# Beaver Valley 1
# 1Q/2007 Plant Inspection Findings

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## Initiating Events

**Significance:** G Aug 18, 2006
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**HOT WORK RESULTS IN FIRE IN UNIT 1 WEST CABLE VAULT**
A self-revealing non-cited violation (NCV) of License Condition DPR-66 Section 2.C.5, Fire Protection Program, was identified for failure to follow plant fire protection procedures related to hot work and ignition control. On August 18, 2006, failure to assess all fire hazards and remove or protect combustible items in the vicinity of hot work resulted in welding activities in the PCA Shop igniting transient combustible material, subsequently igniting plastic sheeting and causing a small class 'A' fire in the adjacent West Cable Vault. The licensee immediately extinguished the fire and stopped all hot work. The event was entered into the licensee's corrective action program (CR-06-04924). A root cause evaluation was initiated by the licensee.

The finding is more than minor because it had a direct impact on the Initiating Events cornerstone objective and could be viewed as a precursor to a more significant event if left uncorrected. Specifically, the licensee's performance deficiency was directly responsible for a Class 'A' fire in the Unit 1 safety-related West Cable Vault of the Safeguards Building. The finding is of very low safety significance because all other normally required fire prevention measures were in place, allowing the fire to be quickly detected and suppressed. No safety-related equipment was affected. The inspectors determined that a contributor of this finding was related to the work practice component of the cross-cutting area of human performance.
Inspection Report# : 2006004 *(pdf)*

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## Mitigating Systems

**Significance:** G Mar 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to effect timely and adequate corrective actions related to deficiencies in a safety-related river water valve pit.**
The inspectors identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, for inadequate and untimely corrective actions regarding deficiencies in a safety-related river water valve pit at Unit 1. Specifically, the NRC identified that FENOC had performed inadequate inspections of the valve pit in February 2006, as evidenced by a recent inspection that revealed an unsealed penetration between two halves of the pit that contain the 'A' and 'B' headers of the river water system. FENOC subsequently utilized the corrective action program, inspected the valve pit, identified additional deficiencies, and aggressively evaluated and dispositioned specific deficiencies based on significance.

The inspectors determined that this finding is more than minor because it impacted the external factors attribute regarding the availability and reliability of the river water system, and the capability to respond to initiating events and prevent undesirable consequences. The inspectors determined that this finding is of very low safety significance, because there was no loss of system or overall function due to the remaining mitigating equipment capability. This finding has a cross-cutting aspect in the area of problem identification and resolution, in that FENOC did not properly identify quality issues completely, accurately, and in a timely manner commensurate with their safety significance, when deficiencies in the valve pit were not identified in a February 2006 inspection [P.1(a)].
Inspection Report# : 2007002 *(pdf)*

**Significance:** G Jul 19, 2006
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**INADEQUATE CORRECTIVE ACTION TO RESOLVE SLEEVE BEARING SET SCREW POSITION**
A self-revealing, non-cited violation of 10 CFR 50,Appendix B, Criterion XVI, “Corrective Action,” was identified on July 17, 2006, when the Unit 1 ‘3B’ motor-driven auxiliary feedwater (MDAFW) pump [1FW-P-3B] inboard motor bearing oil was sampled and determined to contain babbit (CR-06-04345). The finding was determined to be inadequate problem evaluation and resolution of a prior sleeve-type journal bearing failure, caused by improper positioning of bearing housing set screws, and resulted in recurrent bearing failures of the ‘3B’ MDAFW pump motor. Specifically, corrective actions for a prior failure of a similar bearing did not adequately resolve the proper positioning of the bearing housing set screws, thereby preventing proper bearing alignment within the bearing housing. The licensee has performed a root cause evaluation, has determined proper positioning of the bearing housing set screws, and has performed an extent of condition review for other pump motors with sleeve-type journal bearings.

This finding is more than minor because it is associated with the equipment performance attribute of the mitigating systems cornerstone and affects the objective to ensure the availability of systems that respond to initiating events to prevent undesirable consequences. The finding is of very low safety significance because the finding does not represent an actual loss of safety function. The finding is related to the corrective action program component of the problem identification and resolution cross cutting area in that the bearing set screw position was not thoroughly evaluated and resolved.
Inspection Report# : 2006004 *(pdf)*

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# Barrier Integrity

**Significance:** G Jun 30, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**FAILURE TO PERFORM AN ADEQUATE OPERABILITY DETERMINATION FOR CURRENT LEAKAGE PAST MAIN STEAM SAFETY, DUMP AND RESIDUAL RELEASE VALVES**
The inspectors identified an NCV of 10 CFR 50, Appendix B, Criterion 16, for failure to perform an adequate operability evaluations for degraded components to assure off-site dose consequences are bounded in the radiological safety analysis for a SGTR event. Specifically, some barrier integrity components (HCV-1MS-104 and 1MS-26, Atmospheric Steam Dump Valve(s), and Steam Generator Safety Valves) were degraded (leaking) and FENOC did not quantify and evaluate the current leakage regarding additional radiological dose consequences during a design basis accident (SGTR event). The licensee entered this deficiency into their corrective action program and implemented corrective actions to assess the magnitude of additional steam leakage that would be permitted before licensing basis dose results are exceeded.

This finding is more than minor because it was associated with the SSC and Barrier Performance Attribute of the barrier integrity cornerstone and affected the objective of providing reasonable assurance that the physical design barrier (containment) protected the public from radio nuclide releases caused by accidents or events (SGTR). The finding is of very low safety significance because although degraded, the leaking residual heat release valve and other components (e.g., safety valves and atmospheric dump valves) are not important to LERF and do not affect CDF. The cause of this finding is related to the corrective action program component of the PI&R cross-cutting area, in that a degraded component was not adequately evaluated to assure proper operability was determined.
Inspection Report# : 2006003 *(pdf)*

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# Emergency Preparedness

**Significance:** W Aug 22, 2006
Identified By: NRC
Item Type: VIO Violation

**Inadequate dose assessment procedure.**
During an NRC inspection conducted between June 26 and August 22, 2006, for which an exit meeting was held on August 22, 2006, and a Regulatory Conference was held on October 24, 2006, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

10 CFR Part 50.47(b)(9), requires, in part, that adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.

Contrary to the above, as of June 27, 2006, the Beaver Valley Power Station (BVPS) method for assessing actual and potential offsite consequences of a radiological emergency condition was inadequate. Specifically, "BVPS Procedure 1/2-EPP-IP-2.6.3, Dose Projection - ARERAS/MIDAS With Real-Time Inputs, revision 13," stated that "IF the duration of a release cannot be estimated, THEN use 1.0 hour, and repeat the projection as better data become available." This is inadequate because the one hour default release duration may not adequately envelope existing plant conditions and therefore could lead to untimely protective action recommendations. As an example, during the June 27, 2006, emergency preparedness exercise, the licensee used one hour as a release duration time, even though the release had been occurring for at least 30 minutes and there was no reason to conclude the release would be terminated within one hour of initiation.

This violation is associated with a White Significance Determination Process finding.
Inspection Report# : 2006009 *(pdf)*

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## Occupational Radiation Safety

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## Public Radiation Safety

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## Physical Protection

Physical Protection information not publicly available.

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## Miscellaneous

Last modified : June 01, 2007