# Davis-Besse
# 4Q/2006 Plant Inspection Findings

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## Initiating Events

**Significance:** G Sep 30, 2006
Identified By: NRC
Item Type: FIN Finding
**LICENSEE ACTIONS INEFFECTIVE IN ENSURING MATERIAL AROUND SWITCHYARD AND UNDER POWER LINES PROPERLY STORED FOR HIGH WIND CONDITIONS**
A finding of very low safety significance was identified by the inspectors for the failure to control loose materials located adjacent to the switchyard and under power lines from the switchyard to the station's large power transformers. This material could be carried into the switchyard or the power lines by high winds. Once identified, the licensee took action to relocate the material. The issue was more than minor because, if left uncorrected, the prolonged presence of loose items located immediately adjacent to the switchyard increased the risk of an adverse impact to the proper operation of the switchyard and power lines and therefore could become a more significant safety concern. The issue was of very low safety significance because the finding did not contribute to the likelihood of a primary or secondary system loss of coolant accident initiator; the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available; and the finding did not increase the likelihood of a fire or internal or external flooding. The issue was not considered a violation of regulatory requirements because it did not affect safety-related structures, systems, or components. This finding was similar to a previous finding and the cause was related to the cross-cutting area of problem identification and resolution in that corrective actions were not effectively implemented.
Inspection Report# : 2006004 *(pdf)*

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## Mitigating Systems

**Significance:** G Dec 31, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**IMPROPER EVALUATION OF PLANT RISK**
A finding of very low safety significance and an associated NCV of 10 CFR 50.65(a)(4) was identified by the inspectors when the licensee failed to properly evaluate plant risk during station blackout diesel generator (SBODG) maintenance activities during the week of October 8, 2006. Probability risk assessment engineers were aware in December 2005 that the risk associated with the unavailability of an emergency diesel generator (EDG) or an SBODG had been revised from Green to Yellow. However, licensee personnel failed to update the probabilistic risk assessment model or risk profile program used for risk determination to reflect the revised risk level for the SBODG, although the risk program had been updated for other components. Consequently, during SBODG maintenance activities during the week of October 8, 2006, plant risk was treated as Green when it was actually Yellow and compensatory actions to address this increase in risk were not implemented, as required. As part of the licensee's immediate corrective actions, licensee personnel updated the risk profile to properly reflect the risk associated with the unavailability of an EDG and SBODG. The finding was more than minor because the finding was related to a licensee risk assessment that contained incorrect assumptions that had the potential to change the outcome of the assessment. The finding was determined to be of very low safety significance because, although the SBODG was unavailable, the remaining EDGs could have performed their safety function in the event of a loss of offsite power. The inspectors also determined that the cause of the finding was related to the cross-cutting area of human performance because licensee personnel failed to communicate decisions and the basis for decisions to personnel who had a need to know that information in a timely manner.
Inspection Report# : 2006005 *(pdf)*

**Significance:** G Jun 30, 2006
Identified By: Self-Revealing
Item Type: FIN Finding
**UNPLANNED HEATUP OF RCS DUE TO LOSS OF COMPONENT COOLING WATER TO IN-SERVICE DECAY HEAT COOLER DURING MODE 5**
A finding of very low safety significance was self-revealed when, with the plant shut down for a planned refueling outage, an uncontrolled 10 degree Fahrenheit heatup of the reactor coolant system occurred over a period of approximately 1 hour. The licensee had remotely closed a degraded air-operated valve to isolate cooling water flow to the in-service decay heat cooler to control plant heatup. However, because the valve was degraded, it could not be remotely opened to control the heatup. The onshift operating crew was unaware that the valve had been identified as degraded and inoperable by a previous operating crew. The licensee restored cooling by manually opening the valve and generated a condition report to enter the issue into its corrective action program. No violation of regulatory requirements occurred. The finding was more than minor because it was associated with the configuration control 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. The finding was determined to be of very low safety significance because: 1) the finding did not increase the likelihood of a loss of reactor coolant system inventory; 2) the finding did not degrade the licensee's ability to terminate a leak path or add reactor coolant system inventory when needed; and 3) the finding did not significantly degrade the licensee's ability to recover decay heat removal once it was lost. The primary cause of this finding was related to the cross-cutting area of Human Performance.
Inspection Report# : 2006003 *(pdf)*

**Significance:** G Jun 30, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**INOPERABILITY OF EDG2 DUE TO UNTIMELY CORRECTIVE ACTION TO FIND AND REPAIR EXHAUST VALVE ROCKER ARM DAMAGE**
A finding of very low safety significance and an associated Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," was identified by the inspectors when the licensee failed to investigate and correct an abnormal noise associated with emergency diesel generator 2 (EDG2) in a timely manner. As a result, EDG2 was inoperable in excess of the time limits prescribed by the associated Technical Specification (TS) Limiting Condition for Operation (LCO). The licensee repaired EDG2 and generated a condition report to enter this issue into its corrective action program. The finding was more than minor because it was 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. The finding was determined to be of very low safety significance because the change in core damage frequency calculated through an SDP Phase 3 analysis was less than 1E-6. The primary cause of the finding was related to the cross-cutting area of Human Performance.
Inspection Report# : 2006003 *(pdf)*

**Significance:** G Jun 30, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**STARTUP AND BUS TIE TRANSFORMER FAST TRANSFER CAPABILITY NOT REFLECTED IN SR 4.8.1.1.1.b**
The inspectors identified a finding of very low safety significance and a Severity Level IV Non-Cited Violation when the licensee failed to adhere to the requirements of TS Section 6.17, "Technical Specifications Bases Control Program." By adding TS Bases wording, Surveillance Requirement 4.8.1.1.1.b was reinterpreted to include testing of the fast transfer for the 13.8 kV bus tie transformers. This allowed the licensee to credit the fast transfer capability for offsite power operability without submitting a license amendment for NRC review. Because the issue affected the NRC's ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. The finding was determined to be more than minor because the TS Bases change would have required a license amendment and because the finding affected equipment important to safety. The finding was determined to be of very low safety significance because: 1) it did not represent an actual loss of safety function of a system; 2) it did not represent an actual loss of safety function of a single train for greater than its TS allowed outage time; 3) it did not represent an actual loss of safety function of one or more non-TS trains of equipment designated as risk-significant per 10 CFR 50.65 for greater than 24 hours; and 4) it did not screen as

potentially risk significant due to a seismic, fire, flooding, or severe weather initiating event.
Inspection Report# : 2006003 *(pdf)*

**Significance:** Mar 31, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**FAILURE TO EXPAND CODE WELD EXAMINATION SCOPE**
The inspectors identified a finding involving a Non-Cited Violation (NCV) violation of 10 CFR 50.55a(g)4 having very low safety significance for failure to expand the scope of weld examinations after identification of a rejectable flaw in the SG 1-2 main steam nozzle-to-shell weld. As a corrective action, the licensee performed the additional weld examination and entered this issue into the corrective action program. This finding is of more than minor significance because it is associated with the Mitigating System cornerstone attribute of "Equipment Performance" and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to perform additional examinations prior to returning the plant to service in April of 2000, placed the plant at an increased risk for operation with undetected cracking which can lead to component failure. This finding is of very low safety significance because the licensee subsequently performed additional magnetic particle examinations during the 2006 refueling outage with no rejectable indications. The inspectors determined that the finding was not suitable for SDP evaluation because the failure to expand the scope did not directly result in degraded or inoperable equipment. Therefore, this finding was reviewed by Regional Management, in accordance with IMC 0612 Section 05.04c, and determined to be of very low safety significance.
Inspection Report# : 2006002 *(pdf)*

**Significance:** Mar 02, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Non-Conservative Flow Testing Acceptance Criteria for the Number 2 HPI and Number 2 LPI Pumps**
The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," that was of very low safety significance. Specifically, the flow acceptance curves used for two safety-related pumps, the number 2 High Pressure Injection (HPI) pump and the number 2 Low Pressure Injection (LPI) pump, were incorrect and non-conservative. This issue was entered into the licensee's corrective action system and the licensee verified that other safety-related pumps were not similarity affected.

The issue was more than minor because it was associated with the Mitigating System cornerstone attribute of "Design Control," and affected the cornerstone objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, performing required flow testing of safety related pumps with non-conservative acceptance criteria could allow the pumps to operate during an accident with less than acceptable flows. The finding screened as having very low significance (Green) using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for the At-Power Situations," because the inspectors answered "no" to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. While the failure to apply conservative test acceptance criteria during flow tests for the HPI and LPI pumps could have eventually caused the pumps to be outside of their acceptable flow rates without it being detected, the licensee verified that the most recent test data for these pumps would have still been acceptable if the correct pump curves had been used.
Inspection Report# : 2006006 *(pdf)*

**Significance:** Mar 02, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Consider Adverse Ampacity Effects of High Temperature Conditions in the Diesel Fire Pump Room**
The inspections identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," that was of very low safety significance, where the licensee had not evaluated and updated the plant cable ampacity calculation to determine the potential consequences of adverse effects to cabling due to higher temperatures in the Diesel Fire Pump Room and other areas in the plant. The issue was entered into the licensee's corrective action program.

The finding was more than minor because it affected the mitigating system cornerstone attribute of "Design Control" and

affected the cornerstone objective of ensuring reliability of systems that respond to initialing events to prevent undesirable consequences. Specifically, the licensee did not account for high temperature conditions that adversely affected the ampacity of cabling supplying power to equipment important to safety. This finding was of very low safety significance, because the inspectors answered “no” to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. Specifically, the licensee was able to demonstrate that even though the higher temperatures decreased the ampacity margins for the effected cabling, it did not decrease the margins to the limit where the cabling would fail if called upon to provide power to equipment important to safety.
Inspection Report# : 2006006 *(pdf)*

**Significance:** G Mar 02, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Provide Testing for Boron Precipitation Control Flow Instrumentation**
The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” that was of very low safety significance. Specifically, after performing modifications to implement a new Boron Precipitation Control method for post-LOCA (Loss of Coolant Accident) operations, the licensee failed to both identify and establish testing for the flow instrumentation that the operators would use post-LOCA to ensure minimum flow for proper boron dilution. This issue was entered into the licensee’s corrective action system.

The issue was more than minor because if it was left uncorrected, the finding would become a more significant safety concern. Had this issue not been detected, the instrument could have fallen well out of tolerance in the future leading to inaccurate readings. During post-LOCA operations, these inaccurate readings could have caused operators to establish insufficient Boron Precipitation Control (BPC) flows. The finding screened as having very low significance (Green) using IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for the At-Power Situations,” because the inspectors answered “no” to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. While the lack of calibration procedures for the flow instrumentation would have eventually caused the instrument to read inaccurately, the inspection team identified the problem early enough so that the instrument drift (as provided in the vendor instructions) would not be large enough to cause inaccuracies that would adversely affect BPC flows.
Inspection Report# : 2006006 *(pdf)*

**Significance:** G Jan 07, 2004
Identified By: NRC
Item Type: VIO Violation
**Failure to Take Corrective Actions for a Previous NCV Concerning SW Discharge Path Swapover Setpoints**
The team identified a Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." Specifically, the licensee failed to provide a basis for the setpoint to swap the service water system discharge path. This issue was previously identified as a Non-Cited Violation in Inspection Report 05000346/2002014 and the corrective actions taken by the licensee failed to correct the originally identified condition. The primary cause of this violation was related to the cross-cutting areas of problem identification and resolution and human performance, because the licensee did not recognize that the corrective actions taken needed to restore compliance with the identified violation of NRC requirements.

The issue was determined to be more than minor because the licensee had not corrected a previous violation and was relying on non-safety-related equipment to perform a safety function under design bases conditions. Because the issue was previously determined to be of very low safety significance, NRC management concluded that the violation could be categorized as having very low safety significance. (Section 4OA3(3)b.11)
Inspection Report# : 2003010 *(pdf)*

**Significance:** G Jan 07, 2004
Identified By: NRC
Item Type: VIO Violation
**Failure to Take Corrective Actions for a Previous NCV Concerning SW Pump Discharge Check Valve Acceptance Criteria**
The team identified a Cited Violation of Technical Specifications Section 4.05a and 10 CFR 50.55a. Specifically, the

licensee failed to ensure that the service water discharge check valve was tested in accordance with the American Society of Mechanical Engineers Code. The primary cause of this violation was related to the cross-cutting areas of problem identification and resolution and human performance, because the licensee did not recognize that the corrective actions taken needed to ensure compliance with NRC requirements.

The issue was determined to be more than minor because the inadequate test acceptance criteria allowed the licensee to accept a check valve as performing its intended function at less than full system flow. The issue was of very low safety significance using the Phase 1 of the significance determination process based on the licensee's determination that the system was operable but degraded. (Section 4OA3(3)b.12)

Inspection Report# : 2003010 *(pdf)*

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# Barrier Integrity

**Significance:** G Mar 31, 2006
Identified By: NRC
Item Type: FIN Finding
**UNQUALIFIED EXAMINATION OF A PRESSURIZER SURGE LINE DM WELD**
The inspectors identified a finding having very low safety significance for failure to use qualified transducers during ultrasonic examination of a dissimilar metal weld on the pressurizer surge line. As a corrective action, the licensee obtained qualified transducers, repeated the examination, and entered this issue into the corrective action program. This finding is of more than minor significance because it is associated with the Barrier Integrity cornerstone attribute of “Reactor Coolant System Equipment and Barrier Performance,” and affected the cornerstone objective to provide reasonable assurance that physical design barriers (reactor coolant system) protect the public from radionuclide releases caused by accidents or events. Absent NRC intervention, the licensee would have relied on this degraded examination, which would have placed this weld at increased risk for undetected cracking, leakage, or component failure. This finding is of very low safety significance because a qualified examination was subsequently performed with no relevant indications detected. The inspectors determined that the finding was not suitable for SDP evaluation because the failure to use qualified transducers did not directly result in degraded or inoperable equipment. Therefore, this finding was reviewed by Regional Management, in accordance with IMC 0612 Section 05.04c, and determined to be of very low safety significance.
Inspection Report# : 2006002 *(pdf)*

**Significance: N/A** Aug 09, 2002
Identified By: NRC
Item Type: VIO Violation
**I.A - Operating with Reactor Coolant Pressure Boundary Leakage**
Technical Specification 3.4.6.2.a, Amendment 220, dated April 14, 1998, requires, in part, that the licensee shall limit reactor coolant system leakage to "No PRESSURE BOUNDARY LEAKAGE" during Modes 1 through 4.

Contrary to the above, between May 18, 2000, and February 16, 2002, the licensee started up and operated the plant in Modes 1 through 4 with reactor coolant system pressure boundary leakage, i.e. control rod drive penetration leakage. Specifically, the licensee returned the plant to operation following the 2000 refueling outages without fully characterizing and eliminating reactor coolant system pressure boundary leakage on the reactor pressure vessel head as evidenced by significant boric acid deposits on the reactor pressure vessel head at the start and end of the outage and by the development of new and extensive boric acid deposits on reactor containment equipment during the operation cycle.

This is a violation associated with a RED SDP finding. Civil Penalty - $5,000,000 (EA-05-071)
Inspection Report# : 2005012 *(pdf)*
Inspection Report# : 2005013 *(pdf)*

**Significance: SL-I** Aug 09, 2002
Identified By: NRC
Item Type: VIO Violation

**I.B.1 & 2 - Information Included in (1) CR 2000-1037 and (2) WO 00-001846-000 was not complete and accurate in all material respects.**
10 CFR 50.9 requires that information provided to the Commission by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the licensee shall be complete and accurate in all material respects.

10 CFR 50, Appendix B, Criterion XVI requires, in part, that for significant conditions adverse to quality, the cause of the condition and the corrective actions taken to preclude repetition shall be documented.

10 CFR 50, Appendix B, Criterion XVII, requires, in part, that the licensee shall maintain sufficient records to furnish evidence of activities affecting quality and that those records shall include monitoring of work performance.

Condition Report (CR) 2000-1037, closed May 1, 2000, documented corrective actions for the presence of boric acid on the reactor pressure vessel head, a significant condition adverse to quality, that included: "Accumulated boron deposited between the reactor head and the thermal insulation was removed during the cleaning process performed under W.O. [Work Order] 00-001846-000. No boric acid induced damage to the head surface was noted during the subsequent inspection."

Work Order 00-001846-000, "Clean Boron Accumulation from Top of Reactor Head and Top of Insulation," dated April 25, 2000, required the licensee staff to "clean boron accumulation from top of reactor head and on top of insulation." The Work Order Log, included as Page Four of the completed Work Order, documented that the, "work [was] performed without deviation" and was signed by the System Engineer on April 25, 2000.

Contrary to the above,

1 - The information included in CR 2000-1037 relative to the completed corrective actions and the subsequent inspection results were not complete and accurate in all material respects. Specifically, the licensee did not remove the accumulated boron deposits from all areas between the reactor head and the thermal insulation and did not conduct subsequent inspections of the entire reactor head. Instead, the licensee removed accumulated boric acid deposits from a portion of the reactor vessel head and conducted subsequent inspections for those portions of the reactor vessel head where the boric acid deposits had been removed.

2 - The Work Order Log, included as Page Four of completed Work Order 00-001846-000, a record required by Commission regulations to furnish evidence of activities affecting quality, contained information that was not accurate in all material respects. Specifically, the Work Order Log indicated that boron accumulation was cleaned from the top of the reactor head and on top of the insulation, without deviation, when, in fact, boric acid deposits were left on the head after the cleaning was completed on April 25, 2000.

This is a Severity Level I violation (Supplement VII). Civil Penalty $110,000 (EA-05-068)
Inspection Report# : 2005013 *(pdf)*
Inspection Report# : 2005012 *(pdf)*

**Significance: SL-II** Aug 09, 2002
Identified By: NRC
Item Type: VIO Violation
**I.C. 1 & 2 & 3 - Failure to determine the cause of a signficant condition adverse to quality involving three examples of identified boric acid leakage.**
10 CFR 50, Appendix B, Criterion XVI, requires, in part, that licensees shall establish measures to ensure that conditions adverse to quality such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. For significant conditions adverse to quality, the licensee shall establish measures to ensure that the cause of the condition is determined and that corrective actions are taken to preclude repetition.

Plant Procedure NG-NA-00702, "Corrective Action Program," Revision 3, defined a significant condition adverse to quality to be a condition, which, if left uncorrected, could have an undesirable effect on plant safety, personal safety, regulatory position, financial liability, or environmental impact.

Contrary to the above, the licensee did not determine the cause of the condition and did not implement corrective actions to preclude repetition of the condition associated with the identification and removal of boric acid on the reactor vessel head, a significant condition adverse to quality, prior to closing the associated condition reports.

Specifically:

1 - On April 27, 2000, the licensee closed CR 2000-0781, "Leakage from CRD [Control Rod Drive] Structure Blocked Visual Exam of Reactor Vessel Head Studs," issued on April 6, 2000, associated with the accumulation of boric acid deposits on the reactor vessel head studs without determining the cause of the deposits, i.e., identifying the source of the reactor coolant system leakage, and without taking corrective actions to preclude recurrence.

2 - On April 27, 2000, the licensee closed CR 2000-0782, "Inspection of Reactor Flange Indicated Boric Acid Leakage From Weep Holes," issued on April 6, 2000, associated with the accumulation of boric acid deposits on the reactor vessel head, without determining the cause of the boric acid deposits, i.e., identifying the source of the reactor coolant system leakage, without removing all of the known boric acid deposits on the reactor pressure vessel head, and without taking corrective actions to prevent recurrence.

3 - On May 1, 2000, the licensee closed CR 2000-1037, "Inspection of Reactor Head Indicated Accumulation of Boron in Area of the CRD [Control Rod Drive] Nozzle Penetration," issued on April 17, 2000, associated with the accumulation of boric acid deposits on the reactor vessel head, without determining the cause of the boric acid deposits, i.e., identifying the source of the reactor coolant system leakage, without removing all of the known boric acid deposits on the reactor vessel head, and without taking corrective actions to prevent recurrence.

This is a Severity Level II violation (Supplement I) Civil Penalty - $110,000 (EA-05-066)
Inspection Report# : 2005012 *(pdf)*
Inspection Report# : 2005013 *(pdf)*

**Significance: SL-II** Aug 09, 2002
Identified By: NRC
Item Type: VIO Violation
**I.D - Failure to comply with Boric Acid Control procedure in that obstacles were not removed to allow for a complete inspection of the RPV head (failure to implement mod)**
10 CFR 50, Appendix B, Criterion V, requires, in part, that activities affecting quality be accomplished in accordance with written procedures.

Davis-Besse Station Procedure NG-EN-00324, "Boric Acid Corrosion Control Program," Revisions 1/C1 and 2, Step 6.3.1, required, in part, that an initial inspection of boric acid buildup shall be performed to determine the "as found" conditions and to document the inspection results. The procedure also required, in Attachment 3, that insulation and other hindrances to direct visual [inspection] be removed as needed to allow detailed inspections of components suspected of leakage.

Potential Condition Adverse to Quality (PCAQ) 96-0551, initiated on April 21, 1996, documented the licensee's inability to comply with some inspections of the reactor pressure vessel head, as required by Procedure NG-EN-00324, and an inability to accurately determine the reactor pressure vessel head "as found" conditions, associated with boric acid deposits on the reactor pressure vessel head, due to the restrictions resulting from the location and size of the inspection ports, "mouse holes." The PCAQ further documented that only 50 to 60 percent of the reactor pressure vessel head could be inspected using the current inspection ports.

Modification 94-0025, initiated on May 27, 1994, and referenced as corrective action for PCAQ 96-0551, directed the completion of modifications to the reactor pressure vessel head service structure inspection ports to permit the inspection and cleaning of 100 percent of the reactor vessel head in accordance with Procedure NG-EN-00324.

Contrary to the above, on May 18, 2000, and at the end of Refueling Outage 12, the licensee failed to remove obstructions, including boric acid deposit buildups, necessary to conduct a detailed inspection of the reactor pressure vessel head and other components that may be suspected of leakage, as required by Plant Procedure NG-EN-00324, "Boric Acid Corrosion Control Program." The licensee's ability to conduct the inspections was significantly limited as a result of its concurrent deferral of the installation of Modification 94-0025, a corrective action for a significant condition adverse to quality documented in PCAQ 96-0551 and associated with the licensee's failure during previous outages to conduct complete inspections and cleaning of boric acid deposits on the reactor pressure vessel head.

This is a Severity Level II violation (Supplement I) Civil Penalty $110,000 (EA-05-067)
Inspection Report# : 2005012 *(pdf)*
Inspection Report# : 2005013 *(pdf)*

**Significance: SL-I** Aug 09, 2002
Identified By: NRC
Item Type: VIO Violation
**I.E1 & 2 - Licensee Responses to Bulletin 2001-01 dated (1) September 4, 2001, and (2) October 17, 2001, were materially incomplete and inaccurate.**
10 CFR 50.9 requires that information provided to the Commission by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the licensee shall be complete and accurate in all material respects.

NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," required all holders of operating licenses for pressurized water nuclear power reactors to provide information related to the structural integrity of the reactor vessel head penetration (VHP) nozzles for their respective facilities, including the extent of VHP nozzle leakage and cracking that has been found to date, the inspections and repairs that have been undertaken to satisfy applicable regulatory requirements, and the basis for concluding that their plans for future inspections will ensure compliance with applicable regulatory requirements.

Contrary to the above, the licensee, a holder of an operating license for a pressurized water nuclear power reactor, the Davis-Besse Station, provided the Commission responses to Bulletin 2001-01 which included materially inaccurate and incomplete information as follows:

1 - In a September 4, 2001, response to the Bulletin entitled, "Response to Bulletin 2001-01," Serial 2731, the licensee made the following four materially inaccurate and incomplete statements:

(a) The licensee's response to Bulletin Item 1.c, on page 2 of 19, stated: "the minimum gap being at the dome center of the RPV [reactor pressure vessel] head where it is approximately 2 inches, and does not impede a qualified visual inspection."

The licensee's response was materially inaccurate, in that, the statement contradicted statements in the licensee's documents identified as PCAQR 94-0295 and 96-0551, which clearly stated that inspection capability at the top of the reactor vessel head was limited. The limitation was stated to be caused by the restricted access to the area through the service structure "weep holes", the curvature of the reactor pressure vessel head, and by the limited space to manipulate a camera due to the insulation that creates the two inch gap.

(b) The licensee's response to Bulletin Item 1.d, which requested inclusion of a description of any limitations (insulation or other impediments) to accessibility of the bare metal of the reactor pressure vessel head for visual examinations, did not include a description of any limitations.

The licensee's response was materially incomplete in that the response did not mention that accessibility to the bare metal of the reactor pressure vessel head was impeded, during the Eleventh (1998) and the Twelfth (2000) Refueling Outages, by the presence of significant accumulations of boric acid deposits.

(c) The licensee's response to Bulletin Item 1.d, which also requested a discussion of the findings of reactor pressure vessel head inspections, stated that for the Twelfth Refueling Outage (2000), the inspection of the reactor pressure vessel head/nozzles indicated some accumulation of boric acid deposits.

The licensee's response was materially incomplete and inaccurate in that it mischaracterized the accumulation of boric acid on the reactor pressure vessel head and did not mention the evidence of corrosion that was evidenced by the pictures and the video examination of reactor pressure vessel head conditions documented at the beginning and ending of the Twelfth Refueling Outage (2000).

(d) The licensee's response to the Bulletin, on Page 3, stated: "The boric acid deposits were located beneath the leaking flanges with clear evidence of downward flow. No visible evidence of nozzle leakage was detected."

The licensee's response was materially inaccurate in that the boric acid deposits were not all located under leaking flanges and the licensee lacked clear evidence of the absence of downward flow for all nozzles. Specifically, the presence of boric acid deposits was not limited only to the areas beneath the flanges, as implied by that statement. The build-up of boric acid deposits was so significant that the licensee could not inspect all of the nozzles. As a result, the licensee also did not have a basis for stating that no visible evidence of nozzle leakage was detected.

2. In an October 17, 2001, response to the Bulletin entitled, "Supplemental Response to Bulletin 2001-01," Serial 2735, the licensee stated: "In May 1996, during a refueling outage, the RPV [reactor pressure vessel] head was inspected. No leakage was identified, and these results have been recently verified by a re-review of the video tapes obtained from that inspection. The RPV head was mechanically cleaned at the end of the outage. Subsequent inspections of the RPV head in the next two refueling outages (1998 and 2000), also did not identify any leakage in the CRDM [control rod drive mechanism] nozzle-to-head areas that could be inspected. Video tapes taken during these inspections have also been re-reviewed."

The licensee's response was materially inaccurate, in that: (1) each reactor pressure vessel head control rod drive penetration was not inspected in May 1996, as documented in PCAQR 96-0551, and; (2) the reactor pressure vessel head, including the area around each control rod drive penetration, was not completely cleaned, as noted in PCAQR 98-0649, which was prepared at the start of the Eleventh Refueling Outage (1998), which stated that there were old boric acid deposits on the head.

This is a Severity Level I violation (Supplement VII) Civil Penalty $120,000 (EA-05-072)
Inspection Report# : 2005012 *(pdf)*
Inspection Report# : 2005013 *(pdf)*

**Significance: N/A** Aug 09, 2002
Identified By: NRC
Item Type: VIO Violation
**II.A 1 & 2 & 3 - Three examples of inadequate corrective actions involving (1) fouling of containmnet air coolers, (2) fourling of containment rad monitors, and (3) increased trend in unident. leakag**
10 CFR Part 50, Appendix B, Criterion XVI, requires, in part, that the licensee shall establish measures to ensure that conditions adverse to quality such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. Criterion XVI also requires that for significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and that corrective actions are taken to preclude repetition.

Plant Procedure NG-NA-00702, "Corrective Action Program," Revision 3, defined a significant condition adverse to quality to be a condition adverse to quality, which, if left uncorrected, could have an undesirable effect on plant safety, personal safety, regulatory position, financial liability, or environmental impact.

Contrary to the above, the licensee failed to determine the root cause of and take corrective actions to preclude the repetition of:

1. Fouling of containment air cooling fins by boric acid, between June 2000 and February 16, 2002, a significant condition adverse to quality.

2. - Fouling of the containment radiation elements by boric acid and iron oxide, between April 2001, and February 16, 2002, a significant condition adverse to quality, and

3. - An increasing trend in unidentified reactor coolant system leakage, between March 2001, and December 2001, a significant condition adverse to quality.

This is a violation associated with a RED SDP finding (EA-03-025).
Inspection Report# : 2005012 *(pdf)*
Inspection Report# : 2005013 *(pdf)*

**Significance: N/A** Aug 09, 2002
Identified By: NRC

Item Type: VIO Violation

**II.B - Inadequate Boric Acid Corrosion Control procedure.**

10 CFR Part 50, Appendix B, Criterion V, requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.

Procedure NG-EN-00324, "Boric Acid Corrosion Control Program," Revisions 0 through 2 (effective date October 1, 1999), were classified as a procedure affecting quality under the licensee's administrative system.

Contrary to the above, between October 1, 1999, and March 6, 2002, Procedure NG-EN-00324, "Boric Acid Corrosion Control Program," Revisions 0 through 2, were not appropriate to the circumstances and contributed to the licensee's failure to detect and address boric acid corrosion of the reactor vessel head, as follows:

1 - The procedure inappropriately focused on bolted and flanged connections in the definition of leakage (Sections 4.2 though 4.4), the definition of reactor coolant system pressure boundary components (Section 4.9), and the identification of investigation locations (Section 6.1) at the expense of identifying the potential for through-wall leakage.

2 - The procedure did not include adequate guidance, specifications, or threshold levels for initiating a "detailed inspection" in order to ensure consistent implementation of Section 6.3.4 of the procedure.

3 - The procedure did not require the identification of and corrective actions to preclude the repetition of boric acid leaks, a significant condition adverse to quality, but instead only required the preparation of a repair tag or work order to facilitate repair of the leak.

4 - The procedure did not define the qualifications and training necessary to permit engineering staff to conduct inspections and evaluations in a consistent manner, including the use of proper inspection techniques, observations, recording of results, and evaluations.

5 - The procedure inappropriately exempted stainless steel or Inconel components from further examination related to boric acid corrosion, unless the examination was during an ASME Section XI test which might require a bolting examination.

6 - The procedure inappropriately did not require the licensee staff to maintain records necessary to demonstrate the proper completion of activities affecting quality.

This is a violation associated with a RED SDP finding (EA-03-025).
Inspection Report# : 2005012 *(pdf)*
Inspection Report# : 2005013 *(pdf)*

**Significance: SL-IV** Sep 04, 2001
Identified By: NRC
Item Type: VIO Violation

**II.D 1 & 2- Two examples of incomplete and ainaccurate information contained in quality documents, (1) a void request dated 9/12/93, and (2) a QA Audit Report No. AR-00-OUTAG-01.**

10 CFR 50.9 requires that information provided to the Commission by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the licensee shall be complete and accurate in all material respects.

10 CFR Part 50, Appendix B, Criterion XVII, requires, in part, that the licensee shall maintain sufficient records to furnish evidence of activities affecting quality and that those records shall include audits and those actions taken to correct any deficient conditions.

Contrary to the above, the following information was not complete or accurate in all material respects for documents required to be maintained or provided to the Commission:

1 - On September 23, 1993, the licensee processed a "Document Void Request" to cancel Modification 90-012 which stated, "Current inspection techniques using high-powered cameras preclude the need for inspection ports, additionally, cleaning of the reactor vessel head during last three outages was completed successfully without requiring access ports."

However, the quoted statement was not accurate in all material respects, in that, the licensee left boric acid deposits on the reactor vessel head at the end of both the seventh and eighth refueling outages, the two outages preceding this statement.

2 - Quality Assurance Audit Report AR-00-OUTAG-01, dated July 7, 2000, stated, in part, "Boric Acid Corrosion Control Checklists and Condition Reports were initiated by inspectors when prudent to document and evaluate boric acid accumulation and leaks. Boric acid leakage was adequately classified and corrected when appropriate. Engineering displayed noteworthy persistence in ensuring boric acid accumulation from the reactor head was thoroughly cleaned." However, the audit report was not accurate in all material respects in that the licensee did not: 1) thoroughly clean the reactor head during the outage; 2) did not prepare a boric acid corrosion control checklist for the boric acid left on the head after the cleaning attempt; and 3) identify, properly classify, or correct the boric acid accumulation and leaks.

This is a Severity Level IV violation (Supplement VII) (EA-05-070)
Inspection Report# : 2005013 *(pdf)*
Inspection Report# : 2005012 *(pdf)*

**Significance: SL-III** Apr 25, 2000
Identified By: NRC
Item Type: VIO Violation
**II.C - Two PCAQs (98-0649 & 98-0767) were closed as completed based on inaccurate informationon.**
10 CFR 50.9 requires that information provided to the Commission by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the licensee shall be complete and accurate in all material respects.

10 CFR Part 50, Appendix B, Criterion XVII, requires, in part, that the licensee shall maintain sufficient records to furnish evidence of activities affecting quality and that those records shall include actions taken to correct any deficient conditions.

Contrary to the above, the following information was not complete or accurate in all material respects for documents required to be maintained or provided to the Commission:

1 - Potential Condition Adverse to Quality Report (PCAQR) 98-0649, dated April 18, 1998, contained the following closure statement: "Accumulation of boric acid on the reactor vessel caused by leaking CRDMs [control rod drive mechanisms] has not resulted in any boric acid corrosion. This was identified through inspections following reactor vessel head cleaning in past outages....Additionally, B&W [Babcock & Wilcox] documentation discussing CRDM nozzle cracking further stated that boric acid deposits on the head caused by leaking CRDM flanges would not result in head corrosion." However, the quoted statements were not accurate in all material respects in that the licensee had previously not cleaned all areas of the reactor head of boric acid deposits, had not inspected the base metal under all the deposits to determine whether corrosion was present, and no B&W documentation was available to support the claim that boric acid would not result in head corrosion.

2 - Potential Condition Adverse to Quality Report (PCAQR) 98-0767, dated April 25, 1998, Section 4A, Item F, included the following closure justification, "The boric acid deposits were removed from the head." However, the quoted statement was not accurate in all material respects in that the licensee had not removed all of the boric acid deposits from the head as of the end of the eleventh refueling outage.

This is a Severity Level III violation (Supplement VII) (EA-05-069)
Inspection Report# : 2005013 *(pdf)*
Inspection Report# : 2005012 *(pdf)*

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# Emergency Preparedness

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# Occupational Radiation Safety

**Significance:** G Dec 31, 2006
Identified By: NRC
Item Type: FIN Finding
**FAILURE TO ADEQUATELY IMPLEMENT ALARA RADIOLOGICAL DOSE CONTROLS**
A finding of very low safety significance was identified by the inspectors when licensee personnel failed to adequately implement radiological dose controls as a result of ineffective radiological/ALARA planning and controls during Refueling Outage 14 (RFO14). The collective occupational radiation dose received by individuals for some work activities significantly exceeded the planned or intended dose that the licensee determined was ALARA for those work activities. The finding was more than minor because the finding was associated with the Occupational Radiation Safety Cornerstone attribute of ALARA planning/dose projection, and affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation. The finding was determined to be of very low safety significance because, although the finding involved ALARA planning and controls, the 3-year rolling average exposure for Davis-Besse was less than the SDP Green-to-White threshold of 135 person-rem for pressurized water reactors, and the finding did not involve an overexposure, a substantial potential for an overexposure, or an impaired ability to assess dose. As part of the licensee’s corrective actions to address this issue, additional rigor in outage planning was planned. The inspectors also determined that the cause of the finding was related to the cross-cutting area of human performance because licensee personnel failed to effectively plan work activities to adequately implement radiological dose controls.
Inspection Report# : 2006005 *(pdf)*

**Significance:** G Jun 30, 2006
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**RADIATION WORKERS ENTERED POSTED HIGH RADIATION AREAS WITHOUT PROPER AUTHORIZATION**
A finding of very low safety significance and associated Non-Cited Violation of TS 6.12.1.b was self-revealed when, in three separate instances, radiation workers entered posted high radiation areas (HRAs) without appropriate authorization. Specifically, the radiation workers entered posted HRAs although in each instance the radiation work permit used to access the radiologically restricted area did not permit access to HRAs. The finding was more than minor because it was associated with the human performance attribute of the occupational radiation safety cornerstone and affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation. The finding was of very low safety significance because it did not involve: 1) As-Low-As-Reasonably-Achievable planning or controls; 2) an overexposure; 3) a substantial potential for an overexposure; or 4) an impaired ability to assess dose. The primary cause of this finding was related to the cross-cutting area of Human Performance.
Inspection Report# : 2006003 *(pdf)*

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## Public Radiation Safety

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## Physical Protection

Physical Protection information not publicly available.

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## Miscellaneous

**Significance:** G Nov 17, 2006
Identified By: NRC
Item Type: NCV NonCited Violation

**FAILURE TO INITIATE A CONDITION REPORT FOR CONDITIONS ADVERSE TO QUALITY**
The inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” when licensee personnel failed to generate condition reports or notifications to identify deficiencies associated with safety-related equipment. In particular, the inspectors identified eight instances between April 2006 and November 2006 in which licensee personnel failed to document degraded declutch operators associated with safety-related MOVs although personnel were aware of the condition. As part of the licensee’s immediate corrective actions, notifications and/or condition reports were generated to ensure that the identified deficiencies were entered into the corrective action program. The inspectors determined that the finding was more than minor because the issue was 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. The inspectors determined that the issue was of very low safety significance because the finding did not represent an actual loss of a safety function of a system. The cause of the finding was related to the corrective action program aspect of the cross-cutting area of Problem Identification and Resolution because the implementation of the licensee’s corrective action program did not identify declutch operator degradation completely, accurately, and in a timely manner commensurate with the safety significance of the issue.
Inspection Report# : 2006007 *(pdf)*

**Significance: N/A** Nov 17, 2006
Identified By: NRC
Item Type: FIN Finding
**PI&R Summary**
The inspectors concluded that, overall, problems were properly identified, evaluated, and corrected. Generally, licensee personnel properly prioritized and evaluated issues. However, the inspectors identified numerous examples in which degraded manual declutch operators associated with safety-related motor-operated valves (MOVs) were not identified in the corrective action program for resolution. Root cause evaluations for significant problems were appropriately detailed. Corrective actions to address problems were generally adequate. Audits and self-assessments were effective in identifying deficiencies and recommendations were appropriately captured. The use of operating experience was adequate. The inspectors did not identify any weaknesses in the Employee Concerns Program (ECP) that contributed to station performance deficiencies or adversely impacted the establishment of a Safety Conscious Work Environment (SCWE).
Inspection Report# : 2006007 *(pdf)*

**Significance:** G Jun 30, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**FIRE PROTECTION, TRANSIENT COMBUSTIBLES ON THE DRY FUEL STORAGE PAD**
A finding of very low safety significance and an associated Non-Cited Violation of 10 CFR 72.212, “Conditions of general license issued under §72.210,” was identified by the inspectors when the licensee failed to adequately control transient combustible material near the dry spent fuel horizontal storage modules (HSMs) in accordance with procedures. The proper control of transient combustible material was required by fire protection procedures that prescribed compliance with conditions specified in the NRC-issued Certificate of Compliance for the HSMs. As part of their immediate corrective actions, licensee personnel removed the transient combustible material from the area surrounding the HSMs and generated a condition report to enter this issue in their corrective action program. This finding was more than minor because it was associated with protection against potential fire damage to the HSMs, which if left uncorrected could lead to a more significant safety concern. The inspectors determined that the finding was not suitable for SDP evaluation because the issue did not involve permanently installed plant equipment. Therefore, the finding was reviewed by Regional Management in accordance with Inspection Manual Chapter 0612, Section 05.04c, and determined to be of very low safety significance. The plant fire brigade could have been dispatched to extinguish a fire before significant damage to the HSMs occurred. The primary cause of the finding was related to the cross-cutting area of Human Performance.
Inspection Report# : 2006003 *(pdf)*

Last modified : March 01, 2007