Brunswick 1
4Q/2010 Plant Inspection Findings

Initiating Events

Significance: G Jun 30, 2010
Identified By: Self-Revealing
Item Type: FIN Finding
Inadequate Design Change Results in an Automatic Reactor Scram
Green. A self-revealing Green finding was identified for an inadequate design change to the Unit 1 feedwater flow instrument sensing lines (Plant Modification (PM) 77-039). As a result of the inadequate design change, pressure pulsation dampeners (snubbers) were installed in the feed flow instrument sensing lines which prevented the instruments from detecting a loss of feed flow in time to prevent a reactor scram by initiating a recirculation pump runback. This was revealed after a loss of the 1B reactor feed pump (RFP) and a reactor low level scram on May 5, 2010. After the scram, the licensee adjusted the snubbers so that they respond properly to changes in feed flow and entered the issue into their corrective action program (AR #397712).

The inadequate design change implemented by PM 77-039 was a performance deficiency. The finding was more than minor because it was associated with the initiating events cornerstone attribute of design control, and it affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability and that challenge critical safety functions during shutdown, as well as during power operations. Specifically, the performance deficiency caused a reactor scram. Using the Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, inspectors concluded that the transient initiator did not contribute to both the likelihood of a reactor trip and to the likelihood that mitigation equipment or functions would not be available. As a result, the issue was of very low safety significance (Green). The cause of this finding has no cross-cutting aspect because the modification took place in 1977 and is not indicative of current licensee performance.

Inspection Report# : 2010003 *(pdf)*

Significance: G Mar 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Risk Evaluation for Removing the 1A South Condenser from Service
The inspectors identified a Green NCV of 10 CFR Part 50.65 (a)(4), Requirements for monitoring the Effectiveness of Maintenance at Nuclear Power Plants, after Unit 1 experienced a loss of normal reactor feedwater as a result of an abnormal plant configuration during shutdown of the reactor on February 26, 2010.
The licensee did not adequately manage the increase in risk that resulted when the 1B reactor feed pump (RFP) was made unavailable while the 1A south condenser was isolated in the hours leading up to the reactor shutdown. This plant configuration led to a high level in the 1A south condenser hotwell soon after the reactor shutdown, which prevented adequate draining of the 1A RFP turbine casing, and led to the loss of the 1A RFP. After the loss of normal feedwater to the reactor, the licensee restored reactor level using the reactor core isolation cooling (RCIC) system. The licensee entered the issue into its corrective action program (AR #383636).

The failure to adequately evaluate and manage risk associated with equipment configuration during the Unit 1 shutdown is a performance deficiency. This finding is more than minor because it is associated with the initiating events cornerstone attribute of configuration control and it adversely affected the cornerstone objective if limiting the likelihood of those events that upset plant stability and challenge critical safety functions. Specifically, plant stability was upset by the loss of normal feedwater to the reactor. In accordance with IMC 0609, Appendix K, Maintenance Risk Assessment and Risk Management Significance Determination Process, this finding is of very low safety significance (Green) because the Incremental Core Damage Probability Deficit is the licensee did not appropriately plan work activities by incorporating risk insights (H.3(a)). Specifically, activities scheduled prior to the reactor

shutdown were not properly evaluated to determine their impact on the normal reactor feedwater system.
Inspection Report# : 2010002 *(pdf)*

Mitigating Systems

Significance: G Mar 31, 2010
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to Follow Procedures During Reactor Head Disassembly
A self-revealing Green NCV of Technical Specifications (TS) 5.4.1, Procedures, was identified when reactor head piping was disconnected prior to swapping shutdown range reactor water level transmitters resulting in inaccurate water level indication. The plant procedure for disconnection of the reactor head piping, 0SMP-RPV501, Reactor Vessel Disassembly, used in conjunction with 0GP-06, Cold Shutdown to Refueling, specifies that prior to removal of head piping, the Shutdown Range Reactor Water Level Transmitters shall be swapped from level transmitters, B21-LT-N027A and B21-LT-N027B, to level transmitters, B21-LT-7468A
and B21-LT-7468B. Contrary to this requirement, the common reference leg to the level indicators was disconnected prior to swapping transmitters which resulted in loss of accurate indication of current reactor vessel water level. The licensee reinstalled the disconnected piping, refilled the reference legs for the transmitters, and entered the issue into their corrective action program (AR #383779).

The disconnection of the reference leg flange of the reactor vessel head piping prior to realignment of level instrumentation as required by plant procedures is a performance deficiency. The performance deficiency was more than minor because it is associated with the configuration control attribute of the Mitigating Systems
cornerstone because it inappropriately altered the reactor level instrumentation reference leg piping. It affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inaccurate level indication degraded the operator's
ability to control the reactor vessel water level in the prescribed procedural band and would inhibit their ability to diagnose and prevent loss of residual heat removal (RHR) scenario. In accordance with NRC Inspection Manual Chapter (IMC) 0609, Appendix G, "Shutdown Operations Significance Determination Process,"
Attachment 1, Checklist 8, the inspectors conducted a Phase 1 SDP screening and determined the finding required a Phase 2 analysis. The Phase 2 analysis determined the finding is of very low safety significance (Green) because adequate mitigation capability was maintained. The cause of this finding was directly related to the supervisory and management oversight cross-cutting aspect in the work practices component of the Human Performance cross-cutting area because plant supervisors failed to ensure an adequate pre-job brief, failed to enforce proper communications methods at the job site, and failed to properly supervise workers
executing procedure steps (H.4(c)).
Inspection Report# : 2010002 *(pdf)*

Barrier Integrity

Emergency Preparedness

Significance: W Sep 17, 2010
Identified By: NRC
Item Type: VIO Violation
Failure to timely augment on-shift staffing
An NRC-identified, low to moderate safety significance (White), apparent violation (AV) of 10 CFR 50.54(q) was identified in that the licensee failed to meet the requirements of 10 CFR 50.47(b)(2). The Technical Support Center

(TSC), Operations Support Center (OSC), and Emergency Operations Facility (EOF) were not activated until approximately two and one-half hours after the Alert declaration due to delays in the notification and response of the Brunswick emergency response organization (ERO).

10 CFR 50.54(q) requires that the facility shall follow and maintain in effect Emergency Plans which meet the standards in 10 CFR 50.47(b). 10 CFR 50.47(b)(2), states, “On-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available and the interfaces among various onsite response activities and offsite support and response activities are specified.” Brunswick Plant Emergency Procedures 0PEP-02.6.12, 0PEP-02.6.26, and 0PEP-02.6.27 require activation of the OSC, TSC and EOF respectively within 60 – 75 minutes following the declaration of an ALERT or higher emergency classification. Contrary to the above, on June 6, 2010, the Brunswick Steam Electric Plant ERO failed to provide initial facility accident response through timely augmentation of on-shift staffing after declaration of an alert at Brunswick. This resulted in the delay of OSC, TSC, and EOF activation by 75 minutes.

The licensee’s failure to maintain its emergency plan in effect is a performance deficiency and an apparent violation (AV) of 10 CFR 50.54(q). The cause of this finding was directly related to the cross-cutting aspect of, “The licensee conducts self-assessments at an appropriate frequency; such assessments are of sufficient depth, are comprehensive, are appropriately objective, and are self-critical. The licensee periodically assesses the effectiveness of oversight groups and programs such as CAP, and policies.” P.3(a)

Inspection Report# : 2010007 *(pdf)*
Inspection Report# : 2010010 *(pdf)*

Significance: G Sep 17, 2010
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to timely activate ERDS
A self-revealing, very low safety significance (Green), non-cited violation (NCV) of 10 CFR 50.72(a)(4) was identified. The Emergency Response Data System (ERDS) was not activated until 80 minutes after the Alert declaration due to a lack of on-shift staffing experience and inadequate procedural guidance.

10 CFR 50.72(a)(4), states, “The licensee shall activate the Emergency Response Data System (ERDS) as soon as possible but not later than one hour after declaring an Emergency Class of alert, site area emergency, or general emergency. The ERDS may also be activated by the licensee during emergency drills or exercises if the licensee's computer system has the capability to transmit the exercise data.” Contrary to the above, on June 6, 2010, the Brunswick ERO failed to activate the Emergency Response Data System within one hour after declaring an alert at the Brunswick Steam Electric Plant.

Inspection Report# : 2010007 *(pdf)*

Occupational Radiation Safety

Public Radiation Safety

Significance: G Mar 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Ensure Representative Sampling of Particulate Effluents Released from the Reactor Building Roof Vent

The inspectors identified a Green NCV of 10 CFR 20.1302(a) for failure to ensure surveys of particulate radioactive materials in effluents released to unrestricted areas from the reactor building roof vent were adequate to demonstrate compliance with dose limits for individual members of the public. This issue was initially identified as an unresolved item following an inspection in June 2008. The licensee entered the issue into its corrective action program (AR #292216 and AR #393340). The licensee is currently investigating this issue to identify applicable corrective actions.

The failure to ensure that the reactor building roof vent effluents were adequately monitored is a performance deficiency. This finding is more than minor because it is associated with the Public Radiation Safety cornerstone attribute of Plant Facilities/Equipment and Instrumentation (Process Radiation Monitors) and adversely affects the cornerstone objective. Specifically, the cornerstone objective of providing assurance that adequate protection of public health and safety from exposure to radioactive materials released into the public domain as a result of routine civilian reactor operation was affected because the licensee did not ensure that reactor building effluents were accurately monitored. The finding was evaluated using the Public Radiation Safety SDP and determined to be of very low safety significance (Green). The finding, which involved the effluent release program, was determined to be of very low safety significance (Green) because it was not a failure to implement the effluent program and did not result in public dose exceeding the 10 CFR 50 Appendix I criterion or 10 CFR 20.1301 (e). This finding does not have a cross-cutting aspect because the failure to evaluate the effect of line losses on particulate sampling is a historical issue.
Inspection Report# : 2010002 *(pdf)*

Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

Miscellaneous

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