

# Brunswick 1

## 3Q/2010 Plant Inspection Findings

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### 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*)

**Significance:**  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 of 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*)

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## 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, OSMP-RPV501, Reactor Vessel Disassembly, used in conjunction with OGP-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, OPT-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 OPT-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 OPT-34.6.4.1) were implemented correctly. H.4(b)

Inspection Report# : [2009005](#) (pdf)

**Significance:**  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:**  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:**  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)

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## **Barrier Integrity**

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# Emergency Preparedness

**Significance:** **W** Sep 17, 2010

Identified By: NRC

Item Type: AV Apparent 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 OPEP-02.6.12, OPEP-02.6.26, and OPEP-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*)

**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*)

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# Occupational Radiation Safety

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# Public Radiation Safety

**Significance:**  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*)

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## **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.

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## **Miscellaneous**

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