UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
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ATLANTA, GEORGIA 30303-1257

March 14, 2013

Mr. Tom E. Tynan
Vice President - Vogtle
Southern Nuclear Operating Company, Inc.
Vogtle Electric Generating Plant
7821 River Road
Waynesboro, GA 30830

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT - NRC INTEGRATED INSPECTION REPORT 05000424/2012005 AND 05000425/2012005 ERRATA

Dear Mr. Tynan:

On February 01, 2013, the US Nuclear Regulatory Commission (NRC) issued the subject inspection report for the Vogtle Electric Generating Plant, ADAMS accession ML13032A277. In reviewing this report, it was noted that we inadvertently omitted the Operator Workaround sample write-up in section 4OA2.2. Accordingly, we are providing a revised version of Inspection Report 05000424/425/2012005 that documents the above change.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (The Public Electronic Reading Room).

I apologize for any inconvenience this error may have caused. If you have any questions, please contact me at (404) 997-4611.

Sincerely,

/RA/

Frank Ehrhardt, Chief
Reactor Projects Branch 4
Division of Reactor Projects

Docket No.: 50-424, 50-425
License No.: NPF-68, NPF-81

Enclosure: Inspection Report 05000424, 425/2012005

cc w/encl: (See page 2)

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Letter to Tom E. Tynan from Frank Ehrhardt dated March 13, 2013

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT - NRC INTEGRATED INSPECTION REPORT 05000424/2012005 AND 05000425/2012005 ERRATA

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4OA2 Identification and Resolution of Problems

.1 Daily Condition Report Review. As required by inspection procedure 71152, Identification and Resolution of Problems, and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. This review was accomplished by either attending daily screening meetings that briefly discussed major CRs, or accessing the licensee's computerized corrective action database and reviewing each CR that was initiated.

.2 Operator Work-Around Annual Review

a. Inspection Scope

The inspectors performed a review of the licensee's operator work-around associated with the failure of the Unit 1 Loop 2 & 3 outboard MSIVs in the closed position during startup. The goal of the review was to verify that the full extent of the issue was identified, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized. The inspectors also performed a detailed review of this issue in accordance with the operator work-around inspection guidance. The inspectors reviewed the compensatory actions and cumulative effects on plant operation. The inspectors verified this issue was being dispositioned in accordance with plant procedure 10025-C, Work-Around Program. The inspectors evaluated the CR against the licensees corrective action program as delineated in licensee procedure NMP-GM-002, Corrective Action Program, and 10 CFR 50, Appendix B, Criterion XVI, Corrective Action. Documents reviewed are listed in the Attachment.

- 530916 – Unit 1 steam generators 2&3 do not indicate steam flow

b. Findings and Observations

Introduction: A Green self-revealing non-cited violation (NCV) of 10 CFR 50 Appendix B Criterion V, "Instructions, Procedures, and Drawings" was identified for failure to provide adequate work instructions in the operations and maintenance procedures used to open main steam isolation valves (MSIVs) that were bound in their closed seat. Specifically, the operations and maintenance procedures used to open the loop 2 and loop 3 outboard MSIVs did not provide instructions to limit the magnitude of the force applied to the valve stems while attempting to open the valves. Investigation revealed that the cause of the stem failures was excessive force applied to the thermally embrittled stems.

Description: On October 8, with Unit 1 in Mode 2, the operators had begun preparations for power ascension. At 1616, as the main feed pump was being placed on line, the control room operators noted a divergence in RCS loop differential temperatures (ΔTs), steam pressures, and steam flows between loops 1 & 4 and loops 2 & 3. Loops 1 & 4 showed increasing loop ΔTs, lowering steam pressure, and some minimal steam flow, while loops 2 & 3 showed no loop ΔT, increasing steam pressures (to the point of lifting the loop 2 & 3 atmospheric relief valves), and no steam flow. The Main Control Board hand switches indicated that all MSIVs and associated bypass valves were open. The operators identified the potential impact to the core neutron flux and stopped power ascension. Following discussions with plant management and engineering, the operators placed the plant in a safe condition by inserting a manual trip of the reactor at 2155. The licensee subsequently assembled an Issue Response Team (IRT) and a root cause team to investigate the cause of the diverging indications and to determine the required corrective actions.

The investigations revealed that the outboard MSIVs on both loops 2 & 3 were failed in the closed position. Upon disassembly, it was discovered the stems of both the failed MSIVs had undergone brittle fracture just above the T-head, where the valve stem is connected to the valve disk. Westinghouse representatives were consulted on the MSIV issue. They conveyed to the licensee that the material used for the MSIV stems, ASME SA564 Gr. 630PH T 17-4 PH heat treated to 1100^{o}F, is susceptible to embrittlement when exposed to temperatures above 500^{o}F for a sustained period (after about 10 years). Metallurgical analysis performed on the sheared stems validated that thermal embrittlement was the failure mechanism. The failure analysis concluded that both stem fractures were the result of sudden brittle failures from single tensile stress events. Further investigation by the IRT revealed that the loop 2 outboard MSIV stem failed during main steam line warming evolutions conducted on October 6 by operations personnel. The IRT also determined that that the loop 3 outboard MSIV stem failed on the night of October 7 following activities performed by maintenance personnel to lift the valve disk off its closed seat.

The root cause team determined that the root cause of the MSIV stem failures was temperature aging embrittlement of the stem material. The team also determined that the major contributing causes of the event were thermal binding of the valve disks in the closed seat and inadequate procedural guidance, i.e. procedures used to open the MSIVs did not provide instructions or guidance to limit the magnitude of the force applied to the valve stems while attempting to open the valves, which ultimately resulted in the brittle failure of the valve stems. The inadequate procedures specified by the root cause team were operating procedures 12001-C, “Unit Heat Up to Hot Shutdown (Mode 5 to Mode 4)”, and 14850-1/2, “Cold Shutdown Valve In-Service Test”, and maintenance procedure 26854-C, “MSIV Actuator Maintenance”. The licensee conducted ultrasonic testing on the remaining six Unit 1 MSIVs to verify that the valve stems were intact. The two failed valve stems were replaced, and the reactor was restarted on October 17. The licensee entered this issue into their corrective action program as CR 530916.

<u>Analysis</u>: The failure to provide adequate work instructions in the operations and maintenance procedures used to open main steam isolation valves (MSIVs) that were stuck on their closed seat was a performance deficiency. The inspectors concluded that the finding was more than minor because it was associated with the procedure quality attribute of the reactor safety - initiating events cornerstone and it adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the failure to provide adequate work instructions to operations and maintenance personnel resulted in the failure of both the loop 2 and loop 3 outboard MSIVs and the subsequent manual reactor trip.

Using IMC 0609, Attachment 4, “Initial Characterization of Findings” dated June 19, 2012, the inspectors determined that finding affected the Initiating Events cornerstone. The inspectors evaluated the finding using IMC 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated 06/19/12. The inspectors used the Initial Screening and Characterization of Findings (IMC 0609.04 Exhibit 1, dated June 19, 2012) to characterize the finding. Since the inspectors answered “No” to the Exhibit 1, section B, Initiating Events screening question, the inspectors concluded that the finding was of very low safety significance (Green).

The primary cause of the performance deficiency, as determined by the inspectors, was less than adequate work planning and coordination. The inspectors determined that the cause of this finding was related to the work control component of the human performance cross-cutting area due to less-than-adequate work planning [H.3 (a)]. Specifically, the licensees’ procedures used to open the MSIVs that were stuck on their closed seat did not contain instructions or precautions to limit the magnitude of the force applied to the valve stems while attempting to open the valves.

Enforcement: The inspectors determined that the finding represents a violation of regulatory requirements because it involved inadequate operations and maintenance procedures used to operate safety-related plant equipment. 10 CFR 50 Appendix B Criterion V requires, in part, that procedures shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, the licensees’ procedures used to open the loop 2 and loop 3 outboard MSIVs did not provide instructions to limit the magnitude of the force applied to the valve stems while attempting to open the valves. As a result of the violation, the loop 2 and loop 3 MSIVs failed in the closed position, and the reactor was manually tripped on October 8, extending the 1R17 refueling outage for an additional nine days. The licensee conducted ultrasonic testing on the remaining six Unit 1 MSIVs to verify that the valve stems were intact. The stems of the loop 2 and loop 3 outboard MSIVs were replaced, and the Unit 1 reactor was restarted on October 17. Because this violation was of very low safety significance and it was entered into the licensee’s corrective action program as CR 530916, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. (NCV 05000424/2012005-03, Inadequate Operations and Maintenance Procedures Results in Brittle Failure of the Loop 2 and Loop 3 Outboard MSIV Stems.)

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee’s Corrective Action Program and associated documents to identify trends which could indicate the existence of a more significant safety issue. The review was focused on repetitive equipment issues, but also considered the results of inspector daily CR screening and the licensee’s trending efforts. The review nominally considered the six month period of April 2012 through September 2012 although some examples extended beyond those dates when the scope of the trend warranted. The inspectors also reviewed several CRs associated with operability determinations which occurred during the period. Corrective actions associated with a sample of the issues identified in the licensee's trend reports were reviewed for adequacy. The inspectors also evaluated the trend reports against the requirements of the licensee's corrective action program as specified in licensee procedure NMP-GM-002, Corrective Action Program, and 10 CFR 50, Appendix B.

b. Findings and Observations

No findings were identified.

4OA3 Follow-up of Events and Notices of Enforcement Discretion (71153)

.1 (Closed) LER 05000424/2012-003-00 Failure to Comply with Technical Specification LCOs 3.7.14 and 3.0.3

a. Inspection Scope

On August 17, 2012, 1A ESF Chiller condenser vacuum was noted to be 12 inches of mercury, with a vacuum of 15 inches of mercury specified as the low limit on operating logs. The shift supervisor mistakenly believed condenser pressure was one of the parameters which engineering had evaluated and was continuing to monitor with a recorder. Condenser pressure was not one of the parameters being monitored and recorded on a recorder. When the condenser pressure was recorded as out of specification on the operator rounds log sheet, the shift supervisor failed to initiate operability and reportability determination processes. This misinformation was carried forward through subsequent shifts via logs. During the next five days, 1A ESF Chiller condenser vacuum decreased to 4 inches of mercury and stabilized for an additional four days prior to initiation of a CR on August 26, 2012. Subsequent investigation and consultation with the vendor determined the 1A ESF Chiller was inoperable and the TS LCO was entered at 1437 on August 26, 2012. As a result of the delay in recognition of the status of the subject chiller, appropriate actions of LCOs 3.7.14 and 3.0.3 were not taken. The inspectors reviewed the LER, the associated CR and enhanced apparent cause determination, and subsequent action items.

b. Findings

One licensee-identified violation was identified, and is documented in section 4OA7 of this report. This LER is closed.

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors conducted observations of security force personnel and activities to ensure that the activities were consistent with licensee security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours.

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status review and inspection activities.

b. Findings and Observations

No findings were identified.

.2 (Discussed) Temporary Instruction 2515/187 – Inspection of Near-Term Task Force Recommendation 2.3 Flooding Walkdowns

a. Inspection Scope

Inspectors conducted independent walkdowns to verify that the licensee completed the actions associated with the flood protection feature specified in paragraph 03.02.a.2 of this TI. Inspectors are performing walkdowns at all sites in response to a letter from the NRC to licensees, entitled "Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," dated March 12, 2012 (ADAMS Accession No. ML12053A340).

Enclosure 4 of the letter requested licensees to perform external flooding walkdowns using an NRC-endorsed walkdown methodology (ADAMS Accession No. ML12056A050). Nuclear Energy Industry (NEI) document 12-07 titled, "Guidelines for Performing Verification Walkdowns of Plant Protection Features," (ADAMS Accession No. ML12173A215) provided the NRC-endorsed methodology for assessing external flood protection and mitigation capabilities to verify that plant features, credited in the CLB for protection and mitigation from external flood events, and are available, functional, and properly maintained.

b. Findings and Observations

Findings or violations associated with the flooding, if any, will be documented in the 1st quarter integrated inspection report of 2013.

.3 Temporary Instruction 2515/188 – Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns

a. Inspection Scope

The inspectors accompanied the licensee on their seismic walkdowns of the following SWEL 1 and SWEL 2 components:

- Unit 1 Diesel Fuel Oil Transfer Pump A, SWEL 1 item #60, on August 15 in the Diesel Fuel Oil Storage Tank Building
- Unit 1B Diesel Generator Control Panel, SWEL 1 item #61, on August 15 in the Diesel Generator Building

- Unit 1 Turbine-Driven AFW Pump and Turbine Driver, SWEL 1 item #13, on August 16 in the AFW Pump House
- Unit 2 Spent Fuel Pool Heat Exchanger B, SWEL 2 item #1, on August 21 in the Auxiliary Building

The inspectors verified that the licensee confirmed that the following seismic features associated with the above listed components were free of potential adverse seismic conditions:

- Anchorage was free of bent, broken, missing or loose hardware
- Anchorage was free of corrosion that is more than mild surface corrosion
- Anchorage is free of visible cracks in the concrete near the anchors
- Anchorage configuration was consistent with plant documentation
- SSCs will not be damaged from impact by nearby equipment or structures
- Overhead equipment, distribution systems, ceiling tiles and lighting, and masonry block walls are secure and not likely to collapse onto the equipment
- Attached lines have adequate flexibility to avoid damage
- The area appears to be free of potentially adverse seismic interactions that could cause flooding or spray in the area
- The area appears to be free of potentially adverse seismic interactions that could cause a fire in the area
- The area appears to be free of potentially adverse seismic interactions associated with housekeeping practices, storage of portable equipment, and temporary installations (e.g., scaffolding, lead shielding)

The inspectors independently performed their walkdowns and verified that the following components were free of the potential adverse seismic conditions listed above:

- Unit 2A Diesel Generator Air Start Receiver #1, SWEL 1 item #55, on December 17 in the Diesel Generator Building
- Unit 1 Spent Fuel Pool Pump B, SWEL item #2, on December 17 in the Auxiliary Building

Observations made during the walkdowns that could not be determined to be acceptable were entered into the licensee's corrective action program for evaluation.

Additionally, inspectors verified that items that could allow the spent fuel pool to drain down rapidly were added to the SWEL and these items were walked down by the licensee.

b. <u>Findings and Observations</u>

No findings were identified.

4OA6 Meetings, Including Exit

.1 Exit Meeting

On January 11, the resident inspectors presented the inspection results to Mr. Tom Tynan and other members of his staff, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

4OA7 Licensee-Identified Violations

The following violations of very low significance (Green) or Severity Level IV were identified by the licensee and are violations of NRC requirements which meet the criteria of the NRC Enforcement Policy for being dispositioned as Non-cited Violations.

.1 Failure to Comply with Technical Specification LCOs 3.7.14 and 3.0.3

TS 3.0.3 requires, in part, that when a limiting condition of operation (LCO) is not met and the associated actions are not met, an associated action is not provided, or if directed by the associated actions, the unit shall be placed in a mode or other specified condition in which the LCO is not applicable. TS 3.7.14 require that two engineered safety feature (ESF) room cooler and safety-related chiller trains shall be operable. Contrary to the above, on August 17, 2012, at approximately midnight, the unit 1 shift supervisor failed to enter the required action statement for TS LCO 3.7.14, Condition 'A' when the unit 1A ESF chiller condenser purge pressure was noted to be out of specification high. Inoperability of the chiller was not recognized until August 26, 2012, and the LCO entered at 1437. Further, during the extended period during which the 1A ESF chiller was inoperable (albeit unrecognized as inoperable), opposite train supported components as well as redundant room coolers on the train B ESF Chiller and room cooler train were removed from service for unrelated activities which resulted in two occasions during which TS 3.0.3 should have been applied. The licensee documented this event in their corrective action program as CR 507143. Using IMC 0609, dated June 19, 2012, Attachment 4, Table 2, the inspectors verified that the finding affected the mitigation systems cornerstone. IMC 0609 Attachment 4 Table 3 directed the inspectors to use IMC 0609 Appendix A to characterize the finding. Because the finding represented an actual loss of function of one train of ECCS for greater than its TS Allowed Outage Time, a detailed risk evaluation was required. A detailed risk evaluation was performed by a regional senior reactor analyst in accordance with IMC 0609 Appendix A guidance using the NRC Vogtle SPAR model and the Saphire 8 risk analysis code. An Event/Condition Analysis module in Saphire was run with the unit 1A train ESF chiller failed with no recovery allowed for a 9 day exposure period. The dominant sequence was a loss of offsite power with success of reactor trip and emergency power with late failure of feedwater and failure to implement feed and bleed cooling due to failure of the Unit 1B train chiller and loss of the safety related switchgear. The detailed risk evaluation determined that the risk due to the performance deficiency was an increase in core damage frequency of <1E-6/year, a GREEN finding of very low safety significance. The risk was mitigated by the availability of alternate train components and the short exposure period.

.2 Failure to Conduct Required ASME Code Section XI Inspections

On April 12, 2012, Vogtle staff identified that in-service inspections for the second 10-year ISI period were missed for eight ASME Code Class 1 valves. Valves 1/2 1208U6035, 1/2 1208U6036, 1/2 1208U6037 and 1/2 1208U6038 are chemical and volume control system normal and alternate charging check valves to the reactor coolant system. Leakage control devices (seal encapsulation devices) were installed on the Unit 1 valves in 1987 to address recurring body-to-bonnet leakage per an industry approved Westinghouse design change. The seal caps were subsequently installed on the unit 2 valves in 1989. Title 10 CFR 50.55a(g)(4) requires, in part, that licensees follow the pressure test requirements of ASME Code Section XI. ASME Code, Section XI, IWA-5240, requires visual examinations as part of system pressure tests. ASME Code Section XI, IWA-5242, 1998 Edition through 2000 addenda, requires VT-2 visual examinations for pressure retaining bolted connections in borated water systems. Contrary to the above, from October, 1987, to the present, Vogtle did not perform a visual inspection of the valve body-to-bonnet studs. This finding was more than minor because it impacted the initiating events cornerstone and its attribute of equipment performance. Specifically, it affected the objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Using Inspection manual chapter 0609, dated June 19, 2012, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," this finding was determined to be of very low safety significance because the licensee's evaluation was able to demonstrate structural integrity. Specifically, stud stress was not sufficiently close to the yield stress to cause a loss of integrity. Therefore, the finding does not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment will not be available. The licensee has entered this issue into their corrective action program as CRs 438268, 458567, 505111 and 547078. To address the issue for the short term, the licensee plans to follow the 'needed' and 'good practice' recommendations detailed by the PWROG in letter OG-12-330 which was issued on August 16, 2012. The long term corrective actions will be to remove all of the existing seal caps and install a bonnet with a canopy seal weld to remove the need for a seal cap as a way to mitigate the effects of leakage and to allow visual examination of the bolted connections.

.3 Failure to Post High Radiation Area

10 CFR 20.1902(b) requires licensees to post each HRA with a conspicuous sign or signs bearing the radiation symbol and the words "CAUTION, HIGH RADIATION AREA" or "DANGER, HIGH RADIATION AREA". Contrary to this, on September 18, 2012, the entryway into the Unit 1 spent fuel pool cooling system demineralizer valve gallery was discovered to be missing a conspicuous sign bearing the radiation symbol and the words "CAUTION, HIGH RADIATION AREA" or "DANGER, HIGH RADIATION AREA." Accessible areas inside the Valve Gallery room contained dose rates up to 327 mrem/hr at 30 cm. A HP foreman discovered this violation while performing a walkdown of HRA postings in the auxiliary building. The licensee took immediate corrective actions upon discovery including restoration of the HRA posting (CR 519818). There was no evidence of unauthorized worker entry into the affected area. Although this event involved the failure to maintain proper control for a HRA, this finding is of very low safety

significance because it was not related to ALARA planning, nor did it involve an overexposure or substantial potential for overexposure, and the ability to assess dose was not compromised.

ATTACHMENT: SUPPLEMENTAL INFORMATION

4OA2 Identification and Resolution of Problems

.1 Daily Condition Report Review. As required by inspection procedure 71152, Identification and Resolution of Problems, and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. This review was accomplished by either attending daily screening meetings that briefly discussed major CRs, or accessing the licensee's computerized corrective action database and reviewing each CR that was initiated.

.2 Focused Review

a. Inspection Scope

The inspectors performed a detailed review of the following CR which addressed the failure of the Unit 1 Loop 2 & 3 outboard MSIVs in the closed position during startup. The goal of the review was to verify that the full extent of the issue was identified, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized. The inspectors evaluated the CR against the licensees corrective action program as delineated in licensee procedure NMP-GM-002, Corrective Action Program, and 10 CFR 50, Appendix B, Criterion XVI, Corrective Action. Documents reviewed are listed in the Attachment.

- 530916 – Unit 1 steam generators 2&3 do not indicate steam flow

b. Findings and Observations

Introduction: A Green self-revealing non-cited violation (NCV) of 10 CFR 50 Appendix B Criterion V, "Instructions, Procedures, and Drawings" was identified for failure to provide adequate work instructions in the operations and maintenance procedures used to open main steam isolation valves (MSIVs) that were bound in their closed seat. Specifically, the operations and maintenance procedures used to open the loop 2 and loop 3 outboard MSIVs did not provide instructions to limit the magnitude of the force applied to the valve stems while attempting to open the valves. Investigation revealed that the cause of the stem failures was excessive force applied to the thermally embrittled stems.

Description: On October 8, with Unit 1 in Mode 2, the operators had begun preparations for power ascension. At 1616, as the main feed pump was being placed on line, the control room operators noted a divergence in RCS loop differential temperatures (ΔTs), steam pressures, and steam flows between loops 1 & 4 and loops 2 & 3. Loops 1 & 4 showed increasing loop ΔTs, lowering steam pressure, and some minimal steam flow, while loops 2 & 3 showed no loop ΔT, increasing steam pressures (to the point of lifting the loop 2 & 3 atmospheric relief valves), and no steam flow. The Main Control Board hand switches indicated that all MSIVs and associated bypass valves were open. The operators identified the potential impact to the core neutron flux and stopped power ascension. Following discussions with plant management and engineering, the operators placed the plant in a safe condition by inserting a manual trip of the reactor at

2155. The licensee subsequently assembled an Issue Response Team (IRT) and a root cause team to investigate the cause of the diverging indications and to determine the required corrective actions.

The investigations revealed that the outboard MSIVs on both loops 2 & 3 were failed in the closed position. Upon disassembly, it was discovered the stems of both the failed MSIVs had undergone brittle fracture just above the T-head, where the valve stem is connected to the valve disk. Westinghouse representatives were consulted on the MSIV issue. They conveyed to the licensee that the material used for the MSIV stems, ASME SA564 Gr. 630PH T 17-4 PH heat treated to 1100^{o}F, is susceptible to embrittlement when exposed to temperatures above 500^{o}F for a sustained period (after about 10 years). Metallurgical analysis performed on the sheared stems validated that thermal embrittlement was the failure mechanism. The failure analysis concluded that both stem fractures were the result of sudden brittle failures from single tensile stress events. Further investigation by the IRT revealed that the loop 2 outboard MSIV stem failed during main steam line warming evolutions conducted on October 6 by operations personnel. The IRT also determined that that the loop 3 outboard MSIV stem failed on the night of October 7 following activities performed by maintenance personnel to lift the valve disk off its closed seat.

The root cause team determined that the root cause of the MSIV stem failures was temperature aging embrittlement of the stem material. The team also determined that the major contributing causes of the event were thermal binding of the valve disks in the closed seat and inadequate procedural guidance, i.e. procedures used to open the MSIVs did not provide instructions or guidance to limit the magnitude of the force applied to the valve stems while attempting to open the valves, which ultimately resulted in the brittle failure of the valve stems. The inadequate procedures specified by the root cause team were operating procedures 12001-C, “Unit Heat Up to Hot Shutdown (Mode 5 to Mode 4)”, and 14850-1/2, “Cold Shutdown Valve In-Service Test”, and maintenance procedure 26854-C, “MSIV Actuator Maintenance”. The licensee conducted ultrasonic testing on the remaining six Unit 1 MSIVs to verify that the valve stems were intact. The two failed valve stems were replaced, and the reactor was restarted on October 17. The licensee entered this issue into their corrective action program as CR 530916.

<u>Analysis</u>: The failure to provide adequate work instructions in the operations and maintenance procedures used to open main steam isolation valves (MSIVs) that were stuck on their closed seat was a performance deficiency. The inspectors concluded that the finding was more than minor because it was associated with the procedure quality attribute of the reactor safety - initiating events cornerstone and it adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the failure to provide adequate work instructions to operations and maintenance personnel resulted in the failure of both the loop 2 and loop 3 outboard MSIVs and the subsequent manual reactor trip.

Using IMC 0609, Attachment 4, “Initial Characterization of Findings” dated June 19, 2012, the inspectors determined that finding affected the Initiating Events cornerstone. The inspectors evaluated the finding using IMC 0609, Appendix A, “The Significance

Determination Process (SDP) for Findings At-Power," dated 06/19/12. The inspectors used the Initial Screening and Characterization of Findings (IMC 0609.04 Exhibit 1, dated June 19, 2012) to characterize the finding. Since the inspectors answered "No" to the Exhibit 1, section B, Initiating Events screening question, the inspectors concluded that the finding was of very low safety significance (Green).

The primary cause of the performance deficiency, as determined by the inspectors, was less than adequate work planning and coordination. The inspectors determined that the cause of this finding was related to the work control component of the human performance cross-cutting area due to less-than-adequate work planning [H.3 (a)]. Specifically, the licensees' procedures used to open the MSIVs that were stuck on their closed seat did not contain instructions or precautions to limit the magnitude of the force applied to the valve stems while attempting to open the valves.

Enforcement: The inspectors determined that the finding represents a violation of regulatory requirements because it involved inadequate operations and maintenance procedures used to operate safety-related plant equipment. 10 CFR 50 Appendix B Criterion V requires, in part, that procedures shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, the licensees' procedures used to open the loop 2 and loop 3 outboard MSIVs did not provide instructions to limit the magnitude of the force applied to the valve stems while attempting to open the valves. As a result of the violation, the loop 2 and loop 3 MSIVs failed in the closed position, and the reactor was manually tripped on October 8, extending the 1R17 refueling outage for an additional nine days. The licensee conducted ultrasonic testing on the remaining six Unit 1 MSIVs to verify that the valve stems were intact. The stems of the loop 2 and loop 3 outboard MSIVs were replaced, and the Unit 1 reactor was restarted on October 17. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as CR 530916, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. (NCV 05000424/2012005-03, Inadequate Operations and Maintenance Procedures Results in Brittle Failure of the Loop 2 and Loop 3 Outboard MSIV Stems.)

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's Corrective Action Program and associated documents to identify trends which could indicate the existence of a more significant safety issue. The review was focused on repetitive equipment issues, but also considered the results of inspector daily CR screening and the licensee's trending efforts. The review nominally considered the six month period of April 2012 through September 2012 although some examples extended beyond those dates when the scope of the trend warranted. The inspectors also reviewed several CRs associated with operability determinations which occurred during the period. Corrective actions associated with a sample of the issues identified in the licensee's trend reports were reviewed for adequacy. The inspectors also evaluated the trend reports against the requirements of the licensee's corrective action program as specified in licensee procedure NMP-GM-002, Corrective Action Program, and 10 CFR 50, Appendix B.

b. Findings and Observations

No findings were identified.

4OA3 Follow-up of Events and Notices of Enforcement Discretion (71153)

.1 (Closed) LER 05000424/2012-003-00 Failure to Comply with Technical Specification LCOs 3.7.14 and 3.0.3

a. Inspection Scope

On August 17, 2012, 1A ESF Chiller condenser vacuum was noted to be 12 inches of mercury, with a vacuum of 15 inches of mercury specified as the low limit on operating logs. The shift supervisor mistakenly believed condenser pressure was one of the parameters which engineering had evaluated and was continuing to monitor with a recorder. Condenser pressure was not one of the parameters being monitored and recorded on a recorder. When the condenser pressure was recorded as out of specification on the operator rounds log sheet, the shift supervisor failed to initiate operability and reportability determination processes. This misinformation was carried forward through subsequent shifts via logs. During the next five days, 1A ESF Chiller condenser vacuum decreased to 4 inches of mercury and stabilized for an additional four days prior to initiation of a CR on August 26, 2012. Subsequent investigation and consultation with the vendor determined the 1A ESF Chiller was inoperable and the TS LCO was entered at 1437 on August 26, 2012. As a result of the delay in recognition of the status of the subject chiller, appropriate actions of LCOs 3.7.14 and 3.0.3 were not taken. The inspectors reviewed the LER, the associated CR and enhanced apparent cause determination, and subsequent action items.

b. Findings

One licensee-identified violation was identified, and is documented in section 4OA7 of this report. This LER is closed.

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors conducted observations of security force personnel and activities to ensure that the activities were consistent with licensee security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours.

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status review and inspection activities.

b. Findings and Observations

No findings were identified.

.2 (Discussed) Temporary Instruction 2515/187 – Inspection of Near-Term Task Force Recommendation 2.3 Flooding Walkdowns

a. Inspection Scope

Inspectors conducted independent walkdowns to verify that the licensee completed the actions associated with the flood protection feature specified in paragraph 03.02.a.2 of this TI. Inspectors are performing walkdowns at all sites in response to a letter from the NRC to licensees, entitled "Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," dated March 12, 2012 (ADAMS Accession No. ML12053A340).

Enclosure 4 of the letter requested licensees to perform external flooding walkdowns using an NRC-endorsed walkdown methodology (ADAMS Accession No. ML12056A050). Nuclear Energy Industry (NEI) document 12-07 titled, "Guidelines for Performing Verification Walkdowns of Plant Protection Features," (ADAMS Accession No. ML12173A215) provided the NRC-endorsed methodology for assessing external flood protection and mitigation capabilities to verify that plant features, credited in the CLB for protection and mitigation from external flood events, and are available, functional, and properly maintained.

b. Findings and Observations

Findings or violations associated with the flooding, if any, will be documented in the 1st quarter integrated inspection report of 2013.

.3 Temporary Instruction 2515/188 – Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns

a. Inspection Scope

The inspectors accompanied the licensee on their seismic walkdowns of the following SWEL 1 and SWEL 2 components:

- Unit 1 Diesel Fuel Oil Transfer Pump A, SWEL 1 item #60, on August 15 in the Diesel Fuel Oil Storage Tank Building
- Unit 1B Diesel Generator Control Panel, SWEL 1 item #61, on August 15 in the Diesel Generator Building

- Unit 1 Turbine-Driven AFW Pump and Turbine Driver, SWEL 1 item #13, on August 16 in the AFW Pump House
- Unit 2 Spent Fuel Pool Heat Exchanger B, SWEL 2 item #1, on August 21 in the Auxiliary Building

The inspectors verified that the licensee confirmed that the following seismic features associated with the above listed components were free of potential adverse seismic conditions:

- Anchorage was free of bent, broken, missing or loose hardware
- Anchorage was free of corrosion that is more than mild surface corrosion
- Anchorage is free of visible cracks in the concrete near the anchors
- Anchorage configuration was consistent with plant documentation
- SSCs will not be damaged from impact by nearby equipment or structures
- Overhead equipment, distribution systems, ceiling tiles and lighting, and masonry block walls are secure and not likely to collapse onto the equipment
- Attached lines have adequate flexibility to avoid damage
- The area appears to be free of potentially adverse seismic interactions that could cause flooding or spray in the area
- The area appears to be free of potentially adverse seismic interactions that could cause a fire in the area
- The area appears to be free of potentially adverse seismic interactions associated with housekeeping practices, storage of portable equipment, and temporary installations (e.g., scaffolding, lead shielding)

The inspectors independently performed their walkdowns and verified that the following components were free of the potential adverse seismic conditions listed above:

- Unit 2A Diesel Generator Air Start Receiver #1, SWEL 1 item #55, on December 17 in the Diesel Generator Building
- Unit 1 Spent Fuel Pool Pump B, SWEL item #2, on December 17 in the Auxiliary Building

Observations made during the walkdowns that could not be determined to be acceptable were entered into the licensee's corrective action program for evaluation.

Additionally, inspectors verified that items that could allow the spent fuel pool to drain down rapidly were added to the SWEL and these items were walked down by the licensee.

b. Findings and Observations

No findings were identified.

4OA6 Meetings, Including Exit

.1 Exit Meeting

On January 11, the resident inspectors presented the inspection results to Mr. Tom Tynan and other members of his staff, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

4OA7 Licensee-Identified Violations

The following violations of very low significance (Green) or Severity Level IV were identified by the licensee and are violations of NRC requirements which meet the criteria of the NRC Enforcement Policy for being dispositioned as Non-cited Violations.

.1 Failure to Comply with Technical Specification LCOs 3.7.14 and 3.0.3

TS 3.0.3 requires, in part, that when a limiting condition of operation (LCO) is not met and the associated actions are not met, an associated action is not provided, or if directed by the associated actions, the unit shall be placed in a mode or other specified condition in which the LCO is not applicable. TS 3.7.14 require that two engineered safety feature (ESF) room cooler and safety-related chiller trains shall be operable. Contrary to the above, on August 17, 2012, at approximately midnight, the unit 1 shift supervisor failed to enter the required action statement for TS LCO 3.7.14, Condition 'A' when the unit 1A ESF chiller condenser purge pressure was noted to be out of specification high. Inoperability of the chiller was not recognized until August 26, 2012, and the LCO entered at 1437. Further, during the extended period during which the 1A ESF chiller was inoperable (albeit unrecognized as inoperable), opposite train supported components as well as redundant room coolers on the train B ESF Chiller and room cooler train were removed from service for unrelated activities which resulted in two occasions during which TS 3.0.3 should have been applied. The licensee documented this event in their corrective action program as CR 507143. Using IMC 0609, dated June 19, 2012, Attachment 4, Table 2, the inspectors verified that the finding affected the mitigation systems cornerstone. IMC 0609 Attachment 4 Table 3 directed the inspectors to use IMC 0609 Appendix A to characterize the finding. Because the finding represented an actual loss of function of one train of ECCS for greater than its TS Allowed Outage Time, a detailed risk evaluation was required. A detailed risk evaluation was performed by a regional senior reactor analyst in accordance with IMC 0609 Appendix A guidance using the NRC Vogtle SPAR model and the Saphire 8 risk analysis code. An Event/Condition Analysis module in Saphire was run with the unit 1A train ESF chiller failed with no recovery allowed for a 9 day exposure period. The dominant sequence was a loss of offsite power with success of reactor trip and emergency power with late failure of feedwater and failure to implement feed and bleed cooling due to failure of the Unit 1B train chiller and loss of the safety related switchgear. The detailed risk evaluation determined that the risk due to the performance deficiency was an increase in core damage frequency of <1E-6/year, a GREEN finding of very low safety significance. The risk was mitigated by the availability of alternate train components and the short exposure period.

.2 Failure to Conduct Required ASME Code Section XI Inspections

On April 12, 2012, Vogtle staff identified that in-service inspections for the second 10-year ISI period were missed for eight ASME Code Class 1 valves. Valves 1/2 1208U6035, 1/2 1208U6036, 1/2 1208U6037 and 1/2 1208U6038 are chemical and volume control system normal and alternate charging check valves to the reactor coolant system. Leakage control devices (seal encapsulation devices) were installed on the Unit 1 valves in 1987 to address recurring body-to-bonnet leakage per an industry approved Westinghouse design change. The seal caps were subsequently installed on the unit 2 valves in 1989. Title 10 CFR 50.55a(g)(4) requires, in part, that licensees follow the pressure test requirements of ASME Code Section XI. ASME Code, Section XI, IWA-5240, requires visual examinations as part of system pressure tests. ASME Code Section XI, IWA-5242, 1998 Edition through 2000 addenda, requires VT-2 visual examinations for pressure retaining bolted connections in borated water systems. Contrary to the above, from October, 1987, to the present, Vogtle did not perform a visual inspection of the valve body-to-bonnet studs. This finding was more than minor because it impacted the initiating events cornerstone and its attribute of equipment performance. Specifically, it affected the objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Using Inspection manual chapter 0609, dated June 19, 2012, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” this finding was determined to be of very low safety significance because the licensee’s evaluation was able to demonstrate structural integrity. Specifically, stud stress was not sufficiently close to the yield stress to cause a loss of integrity. Therefore, the finding does not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment will not be available. The licensee has entered this issue into their corrective action program as CRs 438268, 458567, 505111 and 547078. To address the issue for the short term, the licensee plans to follow the ‘needed’ and ‘good practice’ recommendations detailed by the PWROG in letter OG-12-330 which was issued on August 16, 2012. The long term corrective actions will be to remove all of the existing seal caps and install a bonnet with a canopy seal weld to remove the need for a seal cap as a way to mitigate the effects of leakage and to allow visual examination of the bolted connections.

.3 Failure to Post High Radiation Area

10 CFR 20.1902(b) requires licensees to post each HRA with a conspicuous sign or signs bearing the radiation symbol and the words “CAUTION, HIGH RADIATION AREA” or “DANGER, HIGH RADIATION AREA”. Contrary to this, on September 18, 2012, the entryway into the Unit 1 spent fuel pool cooling system demineralizer valve gallery was discovered to be missing a conspicuous sign bearing the radiation symbol and the words “CAUTION, HIGH RADIATION AREA” or “DANGER, HIGH RADIATION AREA.” Accessible areas inside the Valve Gallery room contained dose rates up to 327 mrem/hr at 30 cm. A HP foreman discovered this violation while performing a walkdown of HRA postings in the auxiliary building. The licensee took immediate corrective actions upon discovery including restoration of the HRA posting (CR 519818). There was no evidence of unauthorized worker entry into the affected area. Although this event involved the failure to maintain proper control for a HRA, this finding is of very low safety

significance because it was not related to ALARA planning, nor did it involve an overexposure or substantial potential for overexposure, and the ability to assess dose was not compromised.

ATTACHMENT: SUPPLEMENTAL INFORMATION