Brunswick 1
1Q/2010 Plant Inspection Findings

Initiating Events

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)*

Significance: G Sep 30, 2009
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to Follow Plant Procedure Caused Loss of E2 Bus
A self-revealing Green non-cited violation of Technical Specification (TS) 5.4.1, Procedures, was identified when the licensee failed to follow procedure 0PICCNV023, Calibration of Westinghouse & Scientific Columbus Teleductors. During the performance of the calibration, procedural steps were not performed correctly and the E2 electrical bus was inadvertently deenergized, requiring the emergency diesel generator #2 to auto-start and reenergize the bus. Emergency diesel generator #2 auto-started and the E2 bus transferred from off-site power. After the event, the licensee halted the maintenance on the E2 bus instruments and restored off-site power to the E2 bus. The event was entered into the licensee's corrective action program as NCR #344300. The finding was determined to be more than minor because the finding was associated with the Initiating Events Cornerstone attribute of configuration control and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations.

The finding affected configuration control because correct test switch alignment was not maintained. The finding also affected the cornerstone objective because loss of the E2 bus represented an upset to plant stability. The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of findings," Table 4a for the Initiating Events Cornerstone. The finding was determined to be of very low safety significance (Green) because the finding was 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 0PIC-CNV023, Calibration of Westinghouse & Scientific Columbus Teleductors. Specifically, technicians did not utilize adequate error prevention techniques to prevent them from operating the wrong test switch when calibrating instrument 1-E2-AG6-VTR (H.4(a))
Inspection Report# : 2009004 *(pdf)*

Mitigating Systems

Significance: 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)*

Significance: Dec 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
Failure To Identify and Correct Degraded Fire Protection Sprinklers
The inspectors identified a Green non-cited violation of Brunswick Steam Electric Plant (BSEP) Unit 1 Updated Facility Operating License DPR-71, and the Unit 2 Updated Facility Operating License DPR-62, Condition 2.B.(6), for the licensee’s failure to identify and correct degraded fire suppression system sprinklers per the licensee’s fire protection program procedures. Procedure, 0PT-34.6.4.1, “Sprinkler And Spray System Visual Inspection: RX1, RX2, SW, RW, WT, and DG Buildings,” directs the licensee to verify the physical integrity of the spray and sprinkler piping and the absence of sprinkler obstruction or damage for the Unit 1 Reactor Building, Unit 2 Reactor Building,

Service Water Building, Radwaste Building, Water Treatment Building, and Diesel Generator Building. After NRC inspectors identified the degraded sprinklers, the licensee re-performed the procedure and identified 40 spray shields to be noncompliant with the procedure's acceptance criteria. Once identified, the licensee initiated compensatory fire watches. Corrective actions also included replacing or repairing the defective spray shields. This finding was entered into the licensee's corrective action program as NCR #357183.
Failure to follow procedure 0PT-34.6.4.1, "Sprinkler And Spray System Visual Inspection: RX1, RX2, SW, RW, WT, and DG Buildings" was a performance deficiency. The finding was determined to be more than minor because it affected the Mitigating Systems cornerstone objective of availability, reliability, and capability of the fixed fire suppression systems and was associated with the protection against external factors (fire) attribute. Specifically, this failure could affect the ability of the water sprinkler system to respond to a fire because the affected sprinklers' spray patterns are reduced and less effective. The issue was determined to be of very low safety significance (Green) using Manual Chapter (MC) 0609, Appendix F, Attachment 1, because the category of fixed fire suppression was evaluated as having low degradation. The system had low degradation because the sprinkler system is expected to display nearly the same level of effectiveness and reliability as it would, had the degradation not been present. The finding has a procedural compliance cross-cutting aspect in the Work Practices component of the Human Performance cross cutting area, because the licensee failed to ensure procedural instructions (procedure 0PT-34.6.4.1) were implemented correctly. H.4(b)
Inspection Report# : 2009005 *(pdf)*

Significance: G Dec 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
Failure To Adequately Implement Design Control Measures For The Fire Protection Program
The inspectors identified a Green non-cited violation of BSEP Unit 1 Updated Facility Operating License DPR-71, and the Unit 2 Updated Facility Operating License DPR-62, Condition 2.B.(6), for the licensee's failure to implement adequate design control measures for the fire protection program. Plant drawings which specify the configuration of fire suppression sprinklers are inconsistent and inadequate in that they do not provide complete details for sprinkler spray shields. Dimensions for spray shields on some drawings are incomplete because they don't list all of the necessary critical dimensions. Therefore, some ceiling-level spray shields were incorrectly installed and extended below the sprinklers' fusible links. This would have delayed sprinkler response in a fire. After the identification of this design control issue, the licensee implemented corrective actions which included repairing or replacing the degraded sprinklers. This finding was entered into the licensee's corrective action program as NCR #367339.

The licensee's failure to adequately implement design control measures for the fire protection program as required by the operating license (condition 2.B(6)) was a performance deficiency. The finding was determined to be more than minor because it affected the Mitigating Systems cornerstone objective of availability, reliability, and capability of the fixed fire suppression systems and was associated with the design control and protection against external factors (fire) attribute. Specifically, this failure could affect the ability of the water sprinkler system to respond to a fire because the incorrectly installed spray shields delay the ceiling-level sprinklers' response times. The issue was determined to be of very low safety significance (Green) using MC 0609, Appendix F, Attachment 1, because the category of fixed fire suppression was evaluated as having low degradation. The system had low degradation because the sprinkler system is expected to display nearly the same level of effectiveness and reliability as it would, had the degradation not been present. This finding has no cross-cutting aspect because the design drawing deficiency occurred when the plants were licensed and it is not indicative of current licensee performance.
Inspection Report# : 2009005 *(pdf)*

Significance: G Dec 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequately Monitored Maintenance Rule MOV
The inspectors identified a Green NCV of 10 CFR 50.65(a)(1), Requirements for monitoring the effectiveness of maintenance at nuclear power plants, for the licensee's failure to monitor the performance or condition of motor-operated valve (MOV) MS-V28 in a manner sufficient to provide reasonable assurance that it was capable of fulfilling its intended functions. As a result, the licensee did not recognize that the valve was incapable of opening against design differential pressure and failed to take appropriate corrective actions to ensure that the valve could fulfill its

emergency operating procedure (EOP) function. After the issue was identified, the licensee altered its operating procedures to compensate for the valve not opening against design differential pressure and entered it into their corrective action program (AR #356800).
The failure to adequately monitor the performance or condition of MOV MS-V28 in a manner to provide reasonable assurance that the valve was capable of fulfilling its intended function is a performance deficiency. The performance deficiency was more than minor because it is associated with the Mitigating Systems cornerstone attribute of equipment performance, and 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 performance deficiency affected the reliability of the MOV MS-V-28 and its use in EOPs to restore feed to the reactor. Inspectors evaluated the finding using NRC IMC 0609, Significance Determination Process, Appendix A. Since the finding represents an actual loss of a function of non-Technical Specifications equipment designated as risk-significant for greater than 24 hours, the finding required a phase two significance analysis. The Brunswick phase 2 SDP spreadsheet indicated that the finding was greater than green but did not detail to the cases requiring MS-V28 operation therefore a phase 3 SDP analysis was completed by a regional SRA.
The phase 3 SDP analysis was performed in accordance with NRC Inspection Manual Chapter 0609 appendix A utilizing the NRC SPAR model and output from the licensee's full scope PRA model. The result was a risk increase for the finding of <1E-6 for core damage frequency (cdf) and <1E-7 for large early release frequency (LERF). The dominant sequences were transient initiators with spurious level instrument generated main steam isolation valve (MSIV) closure and the inability to restore main feedwater due to the performance deficiency coupled with failure to achieve successful depressurization and use of low pressure makeup systems leading to core damage. The risk was mitigated by the low initiating event frequency for transient conditions which would allow MSIV reopening and recovery of main feedwater. The availability of low pressure injection systems was also a factor reducing the risk. The result of the phase 3 analysis was that the finding was characterized as having very low safety significance, a Green finding. The cause of this finding was directly related to the problem evaluation cross-cutting aspect in the corrective action program component of the Problem Identification and Resolution cross-cutting area because the licensee failed to adequately evaluate the failure of MS-V28 in November 2008. (P.1(c)).

Inspection Report# : 2009005 *(pdf)*

Significance: G Oct 20, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Maintenance Instructions for Maintaining Cleanliness During Emergency Diesel Generator Governor Maintenance
Green. The inspectors identified a Green NCV of Technical Specification (TS) 5.4.1, Procedures, for inadequate cleanliness and flushing procedures for maintaining cleanliness during maintenance on the emergency diesel generator (EDG) governors. This procedural inadequacy resulted in a failure of the emergency diesel generator #4 governor on September 19, 2009. The licensee entered the issue into their corrective action program and replaced the failed governor.

The finding was determined to be more than minor because it is associated with equipment performance and procedure quality attributes of the Mitigating Systems Cornerstone. It also 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 reliability of EDG #4 was reduced because it was susceptible to fouling due to the foreign material in the governor. The finding was evaluated for significance using NRC Manual Chapter 0609, Appendix A, Determining the Significance of Reactor Inspection Findings for At-Power Situations. Using Table 4a of Appendix A to MC 0609, the finding was determined to be of very low safety significance (Green) because the failure of EDG #4 did not represent a loss of safety function, did not represent a loss of EDG #4 operability for greater than its technical specification allowed outage time, and does not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding affects the cross-cutting area of human performance, resources component,
complete and accurate documentation aspect because the licensee did not incorporate adequate guidance for maintaining cleanliness of the EDG governor in their maintenance procedures. (H.2(c))
Inspection Report# : 2009011 *(pdf)*

Significance: G Sep 30, 2009
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Surveillance Test Performed on Incorrect Loop of RHR
A self-revealing Green non-cited violation of TS 5.4.1, Procedures, was identified when the licensee failed to follow work order instructions contained in work order 1280322. This work order directed technicians to perform testing on the B loop of the Unit 1 residual heat removal (RHR) system according to procedure
1MST-RHR28R, RHR Time Delay Relay Channel Calibration. Contrary to these work order instructions, portions of the procedure affecting Loop A were performed instead of Loop B. After the technicians completed the A loop section of the procedure, they reported to the control room where operators recognized the error.
Once the error was recognized, the maintenance was stopped and B loop of RHR was returned to operable. This finding was entered into the licensee’s corrective action program as NCR #344233.

The finding was determined to be more than minor because the finding was associated with the Mitigating Systems Cornerstone attribute of configuration control and affected the cornerstone objective of to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable
consequences. Specifically, as a result of this error on the Loop A RHR relay channels, for a short time, safety interlocks were bypassed on both the low pressure injection coolant (LPCI) outboard injection valve and the RHR heat exchanger bypass valve, and the position of the RHR pump minimum flow bypass valve was
changed out of its normal position. The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase 1 - Initial Screening and Characterization of findings,” Table 4a for the Mitigating Systems Cornerstone. The finding was
determined to be of very low safety significance (Green) because the finding was not a design or qualification deficiency which resulted in loss of operability or functionality, did not represent a loss of system safety function, did not represent an actual loss of safety function of a single train for greater than its TS allowed outage time, and did not represent potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding has a cross-cutting aspect in the Human Performance cross cutting area, Work Practices component, because the licensee failed to ensure surveillance instructions (work order 1280322) were implemented correctly. This resulted in performing a surveillance test on the A loop of the RHR
system while the B loop of the RHR system was disabled (H.4(b))
Inspection Report# : 2009004 *(pdf)*

Significance: G Sep 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Establish Adequate Installation Instructions for Emergency Diesel Generator Service Water Expansion Joint Control Units
The inspectors identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the failure to specify an appropriate quality standard for the installation of the control units on the emergency diesel generator jacket water heat exchanger inlet and outlet expansion joints. As a result,
threaded fasteners on emergency diesel generators #1 and #4 loosened, creating a potential vulnerability to expansion joint failure. The licensee tightened the control unit bolts on all the emergency diesel generator service water expansion joints and initiated an engineering change to prevent the fasteners from loosening. This finding was entered into the licensee’s corrective action program as NCR #346113.

The finding was determined to be more than minor because the finding, if left uncorrected, would have the potential to lead to a more significant safety concern. Specifically, over time, the hex nuts on the expansion joint control units could loosen to the point of expansion joint failure, leading to a loss of service water to the
emergency diesel generators and failure of the emergency diesel generators. The inspectors evaluated the finding in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase 1 - Initial Screening and Characterization of findings,” Table 4a for the Mitigating Systems Cornerstone. The
finding was determined to be of very low safety significance (Green) because the finding was a design or qualification deficiency confirmed not to result in loss of operability or functionality. This finding has no cross-cutting aspect because the design deficiency occurred in 2005 and is not indicative of current licensee
performance.

Inspection Report# : 2009004 *(pdf)*

Significance: Sep 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Include Risk Significant Maintenance in the Site Risk Profile
The inspectors identified a Green non-cited violation of 10 CFR Part 50.65 (a)(4), when the licensee removed the severe accident mitigation guideline (SAMG) diesel generators from service without considering the change in the online plant risk. Online plant risk is modeled and communicated to licensee plant personnel via the equipment out of service (EOOS) profile. The change in online risk was not reflected in the EOOS profile when the SAMG diesel generators were out of service from July 6, 2009 to July 8, 2009. Once the deficiency was identified on July 8, 2009, the EOOS profile was updated by the licensee and reflected the SAMG diesel out of service condition. This finding was entered into the licensee’s corrective action program as NCR #351002.

The finding was determined to be more than minor because the finding related to maintenance risk assessment and risk management issues. Specifically, the licensee’s risk assessment failed to consider risk significant structures, systems, or components that were unavailable during maintenance. The inspectors evaluated the finding in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase 1 - Initial Screening and Characterization of findings,” Table 3a for the Mitigating Systems Cornerstone. The finding was determined to degrade the licensee’s assessment and management of risk associated with performing maintenance activities under all plant operation or shutdown conditions. In accordance with Baseline Inspection Procedure (IP) 71111.13, “Maintenance Risk Assessment and Emergent Work Control,” and IMC 0609, Appendix K, "Maintenance Risk Assessment and Risk Management Significance Determination Process," the finding was determined to be a maintenance risk assessment issue. Flowchart 1, "Assessment of Risk Deficit," requires the inspectors to determine the risk deficit associated with this issue. The finding was determined to be of very low safety significance because the incremental core damage probability deficit was less than 1 x 10E-6. The regional senior reactor analyst reviewed the information and confirmed that the system was a maintenance rule safety significant system. This finding has a cross-cutting aspect in the area of human performance, work control component, because the licensee did not plan and coordinate work activities consistent with nuclear safety. Specifically, the licensee failed to include risk significant maintenance in the EOOS profile when the SAMG diesel generators were out of service from July 6, 2009 until July 8, 2009 (H.3(a))
Inspection Report# : 2009004 *(pdf)*

Significance: Sep 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
Capability of Emergency Diesel Generator Ventilation System to Meet Design and Licensing Requirements
The inspectors identified a Green non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion III, Design Control, for failure to translate a key analytical assumption related to operation of the emergency diesel building back draft and check dampers into specifications and ultimately into the installed hardware. This issue was entered into the licensee’s corrective action program as NCR 00259088 with actions to evaluate the ability of the EDGs actual installed equipment to satisfy the intended safety function during and following the design basis tornado event. Compensatory measures were established to eliminate the concern pending the licensee’s determination of the systems capability to mitigate the effects of a tornado event.

This finding was determined to be more than minor because the finding was associated with the Mitigating Systems Cornerstone attribute of Design Control, i.e. initial design. It impacted the cornerstone objective of ensuring the availability, reliability, and capability of the emergency diesel building ventilation to protect the EDG building structure during a design basis tornado event. Due to the deficiencies between the installed hardware and the assumptions in the calculation, the calculation did not ensure the capability of emergency diesel building ventilation system to perform the safety function. This was determined to be a failure to ensure the availability, reliability, and capability of a safety system that responds to an initiating event to prevent undesirable consequences. The licensee subsequently determined from analysis through modeling and testing that the emergency diesel building ventilation system could perform the safety function during a design basis tornado event with the existing hardware installed. The NRC reviewed this analysis and the results that determined

that the existing condition did not result in the loss of the system safety function. The inspectors assessed the finding using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of findings," Table 4a for the Mitigating Systems Cornerstone. The finding was determined to be of very low safety significance (Green) because there was not an actual loss of safety system function based upon the inspector's verification of the Progress Energy analysis of the emergency diesel building ventilation system. The cause of the finding is not related to a cross-cutting aspect because the occurrence was greater than three years ago and is not indicative of current licensee performance.
Inspection Report# : 2009004 *(pdf)*

Significance: W May 28, 2009
Identified By: NRC
Item Type: VIO Violation
Inability to Operate the EDGs Locally as Required by the Safe Shutdown Analysis Report
A violation of 10 CFR 50, Appendix B, Criterion III, Design Control, was identified for failure to correctly translate the design basis into EC 66274 to replace control relays on all four EDGs. Specifically, termination points for linking control power to the EDG lockout relay reset circuitry were incorrectly designated in the EC. This resulted in the wiring for control relays being installed such that the EDGs could not be operated locally as required by the Safe Shutdown Analysis Report. Upon discovery, the licensee initiated Action Request (AR) 292232 and re-wired and tested each affected EDG. The local control function was restored to all EDGs on August 21, 2008.

The failure to correctly translate the design basis into EC66274 is a performance deficiency. This finding is more than minor because it is associated with the reactor safety mitigating system cornerstone attribute of protection against external events, i.e., fire. It also affects the cornerstone objective of ensuring the availability of systems that respond to events in that the EDGs could not be operated locally as required by the Safe Shutdown Analysis Report. This finding was assessed using the applicable SDP, which resulted in a calculated core damage frequency (CDF) risk increase over the base case between 1E-5 and 1E-6 per year. The dominant accident sequences involved are initiated by a fire situated such as to cause both a loss of offsite power (LOOP) and a forced main control room evacuation. For these dominant accident sequences, the performance deficiency will result in a station blackout (SBO) to either or both units. The exposure period for this condition was one year. As a result, the finding was preliminarily determined to be of low to moderate safety significance (White). The cause of the finding is considered to have a cross-cutting aspect related to accurate design documentation [H.2(c)], as described in the resources component of the human performance cross-cutting area.

Inspection Report# : 2010006 *(pdf)*
Inspection Report# : 2009009 *(pdf)*
Inspection Report# : 2009010 *(pdf)*

Barrier Integrity

Emergency Preparedness

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

Significance: N/A May 08, 2009
Identified By: NRC
Item Type: FIN Finding
Brunswick PI&R Summary
The inspection team concluded that, in general, problems were adequately identified, prioritized, and evaluated; and effective corrective actions were implemented. Site management was actively involved in the corrective action program (CAP) and focused appropriate attention on significant plant issues. The team found that employees were encouraged by management to initiate ARs to address plant issues.

The licensee was effective at identifying problems and entering them into the CAP for resolution, as evidenced by the relatively few deficiencies identified by the NRC that had not been previously identified by the licensee during the review period. The threshold for initiating action requests (ARs) was appropriately low, as evidenced by the type of problems identified and large number of ARs entered annually into the CAP. Action requests normally provided complete and accurate characterization of the problem. However, the team identified two minor equipment issues during system walkdowns involving selected risk-significant safety-related systems, which were not already entered into the CAP.

Generally, prioritization and evaluation of issues were adequate consistent with the licensee's CAP guidance. Formal root cause evaluations for significant problems were adequate, and corrective actions specified for problems did address the cause of the problems. The age and extensions for completing evaluations were closely monitored by plant management, both for high priority nuclear condition reports (NCRs), as well as for adverse conditions of less significant priority. Also, the technical adequacy and depth of evaluations (e.g., root cause investigations) were typically adequate. However, the team identified a minor issue associated with the problem evaluation of a risk significant system, which could have resulted in unresolved issues with incomplete corrective actions.

Corrective actions were generally effective, timely, and commensurate with the safety significance of the issues. However, the team identified two minor issues associated with inadequate and untimely corrective actions that allowed potential unresolved conditions adverse to quality to remain uncorrected involving degraded equipment performance. This example of inadequate corrective actions did not represent a significant safety concern but reflected a lack of attention to detail in the implementation of corrective actions and preventive maintenance activities.

The operating experience program was effective in screening operating experience for applicability to the plant, entering items determined to be applicable into the CAP, and taking adequate corrective actions to address the issues. External and internal operating experience was adequately utilized and considered as part of formal root cause evaluations for supporting the development of lessons learned and corrective actions for CAP issues. However, the team identified an example where a Significant Adverse Condition Investigation report did not evaluate the applicable operating experience as directed by the licensee's investigation procedure.

The licensee's audits and self-assessments were critical and effective in identifying issues and entering them into the corrective action program. These audits and assessments identified issues similar to those identified by the NRC with respect to the effectiveness of the CAP.

Based on general discussions with licensee employees during the inspection, targeted interviews with plant personnel, and reviews of selected employee concerns records, the inspectors determined that personnel at the site felt free to raise safety concerns to management and use the CAP as well as the employee concerns program to resolve those concerns.

Inspection Report# : 2009006 *(pdf)*

Last modified : May 26, 2010