Davis-Besse
4Q/2016 Plant Inspection Findings

Initiating Events

Significance: G Jun 30, 2016
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
Inadequate Evaluation of Trend Related to A25X Fuse Failures
A self-revealed finding of very low safety significance and an associated NCV of Title 10, Code of Federal Regulations (CFR), Part 50, Appendix B, Criterion XVI, "Corrective Action," were identified for the licensee's failure to have adequately addressed an identified adverse trend involving 10 and 15 ampere Gould – Shawmut A25X series fuses. Specifically, the licensee had identified adverse trends related to failures of A25X series fuses in 2005, and again in 2015, and had entered these adverse trends into their corrective action program (CAP) as Condition Reports (CRs) 2005–05314 and 2015–03516. However, the evaluation performed under CR 2015–03516 did not recognize that the fuse failures were occurring much more frequently than originally anticipated and that the previously created Preventative Maintenance (PMs) were not adequate to prevent failures. Additionally, the evaluation did not adequately incorporate industry experience that also identified a trend of failures with the A25X series fuses. Corrective actions by the licensee included replacement of the existing stock of uninstalled A25X series fuses with equivalent fuses of a different style and from a different manufacturer and identification of a plan to replace in plant installed fuses.

This finding was of more than minor safety significance because it affected the attribute of equipment performance of the Initiating Events cornerstone of reactor safety, and adversely impacted the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The finding was determined to be of very low safety significance because it did not represent a deficiency that caused a reactor trip as well as the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition (e.g. loss of condenser, loss of feedwater, etc.) The inspectors determined that the finding had a cross cutting aspect in the area of problem identification and resolution. The inspectors assigned the cross cutting aspect of "Resolution" to the finding because the licensee failed to take action to resolve the identified adverse trend associated with premature failures of A25X series fuses in a timely manner. (P.3)

Inspection Report# : 2016002 *(pdf)*

Significance: G Jun 30, 2016
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
Inadequate Post-Manufacture Quality Control Inspections Performed for ASME Section III, Class 2 Reactor Coolant Pump Seal Cavity Vent Line Flexible Hose
A self-revealed finding of very low safety significance and an associated NCV of Technical Specification (TS) 3.4.13, "Reactor Coolant System (RCS) Operational Leakage," were identified for the failure of a licensee vendor to have ensured that a replacement reactor coolant pump (RCP) seal cavity vent line flexible hose was subjected to adequate quality control testing following manufacture. Specifically, a manufacturing weld defect on the flexible hose assembly for the RCP No. 1–1 first stage seal cavity vent line was not detected by post manufacture testing, such that the hose developed a very minor leak during power operations for the reactor operating cycle occurring before the licensee's

spring 2016 refueling outage. Corrective actions by the licensee included replacement of the failed flexible hose assembly and revising the procurement requirements for subsequently ordered flexible hose assemblies to include enhanced helium tracer probe leak testing.

This finding was of more than minor safety significance because it affected the attribute of equipment performance of the Initiating Events cornerstone of reactor safety, and adversely impacted the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The finding was determined to be of very low safety significance because it was determined that the finding could not have resulted in exceeding the RCS leak rate for a small loss of coolant accident or affected other systems used to mitigate a loss of coolant accident. The inspectors determined that the finding had a cross cutting aspect in the area of human performance. The inspectors assigned the cross cutting aspect of "Field Presence" to the finding because the licensee failed to ensure that their vendor performed adequate quality control testing following manufacture of a safety related flexible hose assembly. (H.2)

Inspection Report# : 2016002 *(pdf)*

Mitigating Systems

Significance: G Dec 01, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
Fire Hazards Analysis Report Incorrectly Described Rooms 511 and 512 as being Continuously Staffed
The inspectors identified a finding of very-low safety significance (Green), and associated Non-Cited Violation of License Condition 2.C.(4) for the licensee's failure to implement and maintain the Fire Protection Program as described in the Updated Final Safety Analysis Report. Specifically, the current Fire Hazards Analysis Report incorrectly listed rooms 511 and 512 as not requiring a separate fire watch, for fire protection impairments, because the rooms were incorrectly assumed to be continuously staffed or visible to the continuously staffed area. The licensee entered this issue into their Corrective Action Program and updated the Fire Hazards Analysis Report to reflect the current operating practice and deleted rooms 511 and 512 from the list of rooms that were continuously staffed.
The inspectors determined that the performance deficiency was more-than-minor because if left uncorrected, it could become a more significant safety concern for the failure to maintain the defense-in-depth element for the Fire Protection Program. The lack of fire watches degraded the ability to recognize conditions which could either increase the likelihood of a fire or the severity of a fire. The finding was representative of a low degradation and screened as having very low safety significance (Green) in Task 1.3.1 of IMC 0609, Appendix F. The finding did not have a cross-cutting aspect associated with it because it was not reflective of current performance. (Section 1R05.10b)
Inspection Report# : 2016009 *(pdf)*

Significance: G Sep 30, 2016
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
Inadequate Instructions to Correctly Assemble Electrical Conductor Seal Assemblies
A self-revealed finding of very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," were identified for the licensee's failure to provide adequate instructions to correctly assemble electrical conductor seal assemblies (ECSAs) used to provide an environmental barrier for resistance temperature detectors (RTDs). Specifically, the midlock ferrules inside two ECSAs were installed backwards during the 18th refueling outage (RFO) in 2014 which rendered multiple post accident monitoring

system (PAMS) indications required by Technical Specification (TS) 3.3.17 inoperable. This issue was entered into the licensee's corrective action program (CAP). Corrective actions by the licensee included, but were not limited to, replacement of the two dual element RTDs impacted and their associated ECSAs during the 2016 RFO, performance of an extent of condition review, development of enhanced procedural guidance, and implementation of additional training on ECSA components.
This finding was of more than minor significance because it was associated with the cornerstone attribute of equipment performance, and adversely 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 inspectors determined the finding to be of very low safety significance because it did not represent a deficiency affecting design or qualification of a mitigating system, structure, and component (SSC); it did not represent a loss of system and/or function; it did not represent an actual loss of function for at least a single train for more than its TS allowed outage time; and it did not represent an actual loss of function of one or more non-TS trains of equipment designated as high safety-significant in accordance with the licensee's maintenance rule program. The inspectors determined that the finding had a cross-cutting aspect in the area of human performance. Specifically, the cross-cutting aspect of "Training" was assigned to the finding because a job task analysis was performed prior to the 2014 RFO and determined that the procedural guidance to correctly assemble the ECSAs was adequate; thus no training or procedural changes were required. But the as-found condition of the RTDs during the 2016 RFO identified that a knowledge gap and procedure deficiency existed. (H.9)

Inspection Report# : 2016003 *(pdf)*

Significance: G Sep 30, 2016
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
Inadequate Modification Design Control Measures Result in Reactor Protection System Inoperability
A self-revealed finding of very low safety significance and an associated NCV of Title 10, Code of Federal Regulations (CFR), Part 50, Appendix B, Criterion III, "Design Control," were identified for the licensee's failure to have adequately prepared and implemented a permanent plant modification associated with steam generator (SG) replacement during the unit's 18th RFO in 2014. Specifically, in conjunction with SG replacement the licensee had also replaced a significant amount of reactor coolant system (RCS) piping and instrumentation, including all RCS hot leg resistance temperature detectors (RTDs). The RTD housings were improperly insulated during the modification, such that over the ensuing reactor operating cycle the RTD wiring insulation degraded to the extent that nearly all the RTDs were rendered inoperable. This issue was entered into the licensee's CAP. Corrective actions by the licensee included replacement of the degraded RTDs.
This finding was of more than minor safety significance because it affected the attribute of design control of the Mitigating Systems cornerstone of reactor safety, and adversely impacted the cornerstone objective of ensuring the availability, reliability, and capability of the unit's RPS. Specifically, the inspectors determined that the licensee's failure to have properly designed and implemented the insulation packages for the RTD housings ultimately resulted in the overheating and degradation of the RTD wiring insulation and inoperability of the RTDs associated with the RCS high temperature and RCS pressure/temperature reactor trips. The finding was determined to be of very low safety significance based on a detailed risk analysis that yielded a change in core damage frequency (CDF) of less than 1E–7 events per year. The inspectors determined that the finding had a cross cutting aspect in the area of human performance. The inspectors assigned the cross cutting aspect of "Field Presence" to the finding because the licensee's SG replacement project management team failed to reinforce the importance of close communication between responsible engineers with overlapping and interfacing modification packages, and did not adequately promote effective work execution through the use of clearly defined work documents that were written and structured to minimize the likelihood for human error. (H.2)

Inspection Report# : 2016003 *(pdf)*

Significance: G Jun 30, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
Failure to Use the Corrective Action Program to Evaluate and Document Degraded Condition with Auxiliary Feedwater Train 2
An NRC identified finding of very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” were identified for the licensee’s failure to have entered a degraded condition associated with Auxiliary Feedwater (AFW) Train No. 2 into their CAP until challenged by the inspectors. Specifically, a flow transient that occurred on May 7, 2016, and that caused damage to components in the AFW recirculation line during AFW Train No. 2 testing was not entered into the licensee’s CAP until May 8, 2016, following challenges from the inspectors. This omission on the part of the licensee’s staff had the effect of bypassing certain features of the licensee’s CAP associated with evaluating and documenting the operability of safety related equipment. The physical event and equipment issues were entered into the licensee’s CAP as CR 2016–06515 on May 8, 2016, following prompting by the inspectors. Corrective actions taken by the licensee included repairs to all damaged equipment, detailed inspections of AFW Train No. 2, and an engineering analysis into why the event occurred. The matter of the licensee’s failure to enter the event into their CAP in a timely manner was documented as CR 2016–06516, with corrective actions including the coaching and counseling of personnel involved regarding the proper use of the CAP.

This finding was of more than minor safety significance because it affected the equipment performance attribute of the Mitigating Systems cornerstone of reactor safety and adversely impacted the cornerstone objective of ensuring the availability, reliability, and capability of the unit’s AFW system. The finding was determined to be of very low safety significance because it did not represent a deficiency affecting the design or qualification of a mitigating system, structure, or component (SSC); it did not, in and of itself, represent a loss of system and/or function; it did not represent an actual loss of function of at least a single train for greater than its TS allowed outage time, or two separate safety systems being out-of-service for greater than their TS allowed outage times; and it did not represent an actual loss of function of one or more non TS trains of equipment designated as high safety significant in accordance with the licensee’s maintenance rule program. The inspectors determined that the finding had a cross cutting aspect in the area of problem identification and resolution. The inspectors assigned the cross cutting aspect of “Identification” to the finding because the licensee’s staff failed to identify the issue with AFW Train No. 2 within their CAP completely, accurately, and in a timely manner in accordance with program requirements. (P.1)
Inspection Report# : 2016002 *(pdf)*

Significance: G Mar 31, 2016
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
OPERATION OF SAFETY RELATED BUTTERFLY VALVES IN A MANNER BEYOND DESIGN
A self-revealed finding of very low safety significance (Green), and an associated NCV of Title 10, Code of Federal Regulations (CFR), Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” were identified for the licensee’s failure to incorporate applicable manufacturer’s limits into the operating procedures and instructions for the service water (SW) outlet isolation/throttle valves for Component Cooling Water (CCW) Heat Exchanger (HX) Nos. 1, 2, and 3 (SW36, SW38, and SW37). Specifically, the licensee’s procedural guidance for the operation of these valves allowed them to be throttled beyond the manufacturer’s recommended limits, and repeated operation of the SW37 valve in this manner beyond its design contributed to its failure. This issue was entered into the licensee’s corrective action program (CAP). Corrective actions by the licensee included repair of the SW37 valve.

This finding was of more than minor safety significance because it affected the attributes of design control and procedure quality of the Mitigating Systems cornerstone of reactor safety, and adversely impacted the cornerstone objective of ensuring the availability, reliability, and capability of the unit’s CCW system. Specifically, the inspectors determined that the licensee's failure to have incorporated the applicable design limits for SW37 throttle position and differential pressure across the valve into applicable operating procedures contributed to the degradation and ultimate inoperability of the valve. The finding was determined to be of very low safety significance since the finding did not result in a loss of operability of any system or component. The inspectors determined that the finding had a cross cutting aspect in the area of human performance. The inspectors assigned the cross cutting aspect of "Design Margins" to the finding because the licensee had failed to ensure that the safety related SW37 butterfly valve was operated and maintained well within the manufacturer's design limits. (H.6)

Inspection Report# : 2016001 *(pdf)*

Significance: G Mar 31, 2016
Identified By: Self-Revealing
Item Type: FIN Finding
LESS THAN SUFFICIENT WORK PACKAGE DOCUMENTATION AND INSTRUCTIONS RESULTED IN AN INADEQUATE PART BEING INSTALLED INTO THE PLANT'S INTEGRATED CONTROL SYSTEM
A self-revealed finding of very low safety significance (Green) was identified for the licensee's failure to include an adequate bench check for a replacement integrated control system (ICS) module that was installed into the system during the plant's 2014 refueling outage (RFO) into the work package instructions for that activity. Specifically, a defeat switch on the replacement Module 5–2–8 for the ICS rapid feedwater reduction (RFR) circuit installed as preventative maintenance during the plant's 18th RFO was incorrectly wired and not detected during pre installation checks. The incorrectly wired module prevented the ICS RFR function from occurring during the unit trip on January 29, 2016, which contributed to the Steam Generator (SG) No. 1 high level condition and the resultant steam and feedwater rupture control system (SFRCS) actuation. This issue was entered into the licensee's CAP. Corrective actions taken by the licensee included replacement of ICS Module 5–2–8 with a spare properly configured for the RFR defeat switch function. Additionally, a proper data package to enable bench checking ICS Module 5–2–8 to verify the capability of the module to perform its intended function was created. The licensee also created training and lessons learned from this event.

This finding was of more than minor safety significance because it affected the design control and procedure quality attributes of the Mitigating Systems cornerstone of reactor safety, and adversely impacted the cornerstone objective of ensuring the availability, reliability, and capability of the unit's main feedwater (MFW) system and main condenser for decay heat removal. The finding was determined to be of very low safety significance because it did not represent a deficiency affecting the design or qualification of a mitigating system, structure, or component (SSC); it did not, in and of itself, represent a loss of system and/or function; it did not represent an actual loss of function of at least a single train for greater than its Technical Specification (TS) allowed outage time, or two separate safety systems being out-of-service for greater than their TS allowed outage times; and it did not represent an actual loss of function of one or more non TS trains of equipment designated as high safety significant in accordance with the licensee's maintenance rule program. The inspectors determined that the finding had a cross cutting aspect in the area of human performance. The inspectors assigned the cross cutting aspect of "Documentation" to the finding because the licensee had failed to ensure that the instructions and other work package guidance available to maintenance personnel performing the ICS Module 5–2–8 replacement had contained provisions for an adequate bench check of the module prior to its installation. (H.7)

Inspection Report# : 2016001 *(pdf)*

Significance: G Mar 31, 2016
Identified By: Self-Revealing
Item Type: FIN Finding
LACK OF SOFTWARE CHANGE CONTROLS AND INADEQUATE CORRECTIVE ACTION FOR AN

OPERATOR WORKAROUND CONTRIBUTES TO COMPLICATIONS EXPERIENCED DURING A REACTOR TRIP

A self-revealed finding of very low safety significance (Green) was identified for the licensee's failure to implement a technically correct software change associated with the SG / Reactor Demand ICS control station. Specifically, a known logic error within the plant's ICS would cause the SG / Reactor Demand control station to trip to manual from automatic coincident with a reactor trip. The licensee had instituted compensatory operator actions for this condition, but removed these actions in December 2015 when they implemented a software change to rectify the problem. However, the corrective actions were inadequate and the SG / Reactor Demand ICS control station unexpectedly tripped to manual from automatic when the unit tripped on January 29, 2016. The unexpected control station mode of operation change, combined with the absence of any compensatory operator actions, contributed to the SG No. 1 high level condition and the resultant SFRCS actuation. This issue was entered into the licensee's CAP. Corrective actions taken by the licensee included initiating work on a new software change to rectify the issue of the SG / Reactor Demand ICS control station tripping from automatic to manual coincident with a reactor trip; reestablishing the operator workaround and associated compensatory actions for control room operators; and revising applicable procedures to incorporate current industry standards for controlling software life cycle changes to certain categories of software that interface with plant systems.

This finding was of more than minor safety significance because it affected the design control and procedure quality attributes of the Mitigating Systems cornerstone of reactor safety and adversely impacted the cornerstone objective of ensuring the availability, reliability, and capability of the unit's MFW system and main condenser for decay heat removal. The finding was determined to be of very low safety significance because it did not represent a deficiency affecting the design or qualification of a mitigating SSC; it did not, in and of itself, represent a loss of system and/or function; it did not represent an actual loss of function of at least a single train for greater than its TS allowed outage time, or two separate safety systems being out-of-service for greater than their TS allowed outage times; and it did not represent an actual loss of function of one or more non TS trains of equipment designated as high safety significant in accordance with the licensee's maintenance rule program. The inspectors determined that the finding had a cross cutting aspect in the area of problem identification and resolution. The inspectors assigned the cross cutting aspect of "Evaluation" to the finding because the licensee had failed to thoroughly evaluate the issue of the SG / Reactor Demand ICS control station unexpectedly tripping from automatic to manual to ensure that the software change intended to resolve the issue actually addressed its cause. (P.2)

Inspection Report# : 2016001 *(pdf)*

Significance: G Mar 31, 2016
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation

LESS THAN ADEQUATE PROCEDURAL INSTRUCTIONS FOR RESTORING MAIN FEEDWATER FOLLOWING A REACTOR TRIP

A self-revealed finding of very low safety significance (Green), and an associated NCV of TS 5.4.1(a) were identified for the licensee's failure to establish and implement adequate procedural guidance for restoring MFW following a reactor trip. Specifically, the guidance in licensee procedure DB–OP–06910, "Trip Recovery Procedure," for restoring MFW to the SGs using the motor driven feedwater pump (MDFP) did not ensure that the MFW piping had been sufficiently re pressurized prior to opening the MFW to SG isolation valves. This lack of satisfactory procedural guidance allowed control room operators to prematurely open the MFW to SG No. 1 isolation valve, which resulted in a SFRCS actuation on the reverse delta pressure (?P) function. This issue was entered into the licensee's CAP. Corrective actions planned by the licensee included changes to licensee procedure DB–OP–06910, "Trip Recovery Procedure," to ensure that MFW header pressure is greater that SG pressure prior to opening the MFW to SG isolation valves.

This finding was of more than minor safety significance because it affected the design control and procedure quality attributes of the Mitigating Systems cornerstone of reactor safety, and adversely impacted the cornerstone objective of ensuring the availability, reliability, and capability of the unit's MFW system and main condenser for decay heat removal. The finding was determined to be of very low safety significance based on the results of a detailed risk evaluation conducted by the NRC Region III Senior Reactor Analyst (SRA). The inspectors determined that the finding had a cross cutting aspect in the area of human performance. The inspectors assigned the cross cutting aspect of "Resources" to the finding because the licensee had failed to ensure that the procedural instructions and guidance available to plant operators restoring MFW during reactor trip recovery actions took into account all relevant technical details (e.g., the differences between MFW piping runs, the amount of time needed to re pressurize MFW piping, etc.) (H.1)

Inspection Report# : 2016001 *(pdf)*

Barrier Integrity

Significance: G Oct 01, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
10 CFR 50.59 Evaluation Failed to Consider Change to Seismic Licensing Basis
A finding of very-low safety significance and an associated NCV of Title 10 of the Code of Federal Regulations (CFR), Part 50.59(b)(1), "Changes, Tests, and Experiments," (effective January 1, 1991) was identified by the inspector for the licensee's failure to maintain records that included a written safety evaluation which provided the bases for determining that the change to seismic licensing basis damping in calculations to support removal of snubbers under modification 90-0079 did not involve an unreviewed safety question. Specifically, licensee safety evaluation SE91-0046 did not provide a suitable basis for concluding that there was no increase in the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety analysis report, in that it did not address how the basis for the NRC's-approval of the seismic design of the reactor coolant system continued to be met with respect to the steam generator slider support (Lubrite plate) damping. In particular, a May 31, 1983, NRC Safety Evaluation Report approved the licensee's use of 0.15g safe shutdown earthquake ground acceleration in its seismic analysis for reactor coolant system design, in part, because "there is sufficient conservatism and margin in the piping systems components and supports at Davis-Besse Unit 1 to ensure safe shutdown and continued shutdown heat removal in the event of a safe shutdown earthquake having a ground acceleration of 0.20g." The licensee subsequently adopted a significantly higher damping value for the steam generator slider support while maintaining a 0.15g acceleration for the design without addressing how "sufficient conservatism and margin" otherwise continued to be met. The licensee entered this issue into its corrective action program.
The inspector determined that the licensee's failure to provide in its 10 CFR 50.59 evaluation, SE91-0046, a suitable basis for the determination that the use of damping higher than established in the seismic licensing basis for the reactor coolant system, specifically the steam generator slider support, was not an unreviewed safety question was a performance deficiency. The issue of concern was determined to be more than minor because the performance deficiency impacted the Barrier Integrity cornerstone objective to provide reasonable assurance that physical design barriers (reactor coolant system) protect the public from radionuclide releases caused by accidents or events and the design control attribute to maintain functionality of the reactor coolant system. The inspector evaluated the underlying technical issue using IMC 0609, "The Significance Determination Process for Findings at Power," Appendix A, Exhibit 1, "Initiating Events Screening Questions." The inspector answered "No" to all the questions in Exhibit 1. In particular, because the reactor coolant system remained operable (capable of performing its safety function during a seismic event), the finding was determined to have very-low safety significance (Green) corresponding to a Severity Level IV violation per Example 6.1.d.2 of the NRC Enforcement Policy. The inspector did not identify a cross-cutting aspect associated with the finding because the finding was not representative of current performance. (Section 4OA5.1).

Inspection Report# : 2016010 *(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

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Miscellaneous

Last modified : February 01, 2017