Brunswick 1
1Q/2011 Plant Inspection Findings

Initiating Events

Significance: 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)*

Mitigating Systems

Significance: TBD Mar 31, 2011
Identified By: Self-Revealing
Item Type: AV Apparent Violation
Failure to adequately evaluate and correct a condition adverse to qualityinvolving a manufacturing defect of barto model 199 duel dampenern differential pressure units
A self-revealing Apparent Violation (AV) of 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action was identified for failure to promptly correct a condition adverse to quality regarding a manufacturing defect of a Barton Model 199 dual dampener differential pressure unit (DPU) used in the 1B residual heat removal (RHR) loop. Specifically, the licensee failed to replace the DPU after the vendor determined that the manufacturing process was incorrect and could lead to a slow response of the component in safety-related applications. This led to a failure of the RHR system 1B loop minimum flow bypass valve, 1-E11-F007B, to operate on February 18, 2011. The failure of the defective DPU was tracked as NCR 448471 in the corrective action program, and the licensee replaced the defective DPU.

The inspectors determined that the licensee's failure to promptly correct a condition adverse to quality regarding a manufacturing defect for Barton Model 199 dual dampener DPUs was a performance deficiency. The finding is more than minor because it is 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 (i.e. core damage). Specifically, the corrosion buildup in the DPU used in the control of the position of the minimum flow bypass valve for the 1B RHR loop had degraded, such that the availability and reliability of the 1B RHR loop was adversely affected. This finding was evaluated using Inspection Manual Chapter 0609, Significance Determination Process (SDP), Phase 1 Worksheet for mitigating systems. The finding required phase two and phase three SDP analyses by a regional senior analyst because the 1B loop of RHR was assumed to be inoperable for longer than its technical specification (TS) allowed outage time. The significance of this finding is designated as To Be Determined (TBD) until the safety characterization has been completed. This finding does not have a cross-cutting aspect because the performance deficiency occurred greater than three years ago and does not reflect current licensee performance. (Section 1R15)

Inspection Report# : 2011002 *(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# : 2010010 *(pdf)*
Inspection Report# : 2010007 *(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

Significance: G Mar 31, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to follow procedures for analyzing radiological air samples for the presence of alpha emitters
The inspectors identified a non-cited violation (NCV) of Technical Specification (TS) 5.4.1, Procedures, for the failure of the licensee to perform initial alpha activity analysis of air samples indicating greater than 0.3 Derived Air Concentration (DAC) beta-gamma activity on an approved alpha counter. Section 9.5.12.h of procedure HPS-NGGC-0024, Alpha Monitoring Guidelines, Rev. 3, states that if gamma scan results indicate the airborne activity is equal to or greater than the beta-gamma DAC-Fraction Action level of 0.3 DAC; (1) perform an initial alpha count on the air sample using a counter approved for air samples; and (2) assess and document the results per site-specific procedures. Contrary to this requirement, on March 10, 11, and 21, 2011, the licensee did not perform an initial alpha count on air samples using a counter approved for air samples and assess and document the results for gamma scan results that exceeded 0.3 DAC. Specifically, air samples for those selected work activities identified DAC concentrations of 0.6589, 0.3152 and 1.45. Licensee corrective actions included instructions to workers to ensure procedural adherence for sample analysis and changes to the software program to prompt the workers to do the sample analysis when the threshold limits were met or exceeded. The licensee entered the issue into its corrective action program as NCR 455307.

This finding is greater than minor because it was associated with the Occupational Radiation Safety Cornerstone attribute of Program and Process (Monitoring and Radiation Protection Controls) and adversely affects the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation from airborne radioactive material during routine civilian nuclear reactor operation. Failure to identify potentially significant contributors to internal dose could lead to unmonitored occupational exposures. The finding was evaluated using IMC 0609, Appendix C, “Occupational Radiation SDP” and was determined to be of very low safety significance (Green) because it was not related to As Low As Reasonably Achievable (ALARA) Planning and the ability to assess dose was not compromised during these instances. In addition, it did not involve overexposure or substantial potential for overexposure because of the relatively low alpha source term in the areas where the surveys were performed. This conclusion was drawn from the results of beta/gamma and alpha smear surveys performed at those selected work locations. However, if left uncorrected, unmonitored internal exposure could have occurred. The cause of this finding was directly related to the cross-cutting aspect of maintaining effective interfaces between work groups in the Work Control component of the Human Performance area. [H.3(b)]. (Section 2RS1)

Inspection Report# : 2011002 *(pdf)*

Public Radiation Safety

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

Last modified : June 07, 2011