Davis-Besse
1Q/2008 Plant Inspection Findings

Initiating Events

Significance: G Mar 31, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO FOLLOW WELDING PROCEDURES
The inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the failure of two contractor welders to adhere to welding procedures for the weld overlay (WOL) repairs on two pressurizer safety relief valve (SRV) nozzles. Specifically, both welders failed to use calculated relative travel speed settings as required by procedure in order to ensure that correct heat input values (an essential variable) were maintained. The licensee entered the issue into their corrective action program and suspended welding activities on the two SRVs until it was determined that the travel speeds used resulted in a heat input that was bounded by the procedure as qualified. This finding is greater than minor because if left uncorrected it would have become a more significant safety concern in that the failure to control heat input could have reduced the impact toughness of the WOL such that it would be susceptible to brittle fracture. The finding is of very low safety significance because calculations determined that the resulting heat inputs were bound by the welding procedure specifications’ (WPS) parameters. As a result, assuming worst case degradation, it is unlikely that there would be reactor coolant system leakage or the loss of safety function of any mitigating system. The cause of the finding is related to the cross-cutting aspect of Human Performance, Work Practices, (Item H.4.(c)) because licensee personnel failed to ensure supervisory and management oversight activities of their contractors such that nuclear safety was ensured.

Inspection Report# : 2008002 *(pdf)*

Significance: G Mar 31, 2008
Identified By: NRC
Item Type: FIN Finding
UNEXPECTED REACTIVITY EXCURSION DUE TO UNIDENTIFIED VALVE POSITION DURING POST REPAIR AIR PRESSURE TESTING
A self-revealing finding was identified for the failure of operators to maintain configuration control of valves during an air pressure test of a repair of a feedwater heater. Specifically, the operators left valve RD198 open during a pressure test of the extraction steam, or shell side, of feedwater heater 1-5 of the Main Feedwater System. This loss of configuration control gave testing air a path to the main condensers and led to degradation of the condenser vacuum, which then caused the Integrated Control System to raise reactor power unexpectedly. No violation occurred. Once the issue was identified, the licensee stopped the air pressure test and entered the finding into their corrective action program. The finding is greater than minor since it was associated with the configuration control-operating equipment lineup attribute of the Initiating Events Cornerstone and because it affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability during power operations. The finding is of very low safety significance since it did not contribute to the likelihood of a primary or secondary system loss of coolant accident, did not contribute to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions would not be available, and did not increase the likelihood of a fire or internal/external flood. The finding was associated with the cross-cutting area of human performance in that work control and specifically the coordination of work activities did not properly record or assess the status of a valve in the test boundary and created a condition that had an operational impact (H.3(b)).
Inspection Report# : 2008002 *(pdf)*

Significance: G Dec 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation

INSPECTION PROCEDURE FOR POLAR CRANE OMITTED VISUAL INSPECTION OF STRUCTURAL COMPONENTS
The inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures and Drawings.” Specifically, the licensee failed to provide a procedure to perform visual inspection of the polar crane structural members required by American National Standards Institute (ANSI) B30.2-1976. The issue was entered into the licensee’s corrective action program, and a licensee procedure was revised to perform visual inspection of the polar crane structural members required by ANSI B30.2-1976. This finding was more than minor because the finding was associated with the Initiating Events Cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown. Specifically, the purpose of the polar crane structural inspections is to limit the likelihood of a polar crane structural component failure to ensure safe load handling of heavy loads over the reactor core, over spent fuel or over safety-related systems. The finding was of very low Safety significance based a Phase 1 screening in accordance with Inspection Manual Chapter (IMC) Appendix G, “Shutdown Operations Significance Determination Process (SDP),” Table 1 qualitative assessment, because no structural concerns were identified when the polar crane was inspected in the previous two refueling outages (12RFO and 13RFO) and the low number of lifts performed by the polar crane during a single refueling outage. The finding has a cross-cutting aspect in the area of human performance because the licensee did not provide a complete, accurate, and up-to-date procedure to plant personnel (H.2(c)).

Inspection Report# : 2007005 *(pdf)*

Significance: G Dec 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
INTERNALS HANDLING ADAPTER DESIGN CALCULATION DID NOT CONSIDER MATERIAL FRACTURE TOUGHNESS REQUIREMENTS
The inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control.” Specifically, the licensee failed to assure and verify that the design of Internals Handling Adapter lifting pins was based on material fracture toughness as required by ANSI N14.6-1978. The issue was entered into the licensee’s corrective action program, and the licensee has initiated an engineering change to replace the Internals Handling Adapter lifting pins prior to removing the reactor vessel head in the next refueling outage 15RFO. This finding was more than minor because the finding was associated with the Initiating Events Cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown. Specifically, the purpose of the Internals Handling Adapter meeting the design requirements of ANSI N14.6-1978 is to limit the likelihood of a structural component failure to ensure safe load handling of heavy loads over the reactor core, over spent fuel or over safety-related systems. The finding was of very low Safety significance based a Phase 1 screening in accordance with IMC Appendix G, “Shutdown Operations SDP,” Table 1 qualitative assessment, because although the fracture toughness of the lifting pin material was not evaluated, the lifting pins did satisfy ANSI N14.6-1978 stress design factors and the lifting pins were subjected to a low number of historical reactor vessel head lifts that utilized the Internal Handling Adapter. The finding has a cross-cutting aspect in the area of problem identification and resolution because the licensee did not take appropriate corrective actions to promptly correct the design bases non-conformance identified in their design calculation (P.1(d)).

Inspection Report# : 2007005 *(pdf)*

Significance: G Sep 30, 2007
Identified By: NRC
Item Type: FIN Finding
IMPROPER DESIGN OF A WELD PATCH FOR A CRACK IN CIRCULATING WATER PIPE
The inspectors identified a finding for the licensee’s failure to properly design a temporary repair for a through wall pipe crack found in the circulating water system. Specifically, the inspectors identified that a stress intensification factor, used in determining the minimum required pipe wall thickness, repair plate thickness, and repair fillet weld size, was improperly calculated. Once identified, the licensee entered the issue into their corrective action program and appropriately modified the design and supporting calculations. No violation of regulatory requirements occurred. The inspectors determined that the finding was more than minor because, if the original design was left uncorrected, a more significant safety concern could have been created. Additionally, the finding was more than minor, as shown in examples of minor issues, IMC 0612, Appendix E, example 3a, because the calculation errors were significant enough

that the modification required revision. The finding 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; did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available; did not increase the likelihood of a fire; and did not involve degradation of a barrier specifically designed to mitigate flooding or involve the total loss of any safety function. The inspectors also determined that the cause of the finding was related to the cross-cutting area of human performance with the component of work practices (H4.(a)) in that self and peer checking did not identify calculation issues with the original design.
Inspection Report# : 2007004 *(pdf)*

Mitigating Systems

Significance: Mar 31, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
AIR VOID IN DECAY HEAT/LOW PRESSURE INJECTION SYSTEM DUE TO INADEQUATE VENTING AFTER MAINTENANCE
A self-revealing NCV of TS 4.5.2b was identified for failure to properly fill and vent a portion of decay heat/low pressure injection train 1 after maintenance which resulted in an approximate 15 cubic foot air void in the discharge piping of the train 1 decay heat/low pressure injection system for approximately 59 days of plant operation. Work packages and procedures used in restoration and refilling of the system did not adequately identify and provide for filling of drained high points in the piping. Upon identification with a 6 inch step decrease in pressurizer level while aligning the decay heat system for refueling operations, the licensee filled and then vented the system from high point vents located in system's discharge piping in the plant's containment. The licensee entered the failure to properly fill and vent the system after maintenance in their corrective action program. The finding is greater than minor because it was associated with the equipment performance attribute of the Mitigating Systems cornerstone, and affected the cornerstone objective in that permitting an air void in the train's discharge piping affected the reliability and capability of the system. The finding is of very low safety significance because it did not result in a loss of function per Part 9900, "Technical Guidance – Operability Determinations and Functionality Assessments," did not represent an actual loss of safety function, and is not potentially risk significant due to external events. The finding is associated with the cross-cutting area of human performance in that the resources and specifically work packages and procedures were not adequate to ensure that the train 1 decay heat/low pressure injection system was restored to a filled and vented system condition (H.2(c)) after completion of maintenance activities.
Inspection Report# : 2008002 *(pdf)*

Significance: Dec 31, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
REDUCED FLOW THROUGH COMPONENT COOLING WATER 1 HEAT EXCHANGER BECAUSE OF IMPROPER VALVE OPENING LIMIT STOP
A self-revealing NCV of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," was identified for failing to include appropriate quantitative or qualitative acceptance criteria for assuring the proper setting of the travel stops on valve SW-36 [Component Cooling Water Heat Exchanger 1 Service Water Outlet Valve] after valve operator maintenance. This resulted in a valve opening setting that, in the event of a safety feature system actuation, would limit service water flow to less than flows analyzed in the approved flow balance calculation for flow to the component cooling water heat exchanger 1. The licensee entered the deficiency into their corrective action program and adjusted the travel stops to provide for the proper service water flow. This finding is greater than minor because the finding was associated with the configuration control attribute of the Mitigating Systems Cornerstone and did affect the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Also, the finding was more than minor, using example 1a of IMC 0612 Appendix E, September 20, 2007, in that testing determined CCW heat exchanger flows to be degraded subsequent to stop setting adjustment and declaring the heat exchanger operable. The finding was evaluated using the SDP and was determined to be a finding of very low safety significance because there was no actual loss of a safety system function. The finding was associated with the cross-cutting area of human performance in that the resources and specifically work packages were not adequate to ensure that work performed restored the

component cooling water system to the analyzed condition (H.2(c)) after completion of maintenance activities.

Inspection Report# : 2007005 *(pdf)*

Significance: G Dec 31, 2007
Identified By: Self-Revealing
Item Type: FIN Finding
FAILURE TO IMPLEMENT RELEVANT OPERATING EXPERIENCE RESULTS IN EMERGENCY DIESEL GENERATOR TEST FAILURE
A self-revealing finding of very low safety significance was identified for the licensee’s failure to replace degraded emergency diesel generator (EDG) air start system hoses in accordance with operating experience (OE). Specifically, the licensee did not properly implement OE that recommended a 12-year lifespan for EDG air start hoses. This resulted in EDG2 failing to start during a monthly test due to an air leak in a hose leading to one of the air start motors. The OE was identified in 2001; at the time of the test failure, the leaking air hose had been installed on the EDG for more than 12 years. There was no violation of regulatory requirements. The licensee entered the issue into their corrective action program and replaced both the degraded hose and another similarly aged hose in the air start system. The finding was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The finding was of very low safety significance because it did not represent an actual loss of a safety function. The failure to replace the degraded hose is related to the cross-cutting element of problem identification and resolution, particularly the implementation of operating experience (P.2(b)) component in that the licensee did not implement and institutionalize relevant OE through changes to station processes, procedures, and equipment.
Inspection Report# : 2007005 *(pdf)*

Significance: G Nov 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Battery Voltage Drop and Sizing Design Calculation
The inspectors identified a finding having very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” in that, the licensee’s measures for verifying the adequacy of design with respect to the battery voltage drop calculations were inadequate. Specifically, the design inputs used in the battery calculation did not assure that adequate voltage would be available to all safety-related loads during a design basis accident condition. This issue was entered into the licensee’s corrective action program.

This finding was more than minor because the finding affected the design control attribute of the Mitigating Systems Cornerstone and if left uncorrected it would become a more significant safety concern in that the batteries would not provide adequate voltage to ensure the availability, reliability, and capability of safety related components to respond to initiating events to prevent undesirable consequences. The finding was of very low safety significance based on a Phase 1 screening in accordance with
IMC 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations,” because the batteries were relatively new and aging was not a current concern. This finding has a cross-cutting aspect in the area of Human Performance, Resources, because the licensee failed to maintain long term plant safety by maintenance of design margins (H.2(a)).

Inspection Report# : 2007007 *(pdf)*

Significance: G Nov 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Periodic Testing of 480V Starter Coils Not Implemented
The inspectors identified a finding having very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control.” Specifically, the licensee failed to assure and verify, following a design basis accident and degraded voltage condition, the minimum available control voltage at the 480 volts alternating current (Vac) motor control center was adequate to energize (pickup) the starter coils. This issue was entered into the licensee’s corrective action program to re-evaluate the schedule for periodic testing to verify the required pickup

voltage for starter coils over the life of the devices.

This finding was more than minor because the failure to assure adequate control voltage was available to energize the starter coils to supply 480 Vac power to safety-related equipment would have affected the capability of the equipment to respond to initiating events. The finding was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations.”
Inspection Report# : 2007007 *(pdf)*

Significance: G Nov 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Adequately Consider Potential Air Entrainment to ECCS during Suction Transfer
The inspectors identified a finding having very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control.” Specifically, the design bases analyses for the transfer of the emergency core cooling system pumps from the borated water storage tank (BWST) to the containment sump did not address the potential of air entrainment under the most limiting conditions. The calculation failed to consider the potential of additional gravity flow directly from the BWST to the containment sump during the suction transfer. As a result, this design basis calculation did not bound the potential air entrainment due to vortexing in the BWST. This issue was entered into the licensee’s corrective action program, and a prompt operability determination was performed to verify system operability.

This finding was more than minor because the existing design analyses did not fully address the potential of air entrainment during the transfer from the BWST to the containment sump. The finding was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations,” because on re-evaluation, the design function was maintained. This finding has a cross-cutting aspect in the area of Human Performance, Resources, because the licensee failed to maintain long term plant safety by maintenance of design margins (H.2(a)).
Inspection Report# : 2007007 *(pdf)*

Significance: G Nov 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Adequately Evaluate Postulated Failure of AFW Suction Piping
The inspectors identified a finding having very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” in that, the design bases analyses for the turbine driven auxiliary feedwater (AFW) pumps suction pressure switch setpoint did not adequately evaluate a postulated failure of the pumps’ common suction piping in the turbine building. Specifically, the licensee failed to consider the loss of inventory that could result from this piping failure. As a result, this design basis calculation did not adequately demonstrate that the turbine driven AFW pumps would be protected from the air entrainment due to this postulated event. This issue was entered into the licensee’s corrective action program, and a prompt operability determination was performed to verify system operability.

This finding was more than minor because the existing design did not adequately protect the turbine driven AFW pumps from the postulated failure of non-safety-related piping in the turbine building. The finding was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations,” because on re-evaluation, the design function was maintained. This finding has a cross-cutting aspect in the area of Human Performance, Resources, because the licensee failed to maintain long term plant safety by maintenance of design margins (H.2(a)).
Inspection Report# : 2007007 *(pdf)*

Significance: G Nov 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Battery Connection Resistance Limit Specified in Technical Specifications Surveillance Requirements Insufficient to Ensure Battery Functionality

The inspectors identified a finding having very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." Specifically, the licensee failed to verify and ensure that the 125 Vdc safety related batteries would remain operable if all the inter-cell and terminal connections were at the resistance value (150 micro-ohms) allowed by Technical Specifications (TS) surveillance requirement (SR) 4.8.2.3.2.b.2 and SR 4.8.2.3.2.c.3. This issue was entered into the licensee's corrective action program.

The finding was more than minor because if left uncorrected, the finding could become a more significant safety concern. Specifically, the 125 Vdc safety-related batteries would become incapable of meeting their design basis function if the inter-cell and connection resistance were allowed to increase to the TS allowed value. The finding was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," because the batteries were relatively new and the recorded inter-cell and terminal connection resistance are not currently significant. This finding has a cross-cutting aspect in the area of Human Performance, Resources, because the licensee failed to maintain long term plant safety by maintenance of design margins (H.2(a)).
Inspection Report# : 2007007 *(pdf)*

Significance: G Jun 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
IMPROPER IMPLEMENTATION OF INDEPENDENT VERIFICATION REQUIREMENTS IN PERFORMANCE OF INSTRUMENT AND CONTROL SURVEILLANCE TEST PROCEDURES FOR TS REQUIRED MITIGATION SYSTEMS
A non-cited violation (NCV) of Technical Specification 6.8.1 was identified by the NRC regarding adherence to the procedural requirements for independent verifications required by safety-related surveillance procedures for instrumentation and control mitigation systems. The licensee used procedure-step verification techniques in their instrumentation and control department that were not in compliance with their procedures. Upon identification, the licensee entered the issue into their corrective action program and instructed personnel to use the procedure-required independent verification methodology. The finding was more than minor because the finding was associated with the configuration control and testing procedure quality attributes of the mitigating systems cornerstone. This finding affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The improper completion of procedure-required verifications provided less than adequate assurance that important components of mitigation systems were properly positioned. The inspectors determined that the finding was of very low safety significance because there was no actual loss of safety function of mitigation systems. The inspectors also determined that the finding affected the cross-cutting area of human performance. The licensee's work practices did not support effective communication of the proper application of human error prevention techniques specified in instrument testing procedures, and supervisory oversight of the instrument testing work did not support proper application of the specified technique (H.4(b)).
Inspection Report# : 2007003 *(pdf)*

Barrier Integrity

Significance: G Jun 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
LICENSED REACTOR THERMAL POWER EXCEEDED DURING NORMAL PLANT OPERATIONS
A self-revealing NCV of the plant operating license was identified during normal plant operations when on June 8, 2007, control room personnel observed that the plant's computer was not scanning reactor coolant letdown flow after work was performed to upgrade computer programs. Letdown flow was a variable used in the computer's calculation of reactor core power. The period of time that the variable was not being scanned was approximately 15 hours. That caused calculated reactor core power to be displayed as 0.15 percent lower than actual, which resulted in the plant exceeding 100 percent power when averaged over an 8-hour period. Exceeding an 8 hour average of 100 percent power was a violation of the plant operating license. The finding was more than minor because it was associated with the fuel cladding thermal limits design control attributes of the barrier integrity cornerstone and did affect the cornerstone objective of reasonable assurance that the fuel cladding physical design barrier provide protection from

radio nuclide release caused by accidents or events. The finding is of very low safety significance because the issue did not have any measurable impact on the fuel cladding. This finding was also associated with the cross-cutting area of human performance because in the work control process the operational impact of computer-upgrade work activities, that affected calculated reactor core power, was not appropriately considered (H.3(b)).

Inspection Report# : 2007003 *(pdf)*

Emergency Preparedness

Significance: G Jun 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
OUT OF SERVICE SEISMIC FORCE MONITORING EQUIPMENT AFFECTING EMERGENCY PLAN RESPONSE
Inspectors identified an NCV of 10 CFR 50.54(q) and 50.47.b(4) for the failure to provide alternate event assessment methods while the seismic force monitor was out-of-service during the period of March 29 through April 10, 2007. The licensee failed to provide a means for the emergency director to promptly classify seismic events at the alert or site area emergency levels while the seismic force monitor utilized by the operators (emergency director) was out of service. The licensee restored the seismic force monitor to service on April 10, 2007, which restored assessment capability. The issue was more than minor because it was associated with the response organization planning standards attribute of the emergency preparedness cornerstone. This issue affected the cornerstone objective of ensuring that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. The finding is of very low safety significance because it did not result in the failure or degradation of a risk significant planning. Also, the unavailability of the seismic monitor did not prevent the declaration of a Site Area Emergency or Alert classification. This finding was also associated with the cross-cutting area of human performance. Licensee's work control process failed to establish compensatory measures for the out-of-service duration of the seismic force monitor (H.3(a)).

Inspection Report# : 2007003 *(pdf)*

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

Miscellaneous