Davis-Besse
1Q/2016 Plant Inspection Findings

Initiating Events

Significance: G Sep 30, 2015
Identified By: Self-Revealing
Item Type: FIN Finding
FLOW ACCELERATED CORROSION MODEL NOT MAINTAINED IN ACCORDANCE WITH INDUSTRY STANDARDS AND GUIDANCE
A self-revealed finding of very low safety significance was identified for the licensee’s failure to maintain an adequate flow accelerated corrosion (FAC) program in accordance with station procedures and applicable industry guidance. Specifically, an incorrect restriction orifice size entered into the FAC program software in the late 1980s significantly underestimated the wear rate of a section of moisture separator reheater (MSR) piping that ultimately failed causing control room operators to conduct a rapid power reduction and manual reactor trip and declare an unusual event in accordance with the station's emergency plan. The failed section of piping had not been previously inspected in accordance with industry guidance and station procedures, and the incorrect FAC program software inputs had never been validated. This finding was associated with the Initiating Events Cornerstone of reactor safety and was of more than minor significance because it directly impacted the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors evaluated the finding using IMC 0609, Appendix A, “The Significance Determination Process for Findings At-Power.” Using Exhibit 1, which contains the screening questions for the Initiating Events Cornerstone of Reactor Safety, the inspectors determined a detailed risk evaluation was required because the finding was a transient initiator that resulted in both a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition (i.e., the loss of the main condenser as a heat sink and the loss of main feedwater). The inspectors contacted the NRC Region III Senior Reactor Analyst (SRA) to perform a detailed risk evaluation. The assumed core damage sequence used by the SRA was that the MSR pipe break occurs, followed by either main steam isolation valve (MSIV) failing to close, followed by any of four in-series main turbine stop valves (SVs) and control valves (CVs) failing to close. Mathematically, the change in core damage frequency (?CDF) was estimated at:
?CDF = 1 (event occurs) x (9.51E-4 + 9.51E-4) x 4 x 1.5E-3 x 1.5E-3 = 1.71E-8/yr
The SRA concluded the risk associated with this performance deficiency was, therefore, of very low safety significance (Green). Because the causes for the finding stemmed from deficiencies going back several years or more, the inspectors concluded that the finding represented a latent issue not necessarily indicative of present licensee performance. As a result, no cross cutting aspect was assigned to this finding.
Inspection Report# : 2015003 *(pdf)*

Mitigating Systems

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: NRC
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)*

Significance: G Dec 31, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
FAILURE TO USE WORST CASE 4160 VAC BUS VOLTAGE IN DESIGN CALCULATIONS
The inspectors identified a finding of very low safety significance (Green), and an associated NCV of Title 10, Code of Federal Regulations (CFR), Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to have adequate analysis related to the degraded voltage relay (DVR) setpoints as specified in Technical Specifications. Specifically, the licensee's analysis failed to demonstrate that the DVR setpoints would provide adequate starting and running voltage to safety related equipment during the most limiting case accident loading. This issue was entered into the licensee's corrective action program (CAP). Corrective actions planned and completed by the licensee included analysis to determine the appropriate DVR setpoints and interim compensatory measures to maintain minimum voltage on 4160 volts alternating current (Vac) essential buses above 4070 Vac to ensure adequate voltage for safety related components. This finding was of more than minor safety significance because it affected the Design Control attribute of the Mitigating Systems Cornerstone of reactor safety, and adversely impacted the cornerstone objective of ensuring the availability, reliability, and capability of the site's 4160 Vac safety related electrical buses. Specifically, the licensee failed to perform and maintain an analysis demonstrating that all safety related loads had adequate starting voltage at the DVR setpoint. 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 there was no cross cutting aspect associated with this finding because the finding represented a legacy issue that was not representative of current licensee performance. (Section 4OA5.1)
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Inspection Report# : 2015004 *(pdf)*

Barrier Integrity

Significance: G Jun 30, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
Departure from Method of Evaluation Required Prior NRC Approval Under 10 CFR 50.59 (c)(2) with respect to licensee acceptance of the shield building laminar cracking.
Severity Level IV - Green. The inspectors identified a Severity Level IV NCV of Title 10, Code of Federal Regulations (CFR) Part 50.59(c)(2), and an associated finding of very low safety significance for the licensee’s failure to request and obtain a license amendment pursuant to 10 CFR 50.90. Specifically, the licensee’s method of evaluation that accepted shield building laminar cracking represented a departure from the method of evaluation described in the Final Safety Analysis Report (as updated), and required prior NRC approval with respect to the design and licensing basis. The licensee entered this finding into its Corrective Action Program; the licensee’s immediate corrective action determined that shield building remained operable and capable to perform its design safety functions; the licensee’s planned corrective actions included revising 10 CFR 50.59 Evaluation 13-00918, and preparation of additional documents for inclusion in a license amendment request.
The finding was determined to be more than minor because the finding was associated with the Barrier Integrity cornerstone attribute of Design Control, and affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The inspectors evaluated the finding using IMC 0609, Appendix A, “The SDP for Findings At Power.” Using Exhibit 3, the inspectors determined that the finding screened as very low safety significance because all the Reactor Containment screening questions for the Barrier Integrity Cornerstone were answered “No.” Specifically, the inspectors concluded that the shield building remained capable of performing its design safety functions despite the identified laminar cracking. The associated violation was categorized as Severity Level IV because the issue was determined to be of very low safety significance under the SDP. This finding had a cross-cutting aspect in the area of Human Performance, Conservative Bias, because the licensee did not take a conservative approach to decision making for evaluation of shield building laminar cracking, particularly when information is incomplete or conditions are unusual. [H.14, Conservative Bias] (Section 4OA2.2)

Inspection Report# : 2014008 *(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.

Miscellaneous

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