signature Analysis System, Rev. 48
0-MI-MRR-067-002.0, Removal/Disassembly/Reassembly Instruction for ERCW Pumps, Rev. 20
0-PI-SFT-067-001.A, ERCW Train A Flow Monitoring, Rev. 2
0-PI-SFT-067-001.B, ERCW Train B Flow Monitoring, Rev. 2
0-SI-OPS-067-682.M, ERCW Non-Safety Related Flow Balance Valve Position Verification, Rev. 34
0-SI-OPS-067-682.Q, ERCW Non-Safety Related Flow Balance Valve Position Verification, Rev. 21
0-SI-SXP-067-201.M, Essential Raw Cooling Water Pump M-B Performance Test, Rev. 18, dated 9/10/09
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0-SI-SXV-063-266.0, ASME Code Valve Testing, Rev. 27, dated 4/28/15
0-SO-67-1, Essential Raw Cooling Water, Rev. 105
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EA-67-1, ERCW Operation, Rev. 8
TI-4, Maintenance Rule Performance Indicator Monitoring, Trending, and Reporting-10CFR50.65, Rev. 28
2-SI-OPS-000-003.Unit 2 Surveillance Instruction, Weekly Shift Log, dated 5/12/2016
0-GO-14-5 AUO Operator Rounds, Rev. 43
NPG-SPP-07.2, Outage Management, REV 0005
0-TI-PDM-000-003.0, Infrared Thermography Inspections, Rev 2
OPDP-1, Conduct of Operations, Rev. 38
AOP-R.01, Steam Generator Tube Leak, Rev. 32
AOP-C.04, Shutdown from Auxiliary Control Room, Rev. 38
AOP-P.04, Loss of Offsite Power, Rev. 34
E-0, Reactor Trip or Safety Injection, Rev. 39
ES-0.5, Equipment Verifications, Rev. 9
E-2, Faulted Steam Generator Isolation, Rev. 16
E-3, Steam Generator Tube Rupture, Rev. 21
1-SO-3-2, Auxiliary Feedwater, Rev. 57
2-SO-3-2, Auxiliary Feedwater, Rev. 50
0-SO-82-1, Diesel Generator 1A-A, Rev. 55
0-SO-82-7, Diesel Generator 2A-A Support Systems, Rev. 23
0-AR-DG-1A-LCL, Diesel Generator 1A-A Local Alarm Panel, Rev. 20
MMTP-102, Erection of Scaffolds, Temporary Work Platforms, and Ladders, Rev. 11

OPDP-8, Operability Determination Process and LCO Tracking, Rev. 21
2-SI-SXV-003-219.S, Turbine Driven Auxiliary Feedwater Check valve 2-3-810, 2-3-818, and 2-3-864 Closure Test, Rev. 5
0-SO-82-1, Diesel Generator 1A-A, Rev. 55
SQN-DC-V-21.0, Environmental Design, Rev. 24
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SPP-2.5, Vendor Manual Control, Rev. 6
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2-SI-SXP-003-202.S, TDFW Pump 2A-S Comprehensive Test, dated 12/17/15
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2-SI-SXP-074-201.B, Residual Heat Removal Pump 2B-B Performance Test, Rev. 17

Drawings
1,2-15E500-1, Key Diagram Station Auxiliary Power System, Rev. 53
1,2-45N700-3, Key Diagram Perm Hydrogen Mitigation, Misc EQPT Plant PWR & EVAC Alarm Sys, Rev. 1
1,2-45N779-39, Wiring Diagram 480V Shutdown Aux Power Schematic Diagram SH-39, Rev. 0
94-13298, ASA Series 300 14” NO S70DD Welding Ends Outside Screw and Yoke Gate Valve with SMB-0 Limitorque Valve Control and Limit Switches, Rev. 901
H-5149-DN, Sectional Illustration- 3 STG 300C- 42” x 24” Type A HD 24” x 2 15/16” Column and Shaft, Rev. 13
1,2-37W206-8, Mechanical Pumping Station Piping and Equipment, Rev. 4
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1-47W845-3, Mechanical Flow Diagram- Essential Raw Cooling Water System, Rev. 43
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1, 2-47W845-5, Mechanical Flow Diagram- Essential Raw Cooling Water System, Rev. 63
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2-47W845-4, Mechanical Flow Diagram- Essential Raw Cooling Water System, Rev. 22
2-47A941-123, Thrust Requirements for Motor Operated Valve 2-FCV-74-003, Rev. 2
1.2-45N 703-1, Wiring Diagrams 125V Vital Battery Board I, Single Line, Sheet 1, Rev. 80
1.2-45N 703-3, Wiring Diagrams 125V Vital Battery Board I, Single Line BOARD III, Sheet 3, Rev. 44
CCD No. 1,2-15E500-3, Transformer Taps & Voltage Limits – Aux Power System, Rev. 38
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CCD No. 2-45N724-3, Wiring Diagrams 6900V Shutdown Bd 2A-A, Rev. 26
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1-FCV-63-072, Documentation of Design Basis Review, Required Thrust Calc and Valve & Actuator Capability Assessment for 1-FCV-63-072, Rev. 6
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MDQ00099920110249, Sequoyah Nuclear Plant JOG MOV Periodic Verification Classification, Rev. 0
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OSG7-0027, Residual Heat Removal System (074) 10CFR50.49 Category and Operating Times, Rev.11
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SQN-APS-003, 480VAC APS Class 1E Load Coordination Study, Rev. 87
SQN-DES1-003, ERCW Screen Wash System Hydraulic Analysis and ERCW Pump NPSHA Analysis, Rev. 3
SQS20200, SQN Probabilistic Risk Assessment- Risk Ranking of Motor Operated Valves (MOV), Rev. 5
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SQN-CPS-057, Vital Control Power System Loading Channel I and Continuous Loading Evaluation of the 120VAC Vital Instrument Boards, Rev. 086
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SQN-DC-V-11.9, Motor Operated Valve Thermal Overload Setting and Torque Switch Bypass, Rev. 2
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SQN-DC-V-21.0, Environmental Design, Diesel generator Building Elevation 722.0, Rev. 24
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300196	1029015	1039435
588328	1145788	1155285
1093759	1061574	1144397
1173365	1061573	1120146
0692691	1080723	1087006
0753504	1080559	1078482
0832717	1133553	1056993
0858533	1144850	

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116500874	115180167	116710955
116561409	115320063	116903044
116773695	115351563	116960055
117021168	115564108	116399027
117133887	116399255	116401170
117331615	116409673	116146711
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G676258, Motor Operated Valves, Rev. 1
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Contract 73-83571, Damper design Data, dated 1/10/73
12-MEB-082-027, Engineering Work Request Diesel Generator Operability, Rev. 0
PMT-33, Post Modification Test Instructions Diesel Generator Building Heating and Ventilation System, Rev. 0
Letter TVA to NRC, Sequoyah Nuclear Plant Revised response to NRC Bulleting 88-04 – Potential safety Related Pump Loss, dated 3/15/90
SQN-VTD-W120-3440, Installation, Operation and Maintenance Manual for Swing Check Valves, Tag 8C54, Rev. 0
SQN-VTD-W120-4160, Installation, Operation and Maintenance for Unit 1 and Unit 2 Residual Heat Removal Pumps, Rev. dated 9/28/93

Specification 1616, Centrifugal Fans for Ventilating Systems Diesel Generator Building, dated 1/7/71
TI-ECS-75, B87950123008, Heat Room for Diesel Engines and Maximum Room Temperature in the Engine Room, Rev. 3
Change #22688, DG Air Intake dampers – Remote MCR Operation, Rev. A
Prompt Determination of Operability Documentation for CR 1170545, dated 5/16/16
Information Notice 90-61, Potential for Residual Heat removal Pump Damage Caused by Parallel Pump Interaction, dated 9/20/90
M-05420-A, Add an 8 inch Swing Check Valve Downstream of Each Heat Exchanger and RHR Miniflow Line, dated 10/10/90

Condition Reports Written Due to this Inspection
1170662, Provide Guidance on Energizing Hydrogen Igniters from FLEX
1172319, FLEX Procedures don't close H2 Igniter Breakers
1174214, Unvalidated 3 Hour Time Limit for Restoring Hydrogen Igniters following SBO
1174546, SQN DG HVAC and KW Rating Calculation Clarifications
1174919, SQN DG HVAC and KW Rating Calculation Clarifications
1175508, Loose Handrail on Steps Leading up to Trailer for Mobile 2MW 480V DG
1175555, 480V FLEX DG B Receptacle Cover Broken
1175748, PM has Handholes Listed with no Cable Submergence Levels Identified
1182240, CDBI procedure weakness related to D/G crankcase pressure trip
1175195, NRC identified JPM deficiency
1170010, No clearance between RHR pipe support eye-rod and 1” process pipe
1170096, Clean mesh over fire damper in the U2 TDAFWP Room
1170471, Consider generation of a SQN NPSH calculation for SQN
1170512, DG Building Corridor Damper grating is hanging down on one side
1170545, Diesel Generator building fire dampers during tornado
1170721, Fabricate special tool to block open DG CO2 dampers
1170753, Analysis for the TDAFW exhaust stack vulnerability to tornado missiles
1171143, Training Needs Analysis on AOP-N.02 tornado compensatory measure
1171265, Perform an air flow test on DG exhaust room fans
1171371, HP un-loader valve of Diesel 2B2 air compressor has an air leak
1171811, NRC 8-hr Notification for unanalyzed condition
1172103, DG Building Exhaust Fan discharge dampers not tested prior to PM
1174546, EDG Room Heat Load calculation
1174919, SQN DG HVAC and KW rating calculation clarifications
1175195, Procedural Deficiency on TDAFW pump using AOP-C.04 Appendix W.1 rather than 1-SO-3-2
1175834, DG exhaust fire damper unexpected closures
1175857, Inappropriate CR disposition on DG fire damper closure
1177144, QA3 bracket installed on safety related DG building supply fire dampers
1178891, Impact of tornado causing actuation of DG crankcase overpressure trip
1180402, Vendor Manual utilized instead of DCN to install brackets on DG fire dampers
1181710, DG Building ventilation system and tornado impacts
1182240, Procedure weakness related to DG crankcase pressure trip
1169607, Apparent missed breaker surveillance
1182472, Insufficient monitoring of the 480V shutdown board transformers