Brunswick 2
1Q/2013 Plant Inspection Findings

Initiating Events

Significance: G Mar 31, 2013
Identified By: NRC
Item Type: FIN Finding
Failure to Follow Procedure for Variable Frequency Drive Reactor Recirculation Pump Design Modification.
An NRC-identified Green finding was identified for the failure of the licensee to follow Procedure EGR-NGGC-0005, Engineering Change (EC), when performing the variable frequency drive (VFD) modification for the reactor recirculation pumps (RRPs). Specifically, between April 4, 2010 and the present, the licensee inappropriately used a Rapid Field Release (RFR) to revise the power supplies for the relays in the VFD system without re-evaluating the EC, the 10 CFR 50.59 Screen/Evaluation, and the Failure Modes and Effects Analysis (FMEA). This resulted in a new failure mode on a loss of the power supply causing a RRP runback and placing the plant in a flow transient, and a loss of cooling to the RRP seals. The licensee entered this issue into the corrective action program (CAP) as nuclear condition report (NCR) 581202.

The performance deficiency associated with this finding was the failure of the licensee to follow Procedure EGR-NGGC-0005, Engineering Change (EC), when performing the VFD modification for the RRPs. The finding was more than minor because it was associated with the design control attribute of the Initiating Events Cornerstone and affects 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 VFD modification inappropriately causes a RRP runback on a loss of 480 VAC and core flow instability, and a loss of cooling to the RRP seals. Using IMC 0609, Appendix A, issued June 19, 2012, The SDP for Findings At-Power, the inspectors determined the finding was of very low safety significance because as a transient initiator due to the RRP runback, the finding did not cause a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition. The inspectors determined the finding was also of very low safety significance because as a loss of coolant accident (LOCA) initiator, after a reasonable assessment of degradation, the finding would not result in exceeding the reactor coolant system leak rate for a small break LOCA or likely affect other systems used to mitigate a LOCA resulting in a total loss of their function. The finding has a cross-cutting aspect in the area of human performance associated with the work control attribute because the licensee did not appropriately coordinate work activities by incorporating actions to address the impact of changes to the work scope, associated with the VFD modification, on the plant. [H.3(b)]
Inspection Report# : 2013002 *(pdf)*

Significance: G Dec 31, 2012
Identified By: NRC
Item Type: FIN Finding
Inadequate Maintenance Procedure for Fluorescent Lights over Safety-related Equipment
•Green. The inspectors identified a Green finding for the licensee not having an adequate procedure for maintenance on fluorescent lights over safety-related equipment. Specifically, between plant startup and August 29, 2012, the licensee did not have instructions for closing S-hooks on fluorescent lights over safety related equipment during maintenance on the fluorescent lights. This resulted in over 40 S-hooks open in safety-related buildings which could result in fluorescent lights falling and impacting safety-related equipment during a seismic event. The licensee's

corrective actions included closing the open S-hooks and adding instructions for closing S-hooks to work order (WO) 431558. The licensee entered this issue into the CAP as NCR 551646.

The performance deficiency associated with this finding was the failure of the licensee to have an adequate procedure for maintenance on fluorescent lights over safety-related equipment. The finding was more than minor because if left uncorrected, the deficiencies could lead to a more significant safety concern. If left uncorrected, the failure to provide procedural guidance to close the S-hooks on fluorescent lights over safety-related equipment could lead to fluorescent lights falling on safety-related instruments during a seismic event resulting in a reactor trip. This finding is also associated with the design control attribute of the Initiating Events Systems Cornerstone. Using IMC 0609, Appendix A, issued June 19, 2012, The Significance Determination Process (SDP) for Findings At-Power, the inspectors determined the finding was of very low safety significance because the finding did not affect the design or qualification of a mitigating SSC, the finding did not represent a loss of system and/or function, the finding did not represent an actual loss of a function of a single train for greater than the TS allowed outage time, the finding did not represent an actual loss of a function of one or more non-TS trains of equipment, and did not screen as potentially risk-significant due to a seismic event since both S-hooks on one fluorescent light were not considered to be completely failed or unavailable, and the finding did not involve the total loss of any safety function. The finding has a cross-cutting aspect in the area of problem identification and resolution associated with the CAP attribute because the licensee did not identify the open S-hook issue completely, accurately, and in a timely manner commensurate with their safety significance during the Fukushima walkdowns. [P.1(a)] (Section 4OA5)

Inspection Report# : 2012005 *(pdf)*

Significance: G Jun 30, 2012
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to follow plant procedure caused loss of E1 bus
A self-revealing Green NCV of Technical Specification (TS) 5.4.1, Procedures, was identified when the licensee failed to follow procedure 0MST-DG11R, Diesel Generator 1 Loading Test. During the preparation for the test, procedural steps were not performed correctly and the E1 electrical bus was inadvertently de-energized, requiring emergency diesel generator (EDG) 1 to auto-start and re-energize the bus. Once EDG 1 was supplying power to bus E1, the licensee exited from the surveillance procedure and restored offsite power to bus E1. The licensee entered the issue into their corrective action program as Action Request (AR) 529330.

The inspectors determined that the failure to follow procedure 0MST-DG11R, Diesel Generator 1 Loading Test, was a performance deficiency. The finding was determined to be more than minor because the finding was associated with the Initiating Events Cornerstone attribute of human performance and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, loss of the E1 bus adversely affected the shut down unit's defense-in-depth for the electrical power availability key safety function. Since Unit 1 was shut down at the time of the event, the finding's significance with regard to Unit 1 was evaluated using IMC 0609 Appendix G, Shutdown Operations Significance Determination Process. Since one offsite transmission network remained available to Unit 1 during the event, per Checklist 7 of IMC 0609 Appendix G, Attachment 1, the finding did not require a quantitative assessment. Therefore, the finding is of very low safety significance (Green) for Unit 1. Unit 2 was at power and was also affected by the finding. IMC 0609 Attachment 0609.04, Phase 1 - Initial Screening and Characterization of Findings, Table 4a for the Initiating Events Cornerstone was used to determine that the finding is of very low safety significance (Green) because the finding is a transient initiator that did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. The finding has a cross-cutting aspect in the Human Performance cross-cutting area, Work Practices component, because the licensee failed to implement adequate error prevention techniques while performing plant procedure 0MST-DG11R, Diesel Generator 1 Loading

Test. Specifically, technicians did not utilize adequate error prevention techniques to prevent them from connecting test recorders incorrectly, H.4(a). (4OA3)

Inspection Report# : 2012003 *(pdf)*

Significance: G Apr 04, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Properly Assemble Reactor Vessel Head Following Maintenance Outage
A self-revealing (Green) non-cited violation (NCV) of 10 CFR 50, Appendix B Criterion V, Instructions, Procedures, and Drawings was identified for failure to properly implement plant procedures for reactor pressure vessel (RPV) reassembly following the Unit 2 maintenance outage in November 2011. This resulted in excessive leakage from the Unit 2 RPV during reactor startup and pressurization on November 15 and November 16, 2011, and the declaration of an Unusual Event for reactor coolant system (RCS) unidentified
leakage in excess of 10 gallons per minute on November 16, 2011. The unit was shut down and depressurized on November 16, 2011, and the issue entered into the licensee's CAP as NCR 500035.
The licensee's failure to correctly implement procedure 0SMP-RPV502, Reactor Vessel Reassembly, to ensure that the RPV head was properly reassembled following the November 2011 Unit 2 maintenance outage was a performance deficiency. The finding was more than minor because it was associated with the Initiating Events cornerstone attribute of equipment performance (the reliability of the RCS barrier integrity) and adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during shutdown or power operations. Specifically, the failure to adequately implement this procedure resulted in excessive leakage from the Unit 2 RPV during reactor startup and pressurization. Inspection Manual Chapter 0609, Significance Determination Process (SDP), Attachment 0609.04, Phase 1 Screening Worksheet was used to screen the significance of the finding. The finding required a Phase
2 SDP analysis because it resulted in unidentified RCS leakage exceeding technical specification limits. Evaluation of the finding using the NRC pre-solved SDP table was not appropriate because the table does not contain a suitable target for RPV vessel integrity. Therefore, a Phase 3 SDP analysis was required. A Phase 3 analysis was performed by the regional Senior Reactor Analyst. Since the finding resulted in a shutdown, the SDP was analyzed as an additional transient that had a small potential to result in a Small Loss of Coolant Accident (SLOCA). The actual leak rate was low enough to not be considered to be a SLOCA, but there was potential for larger leakage. The Phase 2 SDP process uses an order of magnitude increase in the initiating event frequency for issues with the potential to increase the frequency of a particular event. This philosophy was used in the Phase 3 SDP process to allow a risk-informed input to the SDP for the SLOCA potential for this finding, due to the difficulty in calculating an exact percentage of time that the condition of the head closure would result in a larger leak. This resulted in an analysis that assumed a transient occurred that would result in a SLOCA about 1 percent of the time. This result represents an upper bound for the finding. The results were a risk in the low E-7 range, and the finding is GREEN. The SLOCA contribution was less than E-7. Dominant sequences involved loss of secondary side cooling and makeup, with either loss of containment heat removal, or loss of high pressure injection and failure to depressurize the reactor to allow the use of the low pressure systems. Because of Brunswick's concrete lined torus, and the low contribution of the high pressure sequences, the Large Early Release Frequency did not result in an increase in the significance. The cause of this finding was directly related to the cross-cutting aspect of supervisory and management oversight in the Work Practices component of the Human Performance area because oversight of the RPV reassembly was inadequate to insure that workers were able to accurately execute the steps of procedure 0SMP-RPV502, Reactor Vessel Reassembly. [H.4(c)]
Inspection Report# : 2012007 *(pdf)*

Significance: G Apr 04, 2012
Identified By: NRC

Item Type: NCV NonCited Violation
Failure to Perform a Post Maintenance Test After Reactor Pressure Vessel Assembly
A self-revealing (Green) non-cited violation (NCV) of 10 CFR 50, Appendix B Criterion V, Instructions, Procedures, and Drawings was identified for failure to properly implement plant procedure 0PLP-20, Post Maintenance Testing, after reactor pressure vessel (RPV) reassembly following the Unit 2 maintenance outage in November 2011. This resulted in the failure to identify improperly elongated RPV head studs, and contributed to excessive leakage from the Unit 2 RPV during reactor startup and pressurization on November 15 and November 16, 2011. The unit was shut down and depressurized on November 16, 2011, and the issue entered into the licensee's CAP as NCR 500035.
The licensee's failure to comply with procedure 0PLP-20, Post Maintenance Testing, to ensure that a post maintenance test (PMT) was performed to verify that the RPV head was properly reassembled following the November 2011 Unit 2 maintenance outage was a performance deficiency. The finding was more than minor because it was associated with the Initiating Events cornerstone attribute of equipment performance (the reliability of the RCS barrier integrity) and adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during shutdown or power operations. Specifically, the failure to perform a PMT after RPV reassembly contributed to excessive leakage from the Unit 2 RPV during reactor startup and pressurization. Inspection Manual Chapter 0609, Significance Determination Process (SDP), Attachment 0609.04, Phase 1 Screening Worksheet was used to screen the significance of the finding. The finding required a Phase 2 SDP analysis because it resulted in unidentified RCS leakage exceeding technical specification limits. Evaluation of the finding using the NRC pre-solved SDP table was not appropriate because the table does not contain a suitable target for RPV vessel integrity. Therefore, a Phase 3 SDP analysis was required.
The regional Senior Reactor Analyst determined that failure to perform a post maintenance test would have had the potential to mitigate the failure to adequately torque the RPV head studs, which was analyzed to be a Green finding (see NCV 05000324/2012007-01 above).
Since the impact of the mitigation would be less than the impact of the underlying finding, the failure to perform a post maintenance test is also a Green finding. The cause of this finding was directly related to the cross-cutting aspect of conservative assumptions in the decision making component of the Human Performance area because the licensee made nonconservative decisions regarding the need to perform a PMT following RPV assembly.
Inspection Report# : 2012007 *(pdf)*

Significance: G Apr 04, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Perform Adequate Training for Reactor Vessel Reassembly
NRC inspectors identified a Green non-cited violation (NCV) of 10 CFR 50, Appendix B Criterion V, Instructions, Procedures, and Drawings for failure to properly implement plant procedure TRN-NGGC-1000, Conduct of Training for training and qualifications of the reactor pressure vessel (RPV) reassembly team prior to RPV reassembly during the Unit 2 maintenance outage in November 2011. This resulted in inadequate worker knowledge of the tools and procedures associated with RPV reassembly, which contributed to the RPV head studs being inadequately tensioned and excessive leakage from the Unit 2 RPV during reactor startup and pressurization on November 15 and November 16, 2011. The unit was shut down and depressurized on November 16, 2011, and the issue entered into the licensee's CAP as NCR 500035.
The licensee's failure to comply with procedure TRN-NGGC-1000, Conduct of Training, to ensure that the maintenance team performing the RPV reassembly after the November 2011 Unit 2 maintenance outage received adequate training was a performance deficiency. The finding was more than minor because it was associated with the

Initiating Events cornerstone attribute of human performance and adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during shutdown or power operations. Specifically, the failure to adequately implement procedure TRN-NGGC-1000 contributed to the failure to adequately tension the RPV head studs during the Unit 2 November, 2011 maintenance outage, which resulted in excessive leakage from the Unit 2 RPV during reactor startup and pressurization. Inspection Manual Chapter 0609, Significance Determination Process (SDP), Attachment 0609.04, Phase 1 Screening Worksheet was used to screen the significance of the finding. The finding required a Phase 2 SDP analysis because it resulted in unidentified RCS leakage exceeding technical specification limits. Evaluation of the finding using the NRC pre-solved SDP table was not appropriate because the table does not contain a suitable target for RPV vessel integrity. Therefore, a Phase 3 SDP analysis was required. The regional Senior Reactor Analyst determined that adequate training of the RPV assembly team would have had the potential to mitigate the failure to adequately torque the RPV head studs, which was analyzed to be a Green finding (see NCV 05000324/2012007-01 above). Since the impact of the mitigation would be less than the impact of the underlying finding, this finding is also Green. The cause of this finding was directly related to the cross-cutting aspect of training in the Resources component of the Human Performance area because the licensee failed to provide sufficiently trained personnel to reassemble the RPV.
Inspection Report# : 2012007 *(pdf)*

Mitigating Systems

Significance: G Dec 31, 2012
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Inadequate Maintenance Procedure for the EDG Jacket Water Pump Wear Ring Tolerances
•Green. A self-revealing Green NCV of Technical Specification (TS) 5.4.1a, Procedures, was identified because the licensee did not have an adequate maintenance procedure to perform work on the emergency diesel generator (EDG) 3 engine-driven jacket water pump (JWP). Specifically, between July 25, 1992 and November 15, 2012, Procedure 0CM ENG528, Gould Engine Driven Jacket Water Pump Model 3736, did not provide the correct tolerances for the EDG JWP wear rings, resulting in the JWP seizure. The licensee's corrective actions included replacing the casing wear rings with wear rings with the correct tolerance and revising Procedure 0CM-ENG528. The licensee entered this issue into the corrective action program (CAP) as nuclear condition report (NCR) 572546.

The performance deficiency associated with this finding was the failure of the licensee to have an adequate procedure for maintenance on the EDG 3 engine-driven JWP. The finding was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems Cornerstone and affects the cornerstone objective of ensuring the availability and reliability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the inadequate procedure resulted in reduced availability of EDG 3 to repair the engine-driven JWP and reduced reliability of the jacket water system during operation. Using IMC 0609, Appendix A, issued June 19, 2012, The Significance Determination Process (SDP) for Findings At-Power, the inspectors determined the finding was of very low safety significance because the finding did not affect the design or qualification of a mitigating structure, system and component (SSC), the finding did not represent a loss of system and/or function, the finding did not represent an actual loss of a function of a single train for greater than the TS allowed outage time, the finding did not represent an actual loss of a function of one or more non-TS trains of equipment, and did not screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event. The finding does not have a cross-cutting aspect since the performance deficiency is not indicative of current plant performance. Procedure 0CM-ENG528 included the incorrect tolerances since July 25, 1992. (Section 1R19)

Inspection Report# : 2012005 *(pdf)*

Significance: G Dec 31, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Design of EDG 2 ASSD Switch A1
•Green. The inspectors identified a Green NCV of 10 CFR 50 Appendix B, Criterion III, Design Control, for failure to assure that the design basis for EDG 2 Alternate Safe Shutdown (ASSD) Switch A1 was correctly translated into specifications and drawings. Specifically, between original EDG 2 installation and September 1, 2012, a wiring discrepancy existed associated with EDG 2 ASSD Switch A1 which resulted in an induced fault that could have impacted the ability to locally control EDG 2 during certain fire scenarios. The licensee's corrective actions included correcting the EDG 2 control circuit wiring to ensure it was in accordance with the existing approved design and returning EDG 2 to operable status. The licensee entered this issue into the CAP as NCR 557897.

The performance deficiency associated with this finding was the failure to assure that the design basis for EDG 2 ASSD Switch A1 was correctly translated into specifications and drawings. The finding was more than minor because it was associated with the protection against external factors (i.e. fire) attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, an induced fault could have impacted the ability to locally control EDG 2 during certain fire scenarios. Using IMC 0609, Attachment 4, issued June 19, 2012, Initial Characterization of Findings, and IMC 0609, Appendix F, Attachment 1, Part 1: Application of Fire Protection SDP Phase 1 Worksheet, the results of this evaluation required further significance evaluation. A phase 3 analysis was performed by a regional SRA in accordance with NRC IMC 0609 Appendix F. The finding affected the capability to achieve alternate safe shutdown for Unit 1. The result of the analysis was an increase in core damage frequency of <1E-6/year a GREEN finding of very low safety significance. The finding does not have a cross-cutting aspect since the performance deficiency is not indicative of current plant performance. The EDG 2 ASSD Switch A1 wiring discrepancy has existed since original EDG installation. (Section 4OA3)

Inspection Report# : 2012005 *(pdf)*

Barrier Integrity

Emergency Preparedness

Significance: G Sep 30, 2012
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to Maintain Reliability and Availability of Emergency Response Equipment for Emergency Response Facilities
A self-revealing Green NCV of 10 CFR 50.54(q)(2) was identified for the licensee's failure to properly evaluate or consider the impact to emergency response facilities of design change ESR98-00436 which was implemented in 1999. This resulted in the loss of Emergency Response Facility Information System (ERFIS), Emergency Response Data System (ERDS), Safety Parameter Display System (SPDS), and all displays including radiation monitors for the

emergency response facilities. Specifically, the licensee failed to ensure that adequate emergency response facilities and equipment were available as required by the Brunswick Nuclear Plant Radiological Emergency Plan, Section 1.3.1.3 revision 80 and 10 CFR 50.47(b)(8). This issue was captured in the licensee's CAP as AR 542704.

The licensee's failure to properly evaluate or consider the impact to emergency response facilities of design change ESR98-00436 which was implemented in 1999 was a performance deficiency. Specifically, the licensee introduced a single point failure mode which did not meet the design requirements specified in their Design Basis Document (DBD 60) sections 3.6.7.2 and 3.6.7.3. This resulted in the licensee's failure to ensure that adequate emergency response facilities and equipment were available as delineated in the Updated Final Safety Analysis Report (UFSAR) Section 7.7.1.9, and required by the Brunswick Nuclear Plant Radiological Emergency Plan, Section 1.3.1.3, revision 80, and 10 CFR 50.47(b)(8). The finding was more than minor because it adversely affected the Emergency Preparedness Cornerstone objective of ensuring that the licensee was capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. Specifically, the Facilities and Equipment attribute was affected during the time when the ERFIS, ERDS, SPDS, and all displays including radiation monitors for the emergency response facilities were degraded, and as a result did not meet 10 CFR 50.47(b)(8) Planning Standard program element, adequate emergency facilities and equipment to support the emergency response are provided and maintained. The finding was assessed for significance in accordance with NRC IMC 0609, Appendix B Emergency Preparedness Significance Determination Process. Attachment 2 of Appendix B, Failure to Comply Significance Logic is as follows: Failure to comply; Loss of Risk Significant Planning Standard Function (RSPS), No; RSPS Degraded Function, No; Loss of Planning Standard Function, No; the result is a Green finding. The inspectors determined that this resulted in a very low safety significance finding (Green). No cross-cutting aspect was assigned to this finding because the performance deficiency occurred more than three years ago and is not reflective of current plant performance.

Inspection Report# : 2012004 *(pdf)*

Occupational Radiation Safety

Public Radiation Safety

Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

Miscellaneous

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