Monticello
2Q/2011 Plant Inspection Findings

Initiating Events

Significance: Jun 30, 2011
Identified By: Self-Revealing
Item Type: FIN Finding
POOR MAINTENANCE PRACTICES RESULT IN CV-3490 FAILING SHUT.
A finding of very low safety significance was self revealed when, on two separate occasions, CV 3490 (12 reactor feedwater pump recirculation to the condenser) failed closed while the 12 reactor feedwater pump was being placed in service. The cause of each failure was directly related to poor maintenance practices while performing work on CV 3490's valve positioner. Additionally, each failure resulted in an automatic trip of the 12 reactor feedwater pump. The licensee entered this issue into their corrective action program, corrected the mechanical issues, and performed an extent of condition review. The inspectors determined that the performance deficiency affected the cross cutting area of Human Performance, having work practice components, and involving aspects associated with ensuring that supervisory and management oversight of work activities, including contractors, such that nuclear safety is supported. [H.4(c)]

The finding was more than minor because it impacted the configuration control attribute of the Initiating Events Cornerstone objective of limiting those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors applied IMC 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," to this finding. The inspectors utilized Column 1 of the Table 4a worksheet to screen the finding. The inspectors answered 'No' to the question "does the finding contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available" and, therefore, the finding was screened to be of very low safety significance.
Inspection Report# : 2011003 *(pdf)*

Significance: Jun 30, 2011
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
INADEQUATE C.1 STARTUP PROCEDURE REVIEW.
A finding of very low safety significance and non cited violation (NCV) of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was self revealed when an unexpected recirculation pump runback occurred during the performance of Reactor Dynamics Testing. The event was the result of the licensee failing to adequately assess the operational impact of a recent revision to Procedure C.1, "Startup Procedure," which resulted in operating the plant in a manner that challenged feedwater pump protective features. The licensee entered this issue into their corrective action program (CAP 01288070) and initiated corrective actions to address the issue. The inspectors determined that the contributing cause that provided the most insight into the performance deficiency was associated with the cross cutting area of Human Performance, having decision making components, and involving aspects associated with the licensee conducting effectiveness reviews of safety significant decisions to verify the validity of the underlying assumptions, identify possible unintended consequences, and determine how to improve future decisions. [H.1(b)]

The finding was more than minor because it impacted the procedure quality attribute of the Initiating Events Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors applied IMC 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," to this finding. The inspectors utilized Column 1 of the Table 4a worksheet to screen the finding. The inspectors answered 'No' to the questions associated with loss-of-coolant accident (LOCA) initiators, transient initiators, and external events initiator, and screened the finding to be of very low safety significance.
Inspection Report# : 2011003 *(pdf)*

Significance: G Mar 31, 2011
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
INADEQUATE SYSTEM ISOLATION DURING CHECK VALVE MAINTENANCE.
A finding of very low safety significance and associated NCV of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was self-revealed when the licensee failed to adequately implement the requirements of their fleet tagging procedure, a procedure affecting quality, during maintenance on the safety-related CST-88 ‘B’ low pressure coolant injection (LPCI) fill line check valve. This failure resulted in an unintentional breach of the condensate service water (CSW) system and subjected workers to a potentially contaminated, pressurized water source. Additionally, at the time of the breach, the CSW system was one of the water sources being credited in support of the shutdown safety function of inventory control. The licensee entered this issue into the corrective ation program (CAPs 1275935 and 1275963) and took immediate corrective actions to restore the check valve to its installed configuration to terminate the water leakage. At the time of this report, the licensee had assembled a team to perform a root cause evaluation.
The inspectors determined that the licensee’s failure to adequately implement their tagging process to protect workers and equipment from the effects of breaching the pressurized CSW header during maintenance on a safety-related check valve was a performance deficiency because it was the result of the failure to meet a requirement, the cause was reasonably within the licensee’s ability to foresee and correct, and should have been prevented. The inspectors screened the performance deficiency per IMC 0612, “Power Reactor Inspection Reports,” Appendix B, and determined that the issue was more than minor because the performance deficiency could have reasonably been viewed as a precursor to a more significant event. In this instance, the performance deficiency resulted in an unintentional breach of the operating CSW system and subjected workers to a potentially contaminated, pressurized water source. Additionally, at the time of the breach, the CSW system was one of the water sources being credited in support of the shutdown safety function of inventory control. As a result, this finding was evaluated under the Initiating Events Cornerstone.
The inspectors applied NRC IMC 0609, “Significance Determination Process,” Appendix G, "Shutdown Operations Significance Determination," Attachment 1, to this finding. The finding was determined to have very low safety significance because it did not adversely affect core heat removal, inventory control, power availability, containment control, or reactivity guidelines. This finding has a cross-cutting aspect in the area of Human Performance, work control, because the licensee failed to appropriately plan work activities by incorporating job site conditions impacting plant systems and components (H.3(a)).
Inspection Report# : 2011002 *(pdf)*

Mitigating Systems

Significance: G Dec 31, 2010
Identified By: NRC
Item Type: FIN Finding
FAILURE TO PROPERLY STORE LOOSE MATERIAL IN CLOSE PROXIMITY TO SAFETY RELATED EQUIPMENT.
A finding of very low safety significance was identified by the inspectors when the licensee failed to properly control loose material located above the sensing lines for the safety related residual heat removal pump minimum flow switches. No violation of NRC requirements associated with this finding was identified. Once informed of the issue, the licensee took action to relocate the material to a proper storage location. The inspectors determined that the contributing cause that provided the most insight into the performance deficiency was associated with the cross

cutting area of Human Performance, having Work Practices components, and involving aspects associated with the licensee defining and effectively communicating expectations regarding procedural compliance and personnel following procedures. [H.4(b)].
The inspectors determined that the licensee’s failure to properly store loose material located in close proximity to safety related equipment was a performance deficiency, because it was the result of the failure to meet a requirement; the cause was reasonably within the licensee’s ability to foresee and correct; and should have been prevented. The inspectors screened the performance deficiency per IMC 0612, Power Reactor Inspection Reports, Appendix B, and determined that the issue was more than minor because it impacted the protection against external events attribute of the Mitigating System Cornerstone's objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors applied IMC 0609, Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings,” to this finding. The inspectors utilized Column 2 of the Table 4a worksheet to screen the finding. As a result of the inspectors answering “No” to all five questions, the finding was screened to be of very low safety significance.

Inspection Report# : 2010005 *(pdf)*

Barrier Integrity

Significance: SL-IV Jun 30, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO UPDATE USAR FOR CASK LIFT HEIGHT RESTRICTIONS.
“Periodic Update of the Final Safety Analysis Report” and an accompanying Green finding were identified by the inspectors for the licensee’s failure to update the Updated Safety Analysis Report (USAR) with the cask maximum lift height restrictions imposed by Nuclear Regulatory Commission (NRC) staff. As a result, the licensee had not adequately evaluated whether the plant licensing basis necessitated retention of cask lift height limitations when transitioning from the use of the 25 ton NFS 4 or 25 ton NAC 1 spent fuel shipping cask and 70 ton IF 300 spent fuel shipping cask to the heavier 105 ton NUHOMS cask. The licensee entered this issue into its corrective action system.

The inspectors determined that the failure to update the USAR with the cask lift height restrictions for the 25 ton and 70 ton spent fuel cask was contrary to 10 CFR 50.71(e) and was a performance deficiency warranting a significance evaluation. Violations of 10 CFR 50.71 (e) are dispositioned using the traditional enforcement process instead of the SDP because they are considered to be violations that potentially impede or impact the regulatory process. However, if possible, the underlying finding is evaluated under the SDP to determine the significance of the violation. The finding was determined to be more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," because, if left uncorrected, the performance deficiency could have led to a more significant safety concern. Specifically, the inspectors could not readily conclude that the absence of lift height limitations would not require additional calculational analyses and/or require a license amendment. The inspectors determined that the finding was of very low safety significance following a qualitative significance determination review. Specifically, the inspectors determined that only seismic events exceeding the level of an Operational Basis Earthquake (OBE) of 0.03g could impact core damage frequency (CDF). The licensee supplied information that the median annual probability of exceeding the peak ground acceleration for the OBE at Monticello was approximately 7.0E 4/yr. In addition, the predicted shipping cask lifts was 19.2/yr with an average lift duration of 30 minutes. Thus, the frequency of exceeding the OBE while lifting a shipping cask was estimated to be 7.7E 7/year. This value is a bounding frequency estimate for delta CDF in that it does not imply with certainty that there will be a cask drop during an earthquake nor does it imply with certainty of core damage during an earthquake given a cask drop. The Senior Reactor Analyst (SRA) concluded that the risk due to simultaneous occurrence of an OBE or greater seismic event during use of the reactor building crane for shipping cask lifts was best characterized as very low (Green). The inspectors determined that this finding did not reflect current performance because it was a legacy issue with the failure to properly update the USAR occurring almost 30 years ago and, therefore, there was no cross cutting aspect associated with this finding.

The Performance Deficiency associated with this finding is assigned tracking #05000263/2011003-03.
Inspection Report# : 2011003 *(pdf)*

Significance: G Jun 30, 2011
Identified By: NRC
Item Type: FIN Finding
FAILURE TO UPDATE USAR FOR CASK LEFT HEIGHT RESTRICTIONS.
“Periodic Update of the Final Safety Analysis Report” and an accompanying Green finding were identified by the inspectors for the licensee’s failure to update the Updated Safety Analysis Report (USAR) with the cask maximum lift height restrictions imposed by Nuclear Regulatory Commission (NRC) staff. As a result, the licensee had not adequately evaluated whether the plant licensing basis necessitated retention of cask lift height limitations when transitioning from the use of the 25 ton NFS 4 or 25 ton NAC 1 spent fuel shipping cask and 70 ton IF 300 spent fuel shipping cask to the heavier 105 ton NUHOMS cask. The licensee entered this issue into its corrective action system.

The inspectors determined that the failure to update the USAR with the cask lift height restrictions for the 25 ton and 70 ton spent fuel cask was contrary to 10 CFR 50.71(e) and was a performance deficiency warranting a significance evaluation. Violations of 10 CFR 50.71 (e) are dispositioned using the traditional enforcement process instead of the SDP because they are considered to be violations that potentially impede or impact the regulatory process. However, if possible, the underlying finding is evaluated under the SDP to determine the significance of the violation. The finding was determined to be more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," because, if left uncorrected, the performance deficiency could have led to a more significant safety concern. Specifically, the inspectors could not readily conclude that the absence of lift height limitations would not require additional calculational analyses and/or require a license amendment. The inspectors determined that the finding was of very low safety significance following a qualitative significance determination review. Specifically, the inspectors determined that only seismic events exceeding the level of an Operational Basis Earthquake (OBE) of 0.03g could impact core damage frequency (CDF). The licensee supplied information that the median annual probability of exceeding the peak ground acceleration for the OBE at Monticello was approximately 7.0E 4/yr. In addition, the predicted shipping cask lifts was 19.2/yr with an average lift duration of 30 minutes. Thus, the frequency of exceeding the OBE while lifting a shipping cask was estimated to be 7.7E 7/year. This value is a bounding frequency estimate for delta CDF in that it does not imply with certainty that there will be a cask drop during an earthquake nor does it imply with certainty of core damage during an earthquake given a cask drop. The Senior Reactor Analyst (SRA) concluded that the risk due to simultaneous occurrence of an OBE or greater seismic event during use of the reactor building crane for shipping cask lifts was best characterized as very low (Green). The inspectors determined that this finding did not reflect current performance because it was a legacy issue with the failure to properly update the USAR occurring almost 30 years ago and, therefore, there was no cross cutting aspect associated with this finding.

The associated Traditional Enforcement item is Non-Cited Violaton (NCV) 05000263/2011003-02.
Inspection Report# : 2011003 *(pdf)*

Significance: G Mar 31, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO CONTROL A LEVEL 1 FME AREA DURING NEW FUEL RECEIPT ACTIVITIES.
A finding of very low safety significance and associated NCV of Technical Specification 5.4, “Procedures,” was identified by the inspectors when the licensee failed to implement the requirements of their foreign material exclusion (FME) and control procedure during new fuel receipt activities. Specifically, the inspectors observed two operators exiting and re-entering a Level 1 FME area, without the knowledge of the FME monitor, at a point that was not being controlled by the FME monitor. When informed of the issue, the licensee took corrective actions to address the issue.
The inspectors determined that the licensee’s failure to adequately implement the requirements of their FME control procedure during new fuel receipt activities to prevent the unmonitored access of two operators into a Level 1 FME area was a performance deficiency because it was the result of the failure to meet a requirement or a standard, the cause was reasonably within the licensee’s ability to foresee and correct, and should

have been prevented. The inspectors screened the performance deficiency per IMC 0612, “Power Reactor Inspection Reports,” Appendix B, and determined that the issue was more than minor because it impacted the human performance attribute of the Barrier Integrity Cornerstone’s objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. The inspectors applied IMC 0609, Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings,” to this finding. The inspectors utilized Column 3 of the Table 4a worksheet to screen the finding. Since the finding only had the potential to impact the fuel barrier, it screened to be of very low safety significance. This finding has a cross-cutting aspect in the area of Human Performance, Work Practices because the licensee did not define and effectively communicate expectations regarding procedural compliance and perosnnel following procedures (H.4(b)).
Inspection Report# : 2011002 *(pdf)*

Significance: G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO IMPLEMENT CORRECTIVE ACTIONS TO ADDRESS A DEFICIENCY ASSOCIATED WITH THE DOOR INTERLOCK ON AIRLOCK 413.
A finding of very low safety significance and associated NCV of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, was identified by the inspectors when the licensee failed to implement corrective actions for a condition adverse to quality. The condition adverse to quality was a deficiency associated with the door interlock on airlock 413 which contributed to loss of secondary containment boundary event. Subsequent to the August 5, 2010, event, the licensee initiated administrative controls on all airlocks with a similar design to airlock 413 and are currently evaluating other means of addressing air lock integrity. The inspectors determined that the contributing cause that provided the most insight into the performance deficiency was associated with the cross cutting area of Problem Identification and Resolution, having Corrective Action components, and involving aspects associated with thoroughly evaluating problems such that the resolution addresses the causes and extent of condition as necessary. [P.1(c)].
The inspectors determined that the licensee’s failure to implement adequate corrective actions for a condition adverse to quality was a performance deficiency because it was the result of the failure to meet a requirement; the cause was reasonably within the licensee’s ability to foresee and correct; and should have been prevented. The inspectors screened the performance deficiency per IMC 0612, Power Reactor Inspection Reports, Appendix B, and determined that the issue was more than minor because it impacted the configuration control attribute of the Barrier Integrity Cornerstone's objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The inspectors applied IMC 0609, Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings,” to this finding. Since the finding resulted in a momentary loss of the secondary containment boundary, the inspectors evaluated the finding under the Containment Barrier Cornerstone. Utilizing Column 4 of the Table 4a worksheet, the inspectors answered “Yes” to Question 1. Since the finding only resulted in the degradation of the radiological barrier function provided for the control room; auxiliary building; spent fuel pool; or standby gas treatment system; the finding was screened to be of very low safety significance.

Inspection Report# : 2010005 *(pdf)*

Significance: G Sep 30, 2010
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
INADEQUATE ELECTRICAL ISOLATION DURING DEMOLITION ACTIVITY.
A finding of very low safety significance and associated non cited violation (NCV) of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was self revealed when the licensee failed to adequately implement the requirements of their fleet tagging procedure, a procedure affecting quality, during the demolition of the ‘A’ train of the combustion gas control system (CGCS). This failure directly led to workers being unprotected from existing 24 Vdc, and potentially 120 Vac, during the removal of cables C259 SV40008A/1 and C259 SV4009A/1. In addition, cutting of the energized cables resulted in the loss of position indication for three primary

containment isolation valves which are required by Technical Specifications. The licensee promptly took actions to restore the affected containment isolation valves to an operable status and entered this event into their corrective action program for further evaluation. The inspectors determined that the contributing cause that provided the most insight into the performance deficiency was associated with the cross cutting area of Human Performance, having work control components, and involving aspects associated with appropriately coordinating work activities by incorporating job site conditions which may impact human performance and plant systems and components. [H.3(a)]

The inspectors determined that the licensee's failure to adequately implement their work order planning and tagging processes to protect workers and equipment from existing electrical hazards during the demolition of the 'A' train of the CGCS system was a performance deficiency because it was the result of the failure to meet a requirement; the cause was reasonably within the licensee's ability to foresee and correct; and should have been prevented. The inspectors applied IMC 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," to this finding. Since the finding directly resulted in the loss of position indication for three containment isolation valves which are required by Technical Specifications, the inspectors evaluated the finding under the Containment Barrier Cornerstone. Utilizing Column 4 of the Table 4a worksheet, the inspectors answered "Yes" to question 1. Since the finding only resulted in the degradation of the radiological barrier function provided for the control room, auxiliary building, spent fuel pool, or standby gas treatment (SBGT) system, the finding was screened to be of very low safety significance.
Inspection Report# : 2010004 *(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Significance: G Jun 30, 2011
Identified By: Self-Revealing
Item Type: FIN Finding
FAILURE TO MAINTAIN RADIATION EXPOSURE ALARA DURING INBOARD MAIN STEAM ISOLATION VALVE REPAIR.
A finding of very low safety significance (Green) was self revealed due to the licensee having unplanned and unintended occupational collective radiation dose because of deficiencies in the licensee's as-low-as-is-reasonably-achievable (ALARA) planning and work control program. Specifically, the licensee failed to properly incorporate ALARA strategies or insights while planning and executing a maintenance activity on the 'C' inboard main steam isolation valve. This issue resulted in the expansion of collective exposure for this work from 4.044 person rem to 9.654 person rem. The licensee entered this issue into their corrective action program as CAP 1281395.

The finding was more than minor because it was associated with the program and process attribute of the Occupational Radiation Safety Cornerstone. Additionally, this issue affected the cornerstone objective of ensuring the adequate protection of the worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. Also, the finding was similar to Example 6.i in Appendix E of IMC 0612, in that it resulted in a collective exposure of greater than 5 person rem and exceeded the outage goal by greater than 50 percent. The inspectors determined that this finding was of very low safety significance because Monticello Nuclear Generating Plant's (MNGP's) current three-year rolling average collective dose is 136.266 person rem, less than the 240 person rem per unit standard. This finding had a cross cutting aspect in the area of Human Performance, related to the cross cutting component of work control, in that the outage plan did not adequately incorporate actions to address the impact of work on different job activities.
Inspection Report# : 2011003 *(pdf)*

Public Radiation Safety

Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

Miscellaneous

Last modified : October 14, 2011