Davis-Besse
3Q/2015 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)*

Significance: G Jan 09, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
Failure to Install and Control RCP Seal Cavity Vent Flexible Hoses Per Design Basis Analysis
The inspectors identified a finding of very-low safety significance (Green) and associated Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control" for the licensee's failure to install and control the Reactor Coolant Pump (RCP) seal cavity vent flexible hoses per the design basis analysis. Specifically, the licensee failed to correctly translate the design basis installation configuration and installation fatigue analysis in calculation

SP-274-I, “Pipe Stress Analysis: Reactor Coolant Pump 1-1-1 Seal Cavity Vent,” into specifications, drawings, procedures, and instructions. The licensee entered this finding into their Corrective Action Program (CAP) to review the lack of controls over the installation of the flexible hoses, but determined that the flexible hoses remained operable.
The performance deficiency was determined to be more than minor because, if left uncorrected, it would have the potential to lead to a more significant safety concern. Specifically, the failure to install and control the flexible hoses in accordance with the design basis analysis could lead to failure of the hoses due to operation beyond their analyzed limits. The finding screened as of very-low safety significance (Green) because the finding could not result in exceeding the Reactor Coolant System (RCS) leak rate for a small Loss of Coolant Accident (LOCA) after a reasonable assessment of degradation, and it could not have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function after a reasonable assessment of degradation. The inspectors determined this finding had an associated cross-cutting aspect, Design Margins, in the Human Performance cross-cutting area. Specifically, the licensee did not carefully guard and change the RCP seal cavity vent lines, which form part of the RCS fission product barrier, through a systematic and rigorous process. [H.6] (Section 1R17.2.b.(1))

Inspection Report# : 2014007 *(pdf)*

Significance: G Dec 31, 2014
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
INADEQUATE PROCEDURAL GUIDANCE DURING RESTORATION FROM VALVE MAINTENANCE RESULTS IN FEEDWATER HEATER SYSTEM AND PLANT POWER TRANSIENT
A self-revealed finding of very low safety significance and associated NCV of Technical Specification (TS) 5.4.1(a) were identified when the licensee failed to provide proper procedural guidance for the restoration from valve maintenance on HD291G, a manual isolation valve for the level controller for HD291A, the emergency drain valve for High Pressure (HP) Feedwater Heater No. 1–4, on November 13, 2014. Specifically, the licensee's restoration instructions did not isolate HD291A prior to restoring its associated level controller. As a result, when a perturbation in the level controller during restoration caused HD291A to rapidly reposition to the fully open position, the resulting HP Feedwater Train 1 transient caused HP Feedwater Heaters 1–4, 1–5, and 1–6 to trip. The change in plant efficiency that resulted momentarily drove plant power slightly above 100 percent.
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, the inspectors determined that the finding screened as very low safety significance because all screening questions for the Initiating Events Cornerstone of reactor safety were answered “No.” This finding also was determined to have a cross-cutting component in the area of human performance, work management aspect, because during the work planning process for this maintenance activity the licensee failed to identify the risk associated with not isolating the HP Feedwater Heater No. 1–4 Emergency Drain Valve, HD291A, prior to restoring its associated level controller to service. (H.5)

Inspection Report# : 2014005 *(pdf)*

Mitigating Systems

Significance: G Feb 27, 2015

Identified By: NRC
Item Type: NCV Non-Cited Violation
Vulnerability of EDG Crosstie to a Non-Essential Bus (1R21.3.b.(1))
Green. 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 failure to assure that applicable regulatory requirements, and the design basis were correctly translated into specifications, drawings, procedures, and instructions, and verifying the adequacy of design. Specifically, the licensee failed to verify the adequacy of procedures controlling alignment of non essential busses to the emergency diesel generators during a design basis event. The procedure's guidance could put the plant in an unanalyzed alignment with the potential to result in the failure of safety-related equipment. The licensee entered this finding into their Corrective Action Program (CAP), and initiated a Standing Order to preclude the unanalyzed alignment.
Inspection Report# : 2015008 *(pdf)*

Significance: G Feb 27, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
Failure to Comply with IEEE 308-1971 for the Required Independence of Safety-Related Essential Inverter Distribution Systems (1R21.3.b.(2))
Green. The inspectors identified a finding of very-low safety significance (Green), and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to demonstrate compliance with Institute of Electrical and Electronics Engineers (IEEE) 308 1971, "IEEE Standard Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations," for the required independence of essential safety related inverter distribution system channels. Specifically, a common mode failure due to inadequate fault protection on several outside distribution panels could cause the loss of redundant safety-related inverters. The licensee entered this finding into their CAP, and initiated a Standing Order that would identify the affected circuit breakers and require they be opened based on a tornado warning that could potentially affect the site.
Inspection Report# : 2015008 *(pdf)*

Significance: G Feb 27, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
Failure to Incorporate the Design Analysis Required Acceptance Limit into Surveillance Procedure (1R21.3.b.(3))
Green. The inspectors identified a finding of very-low safety significance (Green), and an associated NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," for the licensee failing to incorporate the design requirements and acceptance limits into test procedures. Specifically, the design limit for the minimum voltage on essential safety related inverter YV1 bus during Modes 5 and 6 was not correctly incorporated into surveillance test procedures. The licensee entered this finding into their CAP and re analyzed for the 116 volts as-found value at panel Y1, and determined the loads that did not have adequate rated voltage at the surveillance procedure minimum acceptance criteria voltage, 114 volts, would have had sufficient voltage to operate at the as-found measured voltage to perform their intended safety function.
Inspection Report# : 2015008 *(pdf)*

Significance: G Feb 27, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
Failure to Verify Several CCW System Manual Valves Were in the Correct Position (1R21.3.b.(4))
Green. The inspectors identified a finding of very-low safety significance (Green), and an associated NCV of

Technical Specification (TS) Surveillance Requirement 3.7.7.1, for the licensee's failure to verify several component cooling water (CCW) system manual valves in the flow path servicing safety-related equipment that were not locked, sealed, or otherwise secured, were in the correct position every 31 days. Specifically, the unsecured CCW pump seal water flush isolation valves (two valves per pump) for the two required operable CCW pumps were not verified open every 31 days. The licensee entered this finding into their CAP, verified the correct position of the valves, and planned to revise the Locked Valve Program to include the requirement to have the valves in the locked open position.
Inspection Report# : 2015008 *(pdf)*

Significance: G Feb 27, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
Failure to Comply with Technical Specifications for the Borated Water Storage Tank (1R21.4.b.(1))
Green. The inspectors identified a finding of very-low safety significance (Green), and an associated NCV of TS 3.5.4, "Borated Water Storage Tank (BWST)," for the failure to comply with the limiting condition for operation (LCO) while the BWST was aligned to the non-seismic spent fuel pool purification system, causing the BWST to be inoperable based on no longer meeting the tank's seismic requirement. The licensee entered this finding into their CAP, and initiated an LCO Tracking Log entry to not place the BWST on spent fuel pool purification in Modes 1 through 4.
Inspection Report# : 2015008 *(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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