Palisades
3Q/2015 Plant Inspection Findings

Initiating Events

Significance: G Aug 19, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
Operability Evaluation Not Performed in Accordance with Station Procedure (Section 1R15)
Green. An NRC identified finding of very low safety significance and an associated NCV of Title 10 of the Code of Federal Regulations (CFR), Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," was identified for the licensee's failure to adhere to the site procedure for performing operability determinations during the evaluation of a nonconforming condition associated with nine primary coolant system (PCS) welds susceptible to primary water stress corrosion cracking (PWSCC). The licensee's corrective actions for this finding included completion of an operability determination in accordance with the site operability procedure to include a new analysis which demonstrated the AMSE Code acceptance criteria would continue to be met for the affected welds during the remainder of the operating cycle. The licensee entered the failure to comply with the operability procedure into the CAP (CR PLP-2015-03434).
This finding was determined to be more than minor because it was similar to the "not minor if" aspect of Example 3j in IMC 0612, Appendix E, "Example of Minor Issues," because the errors in operability evaluation CA-1 of CR-PLP-2015-01239 resulted in a condition in which there was a reasonable doubt on the operability of the systems and components that were the subject of the evaluation and dissimilar from the "minor because" aspect of this example since the impact of the errors on the operability evaluation was not minimal. In addition, the performance deficiency was determined to be more than minor because it was associated with the Initiating Event Cornerstone attribute of Equipment Performance and adversely affected the Cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions. The inspectors evaluated the finding in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," Table 3, for the Initiating Events Cornerstone and IMC 0609, Appendix A, "The SDP for Findings At-Power." Because the licensee was able to demonstrate operability of the nine PCS welds susceptible to PWSCC, the inspectors answered "No" to questions A.1 and A.2, of Exhibit 1, "Initiating Events Screening Questions," identified in Appendix A of IMC 609 and, as a result, the finding screened as having very low safety significance (Green). This finding has a cross-cutting aspect in Evaluation for the Problem Identification and Resolution cross-cutting area since the licensee failed to thoroughly evaluate the impact on operability of a nonconforming condition associated with nine PCS welds susceptible to PWSCC [IMC 310, Item P.2]. (Section 1R15)

Inspection Report# : 2015012 *(pdf)*

Significance: G Mar 31, 2015
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
Inadequate Procedure Leads to primary Coolant Pump Seal Degradation
A finding of very low safety significance and an associated NCV of Technical Specification (TS) 5.4.1(a) was self-revealed when the 'C' primary coolant pump (PCP) seal degraded as a result of an inadequate maintenance procedure. Specifically, maintenance procedure PCS–M–54, "N–9000 Primary Coolant Pump Shaft Seal Assembly," did not identify critical steps in the assembly of the PCP seal and, as a result, the work activity was not adequately controlled.

This issue was entered into the licensee's Corrective Action Program (CAP) as CR–PLP–2014–03495, Planned Outage Required Due to Two Stages of the Primary Coolant Pump P 50C Seal Not Performing as Expected, dated June 21, 2014.

The performance deficiency was determined to be more than minor because it was associated with the Equipment Performance attribute of the Initiating Events cornerstone and adversely affected 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. Specifically, the 'C' PCP seal was not correctly assembled or installed during refueling outage (RFO) 1R23, which resulted in premature seal degradation. Based on a detailed risk evaluation performed by a Region III Senior Reactor Analyst (SRA) using SAPHIRE Version 8.20 and the Events and Conditions Assessment Feature of the Palisades Standardized Plant Analysis Risk (SPAR) model (Version 8.1.2), the inspectors determined the finding was of very low safety significance. This finding had a cross-cutting aspect in the Work Management component of the Human Performance cross-cutting area. Specifically, the licensee did not effectively screen the PCP seal assembly through the work management process to identify that it should have been classified as a critical maintenance activity. In addition, insufficient emphasis was placed on in-field vendor oversight during work execution.

Inspection Report# : 2015001 *(pdf)*

Significance: G Feb 27, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
Failure to Determine the Cause of Head Penetration Nozzle J-Grove Weld Cracking (Section 4OA2.1)
Green: The inspector identified a finding of very-low safety significance with an associated NCV of Title 10, Code of Federal Regulations (CFR) Part 50, Appendix B, Criterion XVI, "Corrective Action," for the licensee's failure to establish measures to assure that the cause of the ultrasonic examination leakage path indications and crack indications identified in the J-groove welds of the reactor pressure vessel head penetration nozzles 29 and 30 (a significant condition adverse to quality) was determined. Specifically, the licensee did not complete adequate causal investigations to assure the cause of this significant condition adverse to quality was determined. The licensee entered this issue into the Corrective Action Program (CAP), and initiated an action to conduct a root cause investigation for this issue.

The issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," because it adversely affected the Initiating Events cornerstone attribute of equipment performance and procedure quality. The inspector also answered "Yes" to the more than minor screening question, "If left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, the inspector determined that this issue was more than minor because, if left uncorrected, the licensee would have reduced the frequency of reactor vessel head nozzle penetration examinations which could result in the failure to detect primary water stress corrosion cracking (PWSCC). Undetected PWSCC could increase the risk for through-wall leakage and design basis events such as a loss-of-coolant accident (LOCA). The inspector determined that the finding was of very-low safety significance based on answering "No" to the IMC 0609, Appendix A, Exhibit 1-Initiating Events Screening Questions for LOCA Initiators. Although this performance deficiency occurred more than 10 years ago, it was representative of current licensee performance because in the November 19, 2014, Licensee Event Report Cancelation Letter, the licensee again failed to assure that the cause of the reactor pressure vessel nozzle crack indications in the J-groove welds was determined. Therefore, the finding had a

cross-cutting aspect in the area of Problem Identification and Resolution because the licensee failed to assure the cause was determined for the reactor pressure vessel nozzle crack indications in the J-groove welds, and this decision was not consistent with an organization that thoroughly evaluates issues to ensure that resolutions address causes and extent of conditions commensurate with their safety significance (IMC 310-Item P.2). (Section 4OA2.1.b(1))
Inspection Report# : 2015009 *(pdf)*

Significance: G Feb 27, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
Unqualified Non-Destructive Examinations of J-Grove Welds 29 and 30 (Section 4OA2.1)
Green: The inspector identified a finding of very-low safety significance with an associated NCV of 10 CFR Part 50, Appendix B, Criterion IX "Control of Special Processes," for the licensee's failure to use qualified personnel and procedures for the dye penetrant (PT) examinations of the J-groove welds at nozzles 29 and 30 used to characterize crack indications. Consequently, no quality records existed to validate or confirm the size or extent of the cracking identified in these welds. The licensee documented the use of the unqualified PT examination for characterizing the reactor pressure vessel nozzle J-groove weld cracks in the CAP, and was developing corrective actions at the conclusion of the inspection.

The issue was determined to be more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," because it adversely affected the Initiating Events cornerstone attribute of equipment performance and procedure quality. Further, if left uncorrected, it would become a more significant issue. Specifically, the licensee had based the risk evaluation of the nozzle cracking on the results of the unqualified PT examination, and if this result was not correct, the risk significance of past plant operation with these cracks may have been greater than assumed. Additionally, the licensee had considered the results from this PT examination, as part of the evaluations identified in their November 19, 2014, letter that concluded the flaws identified were caused by embedded weld defects, and not PWSCC. Based upon this revised cause determination, the licensee had elected to reduce the scheduled vessel head examinations, and this reduced inspection schedule may not be adequate to identify PWSCC prior to experiencing a through-wall leak. The inspectors determined that the finding was of very low safety significance based on answering "No" to the IMC 0609, Appendix A, Exhibit 1-Initiating Events Screening Questions for LOCA Initiators. The finding did not have a cross-cutting aspect because it was not indicative of current licensee performance dudue to the age of the performance deficiency.
(Section 4OA2.1.b(2)).
Inspection Report# : 2015009 *(pdf)*

Significance: G Nov 04, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
Failure to Include the Degraded Voltage Channel Time Delay in TS Surveillance Requirement 3.3.5.2a
The inspectors identified a finding having very low safety significance and an associated Non-Cited Violation (NCV) of 10 CFR Part 50.36(c)(3), "Surveillance Requirements," for the failure to ensure the channel time delay for the degraded-voltage monitor was included in Technical Specification (TS) Surveillance Requirement (SR) 3.3.5.2.a.

Specifically, the licensee failed to include in the TS SR the required time delay after the voltage relay trips before the preferred source of power is isolated and 1E electrical loads transferred to the stand-by Emergency Diesel Generators (EDGs). This finding was entered into the licensee's Corrective Action Program and the licensee's preliminary verification determined the degraded voltage monitors were still operable but degraded or non-conforming.

The performance deficiency was determined to be more than minor because if left uncorrected, it would have the potential to lead to more significant safety concern. Specifically, by not incorporating the total time delay requirements into the Technical Specifications, (TS) the time could be changed without going through the TS change process, possibly leading to spurious trips of offsite power sources or possibly exceeding the accident analysis time is the FSAR. The inspectors determined the finding was of very low safety significance (Green) because it did not cause 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. The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of the licensee's present performance.

Inspection Report# : 2014008 *(pdf)*

Mitigating Systems

Significance: G Sep 30, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
Failure to Justify Continued Service of Safety-Related Electrolytic Capacitors Installed Beyond Their Service Live
An NRC identified finding of very low safety significance and an associated NCV of Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix B, Criterion III, "Design Control," was identified for the failure to justify continued service of safety related electrolytic capacitors that were installed beyond their recommended service life associated with the safety related containment floor level indicating transmitters (LITs). Specifically, on June 21, 2015, containment floor LIT LIT–0446B and LIT–0446A did not satisfy the acceptance criteria of the technical specification surveillance monthly channel checks and LIT–0446B was declared inoperable. Further troubleshooting identified a failure of the electrolytic capacitor within the transmitter's converter module and that this failure was most likely due to age since the transmitter had been in service for greater than its recommended service life. In addition to entering this issue into their Corrective Action Program (CAP) as CR–PLP–2015–04972, the licensee replaced the failed components and planned to develop a replacement schedule for non critical, safety related electrolytic capacitors.

The performance deficiency was determined to be more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The finding screened as having very low safety significance based on answering "No" to all of the screening questions in the Mitigating Structures, Systems, and Components (SSCs) and Functionality section of IMC 0609, Appendix A, "The Significance Determination Process for Findings At Power," Exhibit 1, "Mitigating Systems Screening Questions." The finding had a cross cutting aspect of Operating Experience in the Problem Identification and Resolution cross cutting area because the licensee did not effectively and thoroughly evaluate and implement relevant industry operating experience and guidance for age related electrolytic capacitor degradation
Inspection Report# : 2015003 *(pdf)*

Significance: G Jun 30, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
Failure to Take Appropriate Corrective Action for the Charging System While in Maintenance Rule (a)(1) Status
An NRC identified finding of very low safety significance and an associated NCV of Title 10 of the Code of Federal Regulations (CFR) 50.65(a)(1) was identified for the failure to take appropriate corrective actions for the charging system, while in Maintenance Rule (a)(1) status, when performance or condition goals were not met. Specifically, on April 2, 2015, the front cap of the 'B' charging pump cracked, causing volume control tank (VCT) level and pressure to lower, most likely due to excessive local cavity pressures in the pump caused by the suction accumulator pressure being out of specification. Accumulator pressures being out of specification, which causes pressure oscillations and vibrations in the charging pumps and their associated suction and discharge piping, was a similar cause to previous maintenance rule system functional failures that occurred in 2013 and 2014, which transitioned the system to (a)(1) status in July 2014. The licensee documented the issue in their corrective action program (CAP), conducted an equipment apparent cause evaluation (EACE) for the most recent failure, and revised the Maintenance Rule (a)(1) Action Plan to address the on going issues with the suction and discharge accumulators.

The inspectors determined that the performance deficiency was more than minor in accordance with IMC 0612 because it was associated with the Equipment Performance attribute of the Mitigating Systems Cornerstone and adversely impacted the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The charging system provides the critical safety functions of pressure and inventory control in the emergency operating procedures. The finding screened as having very low safety significance (i.e., Green) based on answering "No" to all the screening questions under the Mitigating Structures, Systems, and Components (SSCs) and Functionality section of the significance determination process (SDP). The finding had a cross-cutting aspect of Evaluation in the Problem Identification and Resolution area. Specifically, the organization did not thoroughly evaluate previous data on the suction and discharge accumulators pressures being out of specification and what affect that may have on the system. Also, when the accumulator pressures were found out of specification, sometimes that information was not documented in condition reports (CRs), nor were the preventive maintenance (PM) frequencies re evaluated in a technical and rigorous manner to ensure the correct PM activities were being conducted on these components in a timely manner to assure system reliability

Inspection Report# : 2015002 *(pdf)*

Significance: G Apr 17, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
Failure to Correctly Assess the Suppression System in the Cable Spreading Room in the Probabilistic Risk Assessment for NFPA 805 (Section 1R05.3b)
Green. The inspectors identified a finding of very-low safety significance, and an associated NCV of Title 10, Code of Federal Regulations (CFR) 50.48(c), and National Fire Protection Association Standard 805, Section 2.4.3.3 for the licensee's failure to correctly model the as-built plant in the Fire Probabilistic Risk Assessment (PRA). Specifically, the licensee credited the suppression system located in the cable spreading room in the PRA to suppress type 2 fire scenarios, whereas the actual room contained numerous obstructions due to the stacked cable trays located near the ceiling that interfered with the water spray pattern discharged from the sprinklers. These obstructions could have prevented the suppression system from providing an adequate water density pattern to suppress a fire below the cable trays in areas which contained electrical panels.

The inspectors determined that the performance deficiency was more than minor because the finding, if left uncorrected, would have the potential to lead to a more significant safety concern. Specifically, the licensee's failure to correctly model/analyze the as-built condition of the suppression system located in the cable spreading room in the PRA could potentially affect the risk associated with a fire in the room, and could result in inappropriately screening out the effects of other changes associated with the fire area. Appendix M was used because the existing SDP Appendices do not adequately address the risk of performance deficiencies associated with licensees' PRAs. The Senior Reactor Analyst concluded that the finding was of very-low safety significance (Green) because while there may be a change to the plant's baseline risk as a result of this issue, there is no delta plant risk due to a deficiency in the licensee's PRA model/analysis. This finding has a cross-cutting aspect in the area of Human Performance associated with Team Work because the licensee did not communicate and coordinate activities between the PRA and the fire protection groups. [H.4]

Inspection Report# : 2015008 *(pdf)*

Significance: G Mar 31, 2015
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
Inadequate Procedure Results in Failure of Component Cooling Water Pump
A finding of very low safety significance and an associated NCV of TS 5.4.1(a) was self-revealed on January 6, 2015, after the licensee identified smoke coming from the 'C' component cooling water (CCW) pump (P–52C) as a result of incorrect assembly of the inboard pump bearing in December 2014, due to an inadequate maintenance procedure. This issue was entered into the licensee's CAP as CR–PLP–2015–00063, Workers Reported Smoke Coming from Shaft of P–52C, dated January 6, 2015.

The performance deficiency was determined to be more than minor because it was associated with the Procedure Quality attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Based on a detailed risk evaluation performed by a Region III Senior Reactor Analyst using SAPHIRE Version 8.20 and the Events and Conditions Assessment Feature of the Standiardized Plant Analysis Risk model (Version 8.1.2), the inspectors determined the finding was of very low safety significance. This finding had a cross-cutting aspect in the Avoid Complacency component of the Human Performance cross-cutting area. Specifically, plant staff accepted the practice of bending the 'C' CCW pump oiler nipple to achieve proper level when the oiler could not be properly aligned which compensated for, rather than corrected, an underlying issue of improper alignment when tightening the alignment pin.
Inspection Report# : 2015001 *(pdf)*

Significance: G Mar 31, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
Inoperability of Safety Injection Tank Due to Long-Term Leakage
A finding of very low safety significance and an associated NCV of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," was identified by the inspectors when licensee personnel failed to assure that leakage out of the 'B' safety injection tank (SIT), a condition adverse to quality, was corrected in a timely manner. Specifically, although minor water leakage out of the 'B' SIT had been occurring since at least 2010, the licensee had not corrected the leakage despite several plant outages that provided an opportunity to address the issue. This issue was entered into the licensee's CAP as CR–PLP–2014–04861, B SIT Declared Inoperable Due to Reaching Technical Specification Low Level Setpoint, dated October 7, 2014.

The performance deficiency was determined to be more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the leakage out of the ‘B’ SIT resulted in unexpected inoperability of the tank on October 7, 2014. The finding was determined to be of very low safety significance based on answering "No" to the screening questions in Exhibit 2.A, Mitigating Systems Screening Questions. This finding had a cross-cutting aspect in the Avoid Complacency component of the Human Performance cross-cutting area. Specifically, over time the licensee became confident that the long-term leakage out of the ‘B’ SIT was minor and could be managed without an impact to equipment operability, which proved to be incorrect when the minor leakage resulted in ‘B’ SIT inoperability on October 7, 2014.

Inspection Report# : 2015001 *(pdf)*

Significance: G Mar 31, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
Failure to Verify the Adequacy of Credited High Energy Line Break Barriers
A finding of very low safety significance and an associated NCV of 10 CFR 50, Appendix B, Criterion III, “Design Control,” was identified by the inspectors when the licensee credited fire doors for High Energy Line Break (HELB) protection without a supporting test or evaluation. Specifically, Procedure 4.02 credited fire doors with protection of safety-related equipment against a HELB when the primary HELB
barrier was disabled without a test or evaluation to demonstrate the doors could withstand the HELB environment. This issue was entered into the licensee’s Corrective Action Program as CR–PLP–2015–00371, NRC Concerns with Calculation EA–PSA–CCW–HELB–02–17, dated January 22, 2015.

The performance deficiency was determined to be more than minor because it was associated with the Design Control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee did not have an evaluation to demonstrate that barriers relied upon to protect mitigating systems from a HELB initiating event could perform the credited protection function. The inspectors answered "No" to the questions in Exhibit 2.A, Mitigating Systems Screening Questions, and as a result determined the issue was of very low safety significance. This finding was not associated with a cross-cutting aspect since the calculation in question was created in 2003 and therefore did not represent current performance.

Inspection Report# : 2015001 *(pdf)*

Significance: G Mar 31, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
Failure to Evaluate the Adverse Effects of the Use of Non-Seismic Temporary Jumpers
A Severity Level IV NCV of 10 CFR 50.59(d)(1), “Changes, Tests, and Experiments,” and an associated finding of very low safety significance was identified by the inspectors when licensee personnel failed to maintain a written safety evaluation that provided a basis that the use of temporary alligator clip jumpers to maintain emergency diesel generator (EDG) operability during certain maintenance activities did not require a license amendment. Specifically, the licensee did not address the adverse effects of the use of alligator jumpers on the design and qualification of the diesel generator (DG) circuit breaker used per Engineering Change 50310 and changes to procedure SPS–E–1, “2400 Volt and 4160 Volt Allis Chalmers and Siemens Vacuum Circuit Breaker Auxiliary Switch Adjustments,” Revision 34. This issue was entered into the licensee’s CAP as CR–PLP–2014–04859, NRC Identified 50.59 Issue, dated October 7, 2014.

The inspectors evaluated the underlying technical issue and determined the finding was of very low safety significance. In accordance with Section 6.1.d.2 of the NRC Enforcement Policy, this violation was categorized as

Severity Level IV because the finding associated with this violation was determined to be of very low safety significance.

Inspection Report# : 2015001 *(pdf)*

Significance: G Mar 31, 2015
Identified By: NRC
Item Type: FIN Finding
Failure to Evaluate the Adverse Effects of the Use of Non-Seismic Temporary Jumpers
A Severity Level IV NCV of 10 CFR 50.59(d)(1), “Changes, Tests, and Experiments,” and an associated finding of very low safety significance was identified by the inspectors when licensee personnel failed to maintain a written safety evaluation that provided a basis that the use of temporary alligator clip jumpers to maintain emergency diesel generator (EDG) operability during certain maintenance activities did not require a license amendment. Specifically, the licensee did not address the adverse effects of the use of alligator jumpers on the design and qualification of the diesel generator (DG) circuit breaker used per Engineering Change 50310 and changes to procedure SPS–E–1, “2400 Volt and 4160 Volt Allis Chalmers and Siemens Vacuum Circuit Breaker Auxiliary Switch Adjustments,” Revision 34. This issue was entered into the licensee’s Corrective Action Program as CR–PLP–2014–04859, NRC Identified 50.59 Issue, dated October 7, 2014.

The performance deficiency was determined to be more than minor because it was associated with the Design Control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the change that was implemented adversely affected the seismic qualification of the electrical circuit that was relied upon to ensure safety bus 1C would be loaded by the 1–1 DG upon a loss of offsite power. The inspectors evaluated the underlying technical issue and determined the finding was of very low safety significance. In accordance with Section 6.1.d.2 of the NRC Enforcement Policy, this violation was categorized as Severity Level IV because the finding associated with
this violation was determined to be of very low safety significance. This finding had a cross-cutting aspect in the Conservative Bias component of the Human Performance cross-cutting area. Specifically, the licensee did not use all available information and relevant guidance, such as Nuclear Energy Institute 96–07, to demonstrate that the proposed activity was safe and did not require a license amendment prior to implementation.

Inspection Report# : 2015001 *(pdf)*

Significance: G Dec 31, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
Failure to Follow Procedure for Storage of Equipment in the Vicinity of Safety-Related Equipment
The inspectors identified a finding of very low safety significance (Green) with an associated non-citied violation of Technical Specification (TS) 5.4.1, Procedures and Programs, for the failure to follow site procedures covering the storage of material in the vicinity of safety-related equipment. Specifically, on three occasions the inspectors identified ladders at ladder station 42 in the 590’ elevation of the component cooling water room that were either in contact with safety-related equipment or were capable of toppling into safety-related equipment. For immediate corrective actions, licensee personnel properly stored the ladder after each issue was identified by the inspectors. This issue is documented in the licensee’s corrective action program (CAP) as Condition Report CR-PLP-2015-00126.

The performance deficiency was determined to be more than minor based on Inspection Manual Chapter (IMC) 0612, Appendix E, Example 4.a, which determined that low-level procedural errors without a safety consequence are more than minor when they become a repetitive/routine occurrence. Specifically, unrestrained ladders could impact safety-

related equipment during a design basis seismic event. The inspectors evaluated the significance of the finding in accordance with IMC 0609, Attachment 4, "Initial Characterization of Findings." In accordance with Table 2, the finding was determined
to affect the Mitigating Systems Cornerstone. The inspectors answered 'No' to the questions in Table 3 and continued the significance evaluation in accordance with IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power." The inspectors answered 'No' to the Mitigating Systems Screening Questions contained in Exhibit 2 and determined the finding was of very low safety significance (Green). This finding was associated with a cross-cutting aspect of Identification in the Problem Identification and Resolution cross-cutting area (P1).

Inspection Report# : 2014005 *(pdf)*

Significance: G Nov 04, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
Failure to Ensure Engineered Safeguards Systems Aren't Adversely Affected By Air Entrainment
The inspectors identified a finding of very low safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control" for the failure to ensure the safety-related Engineered Safeguard Systems trains would not be adversely affected by air entrainment when aligned to the Safety Injection and Refueling Water (SIRW) Tank. Specifically, calculation EA-C-PAL-0877D, assumed incorrectly only one train of the Engineered Safeguards System (ESS) was in operation when evaluating if the SIRW Tank reaches the limit for critical submergence during a tank drawdown. As part of their corrective actions, the licensee re-evaluated the scenarios of concern, performed an operability evaluation, and implemented compensatory actions.

The performance deficiency was determined to be more than minor because it impacted the Equipment Performance attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, air entrainment into the ESS systems could potentially impact the operability of the system by air binding the pumps, reduce discharge flow, discharge pressure and/or delay injection. The inspectors determined the finding was of very low safety significance (Green) because the finding was a deficiency affecting the design or qualification of a mitigating structure system or component (SSC) but the SSC maintained its operability. The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of the licensee's present performance.
Inspection Report# : 2014008 *(pdf)*

Significance: G Nov 04, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
Undersized Supply Cables from Startup Transformer to 2400V Buses
The inspectors identified a finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to ensure the incoming feeder cables from startup transformer 1-2 to 2400 V safety related Buses 1C and 1D were sized in accordance with their design basis, as described in Palisades FSAR Section 8.5.2. Specifically, the licensee failed to ensure the ampacity of the cables was at least as high as their maximum steady-state current. The licensee entered this finding into their Correction Action Program and verified the operability of the cables.

The performance deficiency was determined to be more than minor, because it impacted the Design Control attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, cables were undersized with respect to the loading that would automatically occur as the

result of a design basis accident. The inspectors determined the finding was of very low safety significance (Green) because the SSC maintained its operability and functionality. This finding had a crosscutting aspect in the area of Human Performance, associated with the Design Margin component, because the licensee did not ensure that equipment is operated and maintained within design margins, and margins are carefully guarded and changed only through a systematic and rigorous process.
Inspection Report# : 2014008 *(pdf)*

Significance: G Nov 04, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
Undersized Motors
The inspectors identified a finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the licensee’s failure to ensure electric motors are sized in accordance with the design basis, as discussed in Palisades FSAR Section 6.2.3.1. Specifically, the horsepower ratings of certain motors are less than power demands of their driven equipment, and they were not analyzed to ensure overheating would not occur. The licensee entered this finding into their Correction Action Program with a recommended action to analyze the effect of the condition, and has verified the operability of the motors.

This performance deficiency was determined to be more than minor, because it impacted the Design Control attribute of the Reactor Safety, Mitigating Systems Cornerstone and
adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, motors serving loads with power demands in excess of the motor horsepower ratings were not analyzed to ensure that motor damage would not occur. The inspectors determined the finding was of very low safety significance (Green) because the SSC maintained its operability and functionality. This finding had a crosscutting aspect in the area of Human Performance, associated with the Design Margin component, because the licensee failed to ensure that equipment is operated within design margins, and margins are carefully guarded and changed only through a systematic and rigorous process.
Inspection Report# : 2014008 *(pdf)*

Significance: G Nov 04, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
Failure to Ensure that 480V System Voltages do not Exceed Equipment Ratings
The inspectors identified a finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the licensee’s failure to ensure that voltages on the 480V system do not exceed equipment ratings. Specifically, the licensee increased the output voltage of the supply transformers to the 480V safety-related buses by 2.5 percent, but failed to ensure the resulting voltages would not exceed equipment ratings when the system is powered from the station power transformer or emergency diesel generator. The licensee entered this finding into their Correction Action Program and verified the operability of the affected equipment.

The performance deficiency was determined to be more than minor, because it impacted the Design Control attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to verify or check the voltage increase on the 480V system to ensure the maximum allowable voltage would not exceed equipment ratings. The inspectors determined the finding was of very low safety significance (Green) because the affected SSCs maintained their operability and functionality. The inspectors did not identify a cross-cutting aspect associated with this finding, because the finding was not representative of the licensee’s present performance.

Inspection Report# : 2014008 *(pdf)*

Significance: G Nov 04, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
Failure to Perform Comprehensive Pump Testing of Containment Spray Pump P-54A in Accordance with the Inservice Testing Program
The inspectors identified a finding of very low safety significance and associated Non-Cited Violation of Technical Specifications 5.5.7, "Inservice Testing Program," for the failure to perform comprehensive pump testing of Containment Spray Pump P-54A in accordance with the code of record. Specifically, the licensee did not rerun a comprehensive pump test, as required by the code's ISTB-6300 "Systematic Error" section. As part of their corrective actions, the licensee entered the issue into the Corrective Action Program, and determined the component remained operable.

The performance deficiency was determined to be more than minor because it impacted the Equipment Performance attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, failing to perform testing as required could result in the degradation of the equipment being undetected. The finding screened as having very low safety significance because the finding was a deficiency affecting the design or qualification of a mitigating structure system or component (SSC) but the SSC maintained its operability. The findings had a cross-cutting aspect in the area of Problem Identification and Resolution, Evaluation, because the licensee failed to thoroughly evaluate the issue to ensure that resolutions address causes and extents of conditions commensurate with their safety significance.

Inspection Report# : 2014008 *(pdf)*

Significance: G Nov 04, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
Non-Conservative Surveillance for Emergency Diesel Generator Largest Load Reject Test
The inspectors identified a finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," for the licensee's failure to have adequate acceptance criteria in the emergency diesel generator surveillance procedures. Specifically, the licensee failed to ensure the surveillance test procedures for the emergency diesel generator largest load rejection test bounded the power demand of the largest load, as required by Technical Specification SR 3.8.1.5. The licensee entered this finding into their Correction Action Program and verified the operability of the emergency diesel generators.

The performance deficiency was determined to be more than minor, because it impacted the Procedure Quality attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems to respond to initiating events to prevent undesirable consequences. Specifically, the surveillance procedure error could result in acceptance of test results that did not satisfy Technical Specification SR 3.8.1.5 for rejection of a load greater than or equal to the emergency diesel generator's single largest predicted post-accident load. The inspectors determined the finding was of very low safety significance (Green) because the SSC maintained its operability and functionality. This finding had a cross-cutting aspect in the area of Human Performance, associated with the Resources component, because the licensee failed to ensure that personnel, equipment, procedures, and other resources are adequate to assure nuclear safety by maintaining long term plant safety.
Inspection Report# : 2014008 *(pdf)*

Barrier Integrity

Significance: G Nov 04, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
Failure to Correctly Translate Valve Leakage Limits into Test Procedure
The inspectors identified a finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the licensee’s failure to correctly translate design valve leakage limits into the applicable test procedure. Specifically, the acceptance criterion for emergency core cooling system (ECCS)/containment spray (CS) recirculation isolation valves CV-3027 and CV-3056 had not been correctly adjusted to account for the higher differential pressure associated with ECCS operation under post-accident conditions. The licensee entered this finding into their Corrective Action Program to correct the valve leakage limit.

The performance deficiency was determined to be more than minor because it impacted the Design Control attribute of the Barrier Integrity Cornerstone and adversely affected the associated cornerstone objective to provide reasonable assurance that containment could protect the public from radionuclide releases caused by accidents or events. Specifically, leakage approaching the procedural values would exceed analyzed dose calculations. The finding screened as of very low safety significance (Green) because the finding did not represent an actual open pathway in the physical integrity of reactor containment, containment isolation system, or heat removal components and did not involve an actual reduction in function of hydrogen igniters in the reactor containment. The inspectors determined this finding did not have an associated cross-cutting aspect because it was not representative of present performance.

Inspection Report# : 2014008 *(pdf)*

Significance: G Nov 04, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
Failure to Identify Non-Safety-Related Sub-Components Improperly Supplied with Safety-Related Valves
The inspectors identified a finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion VII, “Control of Purchased Material, Equipment, and Services,” for the licensee’s failure to identify non-safety-related
sub-components improperly supplied with safety-related valves. Specifically, ECCS/CS recirculation isolation valves CV-3027 and CV-3056, which were installed in 2007, were supplied with non-safety-related sub-components. These components were identified as non-safety-related on the vendor drawings. In addition, the licensee later installed a section of non-safety-related tubing on valve CV-3027 based on the incorrect vendor drawing. The licensee entered this finding into their Corrective Action Program to correct the valve drawings and replace the non-safety-related parts.

The performance deficiency was determined to be more than minor because it impacted the Design Control attribute of the Barrier Integrity Cornerstone and adversely affected the associated cornerstone objective to provide reasonable assurance that containment could protect the public from radionuclide releases caused by accidents or events. Specifically, the licensee failed to identify non-safety-related sub-components improperly supplied with safety-related valves which would form part of the containment barrier under post-accident conditions. The finding screened as of very low safety significance (Green) because the finding did not represent an actual open pathway in the physical integrity of reactor containment, containment isolation system, or heat removal components and did not involve an actual reduction in function of hydrogen igniters in the reactor containment. The inspectors determined this finding did not have an associated cross-cutting aspect because it was not representative of the licensee’s present performance.

Inspection Report# : 2014008 *(pdf)*

Significance: G Nov 04, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
Failure to Establish an Adequate Test Program for the Shutdown Cooling Heat Exchangers
The inspectors identified a finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," for the licensee's failure to establish an adequate test program for the Shutdown Cooling (SDC) Heat Exchangers (HXs) to demonstrate they can perform as designed. Specifically, the licensee failed to take actions to ensure the SDC HXs' heat transfer capability met its design bases, as assumed in design bases calculations.

The performance deficiency was determined to be more than minor because it impacted the Design Control attribute of the Barrier Integrity Cornerstone and adversely affected the associated cornerstone objective to provide reasonable assurance that containment could protect the public from radionuclide releases caused by accidents or events. Specifically, the licensee failed to verify the SDC HXs heat transfer capability met their design bases, as assumed in design bases calculations, to limit containment temperatures and pressures during an event. The finding screened as of very low safety significance (Green) because the finding did not represent an actual open pathway in the physical integrity of reactor containment, containment isolation system, or heat removal components and did not involve an actual reduction in function of hydrogen igniters in the reactor containment. The inspectors determined this finding had an associated cross-cutting aspect, Conservative Bias, in the Human Performance cross-cutting area. Specifically, on several occasions when the licensee identified the need to perform testing and/or inspection of the SDC HXs, the licensee did not take actions because they did not believe any regulatory requirements or technical issues existed that required the testing and/or inspections.
Inspection Report# : 2014008 *(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Significance: G Sep 30, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
Failure To Establish, Implement, and Maintain the Offsite Dose Calculation Manual
A finding of very low safety significance and an associated NCV of Technical Specification (TS) 5.5.1, "Offsite Dose Calculation Manual," was identified for the failure to establish, implement, and maintain the Offsite Dose Calculation Manual (ODCM) relative to dose calculation parameters. Specifically, the licensee failed to modify the parameters used in public radiation calculations when changes in the use of unrestricted areas were identified. As a result, the quarterly and annual doses that were calculated every 31 days, as required by the ODCM, were incorrect and non conservative. In addition to entering this issue into their Corrective Action program (CAP) as CR–PLP–2015–2972, the licensee recalculated the dose using the correct calculation parameters.

The performance deficiency was determined to be more than minor because it was associated with the Program and Process attribute of the Public Radiation Safety cornerstone and adversely affected the cornerstone objective of ensuring the adequate protection of public health and safety from exposure to radioactive materials released into the public domain as a result of routine civilian nuclear reactor operation. The finding was determined to be of very low safety significance in accordance with IMC 0609, Appendix D, "Public Radiation Safety Significance Determination Process," because the issue did not represent a significant deficiency in evaluating a planned or unplanned effluent release since the resulting dose was not grossly underestimated. The finding had a cross cutting aspect of Training in the Human Performance cross cutting area because the licensee did not ensure adequate knowledge transfer to maintain a knowledgeable, technically competent workforce.
Inspection Report# : 2015003 *(pdf)*

Significance: G Jun 30, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
Failure to Wear Prescribed Respiratory Protection
A self revealed finding of very low safety significance and an associated NCV of Technical Specification (TS) 5.4.1 was identified for insulation work activities during the refueling outage associated with pressurizer spray valve CV–1057. Specifically, prior to the work beginning, the licensee determined that the use of powered air purifying respirators would be required to minimize worker dose and maintain exposures as low as reasonably achievable (ALARA), but the work was performed using only face shields, and as a result a worker was contaminated externally and internally. Corrective actions included creation of an administrative requirement to revise any radiation work permit (RWP) task that required respiratory protection to more clearly state the requirements.

The inspectors determined that the performance deficiency was more than minor in accordance with IMC 0612 because it was associated with the Program and Process attribute of the Occupational Radiation Safety Cornerstone and adversely affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation. Specifically, the failure to wear required respiratory protection during the reinsulating of CV–1057 resulted in personal contamination and the intake of radioactive material. The inspectors concluded that the radiological hazards had the potential to result in higher exposures to the individuals had the circumstances been slightly altered. The finding was determined to be of very low safety significance (Green) in accordance with IMC 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," because it was not an ALARA planning issue, there was neither an overexposure nor a substantial potential for an overexposure, and the licensee's ability to assess dose was not compromised. The inspectors concluded that the cause of the issue involved a cross cutting aspect in the area of Human Performance, Basis for Decisions. Specifically, the bases for operational decisions were communicated in a timely manner.

Inspection Report# : 2015002 *(pdf)*

Significance: W Oct 30, 2014
Identified By: NRC
Item Type: VIO Violation
Failure to Monitor the Highest Exposed Part of the Compartment When Using EDEX
The NRC identified one finding and two violations of NRC requirements associated with the replacement of Control Rod Drive (CRD) housings between February 6 and March 8, 2014. Specifically, the inspectors identified a violation of Title 10 of the Code of Federal Regulations (CFR) Part 20.1201, "Occupational Dose Limits for Adults," because the licensee failed to ensure that radiation worker dosimeters calibrated to the Deep Dose Equivalent (DDE) were located at the highest exposed portion of the respective compartment, a condition of the NRC-approved method for determining effective dose equivalent external (EDEX). The inspectors also identified a violation of Technical Specification 5.4 "Procedures," associated with this finding. Upon identification of this issue, the licensee suspended

the use of EDEX and tungsten shield vests. The licensee re-calculated the dose received for the workers involved and updated the nuclear power industry's dose tracking system with the revised dose results. Additionally, a root cause evaluation was initiated under Condition Report CR-PLP-2014-04683.
The inspectors reviewed the guidance in IMC 0612, Appendix E, "Examples of Minor Issues," and did not find any similar examples. The performance deficiency was determined to be of more than minor safety significance in accordance with IMC 0612 Appendix B, "Issue Screening," because it was associated with the program and process attribute of the Occupational Radiation Safety Cornerstone and adversely affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation, in that inaccurate radiation monitoring affects the licensee's ability to control and limit radiation exposures. Therefore, the performance deficiency was a finding. The finding did not involve as-low-as-reasonably-achievable (ALARA) planning or work controls and there was no overexposure or substantial potential for an overexposure. However, the NRC determined that the licensee's ability to assess dose was compromised. Consequently, the NRC concluded that the finding was of White safety significance. The finding had a cross-cutting characteristic in the area of human performance related to the cross-cutting aspect of change management, in that, the licensee's procedures did not include all of the requirements for implementing EDEX when the methods were approved by the NRC and did not provide adequate guidance for the new tungsten shield vests.
Inspection Report# : 2014010 *(pdf)*
Inspection Report# : 2015007 *(pdf)*

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

Significance: N/A Aug 19, 2015
Identified By: NRC
Item Type: VIO Violation
Innaccurate/Incomplete Information Submitted for Relieft Request 4-18 (Section 1R15)
• TBD. An apparent violation (AV) of Title 10 of the Code of Federal Regulations (CFR) 50.9 was identified by the licensee, related to a failure to provide information that was complete and accurate in all material respects to the NRC in letter PNP 2014-015, "Relief Request (RR) Number 4-18 - Proposed Alternative Use of Alternate ASME [American Society of Mechanical Engineers] Code Case N-770-1 Baseline Examination." Specifically, in this document the licensee stated, "In the unlikely case that crack initiation were to occur, crack growth calculations considering primary water stress corrosion cracking (PWSCC) as the failure mechanism demonstrate that the hot leg drain nozzle weldment satisfies ASME Code acceptance criteria for 60 effective full power years [EFPY] for a

circumferential flaw, and more than 34 years for an axial flaw." However, this statement was not correct or accurate in that, the ASME Code acceptance criteria were not satisfied for 60 EFPY for a circumferential flaw and 34 years for an axial flaw, where correct information was 20 EFPY for a circumferential flaw, and 11.3 years for an axial flaw. This AV was not an immediate safety concern because the licensee demonstrated an adequate basis for continued operability of the nine affected primary coolant system (PCS) welds. The licensee corrective actions for this AV included completion of an operability evaluation, submittal of a corrected analysis to the NRC, and entering this issue into the Corrective Action Program (CAP) (CR-PLP-2015-03441).
If the NRC was provided with the correct information in letter PNP 2014-015, where the affected welds satisfied ASME Code acceptance criteria (i.e., 75 percent through-wall) for only 20 effective full power years for a circumferential flaw, and 11.3 years for an axial flaw, the NRC would not likely have approved RR 4-18 and, as a minimum, would have requested additional supporting analysis (e.g., required substantial further inquiry). Further, the need for substantial further inquiry was illustrated by the licensee's subsequent decision in RR 4 21 to abandon the prior analytical approach used in RR 4 18. The inspectors evaluated the underlying technical issue in accordance with the SDP to determine the risk significance of this AV. The issue of concern was of more than minor significance because it was similar to the "not minor if" aspect of Example 3j in IMC 0612, Appendix E, "Example of Minor Issues." Specifically, the erroneous information provided in letter PNP 2014-015 resulted in a condition in which there was a reasonable doubt on the operability of the systems and components that were the subject of the evaluation and dissimilar from the "minor because" aspect of this example since the impact of the error for the operability of nine PCS welds was not minimal. In addition, the performance deficiency was determined to be more than minor because it was associated with the Initiating Event Cornerstone attribute of Equipment Performance and adversely affected the Cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions. The inspectors evaluated the finding in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," Table 3, for the Initiating Events Cornerstone, and IMC 0609, Appendix A, "The SDP for Findings At-Power." Because the licensee was able to demonstrate operability of the nine PCS welds susceptible to PWSCC, the inspectors answered "No" to questions A.1 and A.2, of Exhibit 1, "Initiating Events Screening Questions," identified in Appendix A of IMC 609 and, as a result, the finding screened as having very low safety significance (Green). No cross-cutting aspect was assigned because this Green finding was identified by the licensee. (Section 1R15)

- A final significance determination letter, SL III, Notice of Violation for EA-15-171 was issued on November 24, 2015. ADAMS Accession Number ML15328A534.
The failure to provide complete and accurate information is of significant safety concern to the NRC because the inaccurate information impacted the NRC's ability to perform its regulatory function. The NRC relied on the inaccurate information to make a licensing decision approving Relief Request 4-18. If the information had been correct the NRC would have undertaken substantial further inquiry and reconsidered its regulatory position. Therefore, this violation has been categorized in accordance with the NRC Enforcement Policy at Severity Level III.
Inspection Report# : 2015012 *(pdf)*

Last modified : December 15, 2015