Davis-Besse
4Q/2007 Plant Inspection Findings

Initiating Events

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: 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)*

Significance: G Mar 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO CONDUCT SIMULATOR MALFUNCTION PERFORMANCE TESTING IN A SUFFICIENT MANNER TO DEMONSTRATE FIDELITY
The inspectors identified a Non-Cited Violation (NCV) of 10 CFR 55.46(d)(1), "Continued assurance of simulator

fidelity," when the facility licensee failed to conduct a simulator "Generator Trip" malfunction performance test in a manner sufficient to ensure that simulator fidelity had been demonstrated and met. The "Generator Trip" malfunction performance test is one of 25 tests (Item Number 16) required by Section 3.1.4 in ANSI/ANS-3.5-1998, "Nuclear Power Plant Simulators for Use in Operator Training." The facility licensee is committed to adhering to the requirements of this standard. Specifically, the licensee failed to adequately conduct the required "Generator Trip" malfunction performance testing to ensure that simulator fidelity was demonstrated and met to allow conduct of the generator trip evolution. The licensee's corrective actions included revising the simulator "Generator Trip" test procedure, and then performing the revised procedure to adequately test the generator trip malfunction. This finding was considered more than minor because of the realistic potential of providing negative training based on significant simulator deficiencies compared to the actual plant. This resulted from inadequate testing of the simulator to assure that the simulator appropriately replicated the actual plant and would not negatively affect operator actions on the actual plant. The finding was of very low safety significance because the discrepancy was on the simulator and the real plant functioned properly. Furthermore, no actual plant emergency occurred and there was no actual impact on equipment or personnel safety.
Inspection Report# : 2007002 *(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

Significance: G Mar 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE CORRECTIVE ACTIONS (ISFSI)
The inspectors identified a Severity Level IV Non-Cited Violation of the Certificate of Compliance, No. 1004, Condition 1.1.3, Quality Assurance and 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for failure to correct a condition adverse to quality. Specifically, the licensee failed to remove transient combustible material within 50 feet from the Horizontal Storage Modules (HSMs) to restore compliance with the NRC issued 10 CFR Part 72 license and its fire protection procedure. After the issue was identified, the licensee took immediate corrective actions to remove all transient combustible material inside the 75-foot zone around the HSMs and generated a condition report to enter this issue into the corrective action program. This finding was more than minor because the lack of adequate corrective actions resulted in a more significant safety concern since the prolonged presence of combustible materials within 50 feet of HSMs for approximately 10 months increased the vulnerability of the HSMs to a fire. In addition, the lack of adequate corrective actions had the potential to become a programmatic issue and could have adversely affected NRC regulatory oversight and enforcement processes, as the agency relied on the licensee's adequacy of corrective actions to correct an NRC identified violation. The inspectors determined that the finding was not suitable for SDP evaluation because the noncompliance involved 10 CFR Part 72 dry fuel storage activities. Therefore, this finding was reviewed by Regional Management and dispositioned using traditional enforcement. The finding was determined to be of very low safety significance. The combustible material was contained within metal containers which could have mitigated the spread of a potential fire. Also, the plant fire brigade could have been dispatched to extinguish a fire involving the transient combustible material before the HSMs incurred significant damage. The primary cause of this finding was related to the cross-cutting area of problem identification and resolution because licensee personnel failed to thoroughly evaluate the problem (P.1(c)).
Inspection Report# : 2007002 *(pdf)*

Last modified : February 04, 2008