Monticello
4Q/2011 Plant Inspection Findings

Initiating Events

Significance: G Dec 31, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
'E' CONDENSATE DEMINERALIZER ALARM RESPONSE PROCEDURE LIMITS EXCEEDED.
The inspectors identified a finding of very low safety significance and non cited violation (NCV) of Technical Specification (TS) 5.4.1, “Procedures,” when the operators did not take conservative action to address a high differential pressure condition on an inservice condensate demineralizer vessel. Specifically, operators allowed the ‘E’ condensate demineralizer to exceed differential pressure operating limits prescribed in Alarm Response Procedure 80 DPAH 2215, “Vessel T 7E D/P High,” and remain above those prescribed limits for approximately a shift before taking action to correct the abnormal condition. Specific corrective actions taken by the licensee to address this issue included updating the applicable alarm response procedures and operating procedures to reflect current system limitations; engineering management reinforcing the expectation that informal processes are not acceptable when communicating technical guidance to operations staff; and site management reinforcing the expectation that, once a degrading trend is recognized, actions must be taken in sufficient time to prevent crossing established operating limits.

The inspectors determined that the licensee’s failure to maintain the ‘E’ condensate demineralizer differential pressure within prescribed operational limits 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 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. For transient initiators, 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 of functions will not be available,” and determined the finding to be of very low safety significance. 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 the licensee planning and coordinating work activities, consistent with nuclear safety, specifically the need for planned contingencies, compensatory actions, and abort criteria [H.3(a)].

Inspection Report# : 2011005 *(pdf)*

Significance: G Dec 31, 2011
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
INADEQUATE COMPLETION OF CAPRS ASSOCIATED WITH 2RS TO 2R FEEDER CABLE TESTING.
A finding of very low safety significance and NCV of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions,” was self revealed following a reactor scram, which was the direct result of an electric plant realignment caused by a faulted feeder cable and lockout of the station’s 2R transformer. Specifically, annual testing to monitor the performance of the 2R feeder cables, which was put in place as a corrective action to prevent recurrence to address issues identified subsequent to a similar event in 2008, had not been performed since the cables were placed back in service following that event. To address the identified material deficiencies, the licensee replaced and tested the electrical cables between 2RS and 2R in their entirety, employing a new route designed to avoid cable submergence. Additional corrective actions were put in place to strengthen the licensee’s planned maintenance deferral process and their cable condition monitoring program.
The inspectors determined that the licensee’s failure to perform annual testing of the 2R transformer feeder cables, as

required by the station’s planned maintenance program, 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 determined that the issue was more than minor because it impacted the Configuration Control 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. Because the finding contributed to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions would not be available, the Region III Senior Reactor Analyst (SRA) performed a Phase 3 analysis, and screened the finding to be of very low safety significance. 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 licensees’ making safety significant or risk-significant decisions using a systematic process to ensure safety is maintained [H.1(a)].

Inspection Report# : 2011005 *(pdf)*

Significance: G Dec 31, 2011
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
ROD WORTH MINIMIZER INOPERABLE DURING REACTOR PLANT STARTUP.
A finding of very low safety significance and NCV of TS 3.3.2.1, “Control Rod Block Instrumentation,” was self revealed to the operating crew, when normal startup testing could not be accomplished due to improperly configured equipment. Specifically, the operating crew transitioned from Mode 4 to Mode 2, with the rod worth minimizer (RWM) mode switch in the BYPASS position. With the RWM mode switch in the BYPASS position and the required actions of 3.3.2.1(c) not met, the requirements of TS 3.3.2.1, that the RWM be operable in Mode 1 and Mode 2 when thermal power is less than or equal to 10 percent rated thermal power, could not be met. Actions taken by the licensee in response to this event included declaring the event a reactivity management event; making an NRC notification under 50.72(b)(3)(v)(D); resetting their site event clock; providing additional training for the applicable operating crew; and revising procedures associated with this event to clarify the sequencing of key activities associated with the transition between Mode 4 and Mode 2.
The inspectors determined that the licensee’s failure to properly control the configuration of the RWM prior to entering an operating mode that required its operability 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 determined that the issue was more than minor because it impacted the Configuration Control 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 answered ‘No’ to the questions associated with transient initiators and screened the finding to be of very low safety significance. 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 personnel work practices that support human performance, specifically in the areas of pre job briefing, self and peer checking, and proper documentation of activities [H.4(a)].

Inspection Report# : 2011005 *(pdf)*

Significance: G Jul 15, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
HYDROGEN BOTTLES LOCATED BELOW RHR SYSTEM CABLES.
The inspectors identified a finding of very low safety significance and associated NCV of Title 10, Code of Federal Regulations (CFR) Part 50, Appendix B, Criterion III, “Design Control,” for the failure to evaluate the impact of the installation of the hydrogen/oxygen analyzer system on safety-related residual heat removal (RHR) system cables. Specifically, the licensee failed to evaluate how a failure of the hydrogen bottles and the resulting fire or explosion could impact RHR cables located directly above the hydrogen bottles. The licensee entered this issue into their

corrective action program to review the placement of the hydrogen bottles.

The inspectors determined that the finding was more than minor because the finding was associated with the Initiating Events cornerstone attribute of Protection Against External Factors (Fire) and affected the cornerstone's objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The finding was of very low safety significance due to the low fire initiating frequency and the availability of remaining mitigating systems. This finding did not have a cross-cutting aspect because the finding was not representative of current performance.

Inspection Report# : 2011008 *(pdf)*

Significance: G 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: G 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: SL-IV Dec 31, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO MAKE A REQUIRED 60 DAY EVENT REPORT PER 10 CFR 50.73(a)(2)(vii)(A-D).
The inspectors identified a Severity Level IV NCV and associated finding of very low safety significance of 10 CFR 50.73(a)(2)(vii)(A D), “Licensee Event Report System,” for the failure to report an event to the NRC within 60 days, where a single cause or condition caused two independent trains to become inoperable in a single system designed to

help maintain safe reactor shut down, remove residual heat, control radioactive releases, or mitigate accidents. Specifically, on September 29, 2011, the licensee identified that the surveillance test procedures being used to demonstrate load reject capabilities of both EDGs had never contained the correct load rejection testing requirements from the applicable design documents. As a result, the surveillances were considered never met, and both EDGs were declared inoperable. During their evaluation and subsequent reporting of the issue, the licensee failed to recognize that the inoperability of both diesel generators caused by a single common cause was reportable to the NRC within 60 days under the 50.73 common cause criterion. The licensee entered this issue into their corrective action program (CAP 1318116). Corrective actions for this issue included plans to revise their existing licensee event report (LER) and to perform an apparent cause evaluation to further evaluate the issue.
The inspectors determined that the failure to report required plant events or conditions to the NRC in accordance with reporting requirements 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. In addition, it had the potential to impede or impact the regulatory process. As a result, the NRC dispositions violations of 10 CFR 50.73 using the traditional enforcement process instead of the SDP. However, if possible, the underlying technical issue is evaluated using the SDP. In this case, the inspectors determined that the licensee failed to develop and implement adequate Emergency Diesel Generator (EDG) testing procedures during their transition to the Improved Technical Specifications in 2006, which resulted in both EDGs being declared TS inoperable, but available for use. The inspectors determined that the performance deficiency was more than minor because it was associated with the Mitigating Systems Cornerstone attributes of Human Performance and Procedure Quality and 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). Using IMC 0609, Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings,” the inspectors determined that the finding had very low safety significance because they answered ‘No’ to all five questions contained in Column 2 of the Table 4a worksheet. As a result, the inspectors determined that the finding had very low safety significance (Green). In accordance with Section 6.9.d.9 and 6.9.d.10 of the NRC Enforcement Policy, this violation was categorized as Severity Level IV because it was an example where the licensee failed to make a report required by 10 CFR 50.73; it represented a failure to identify all applicable reporting codes on an LER that may impact the completeness or accuracy of other information submitted to the NRC; and the underlying technical issue was evaluated by the SDP and determined to be of very low safety significance. The inspectors determined that the contributing cause that provided the most insight into the performance deficiency affected the cross cutting area of Problem Identification and Resolution, having corrective action program components, and involving aspects associated with properly classifying and evaluating for reportability conditions adverse to quality [P.1(c)].

The Performance Deficiency associated with this finding is assigned tracking #05000263/2011005-06.
Inspection Report# : 2011005 *(pdf)*

Significance: G Dec 31, 2011
Identified By: NRC
Item Type: FIN Finding
FAILURE TO MAKE A REQUIRED 60 DAY EVENT REPORT PER 10 CFR 50.73(a)(2)(vii)(A-D).
The inspectors identified a Severity Level IV NCV and associated finding of very low safety significance of 10 CFR 50.73(a)(2)(vii)(A D), “Licensee Event Report System,” for the failure to report an event to the NRC within 60 days, where a single cause or condition caused two independent trains to become inoperable in a single system designed to help maintain safe reactor shut down, remove residual heat, control radioactive releases, or mitigate accidents. Specifically, on September 29, 2011, the licensee identified that the surveillance test procedures being used to demonstrate load reject capabilities of both EDGs had never contained the correct load rejection testing requirements from the applicable design documents. As a result, the surveillances were considered never met, and both EDGs were declared inoperable. During their evaluation and subsequent reporting of the issue, the licensee failed to recognize that the inoperability of both diesel generators caused by a single common cause was reportable to the NRC within 60 days under the 50.73 common cause criterion. The licensee entered this issue into their corrective action program (CAP 1318116). Corrective actions for this issue included plans to revise their existing licensee event report (LER) and to perform an apparent cause evaluation to further evaluate the issue.
The inspectors determined that the failure to report required plant events or conditions to the NRC in accordance with reporting requirements 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. In addition, it had the potential to impede or impact the regulatory process. As a result, the NRC

dispositions violations of 10 CFR 50.73 using the traditional enforcement process instead of the SDP. However, if possible, the underlying technical issue is evaluated using the SDP. In this case, the inspectors determined that the licensee failed to develop and implement adequate Emergency Diesel Generator (EDG) testing procedures during their transition to the Improved Technical Specifications in 2006, which resulted in both EDGs being declared TS inoperable, but available for use. The inspectors determined that the performance deficiency was more than minor because it was associated with the Mitigating Systems Cornerstone attributes of Human Performance and Procedure Quality and 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). Using IMC 0609, Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings,” the inspectors determined that the finding had very low safety significance because they answered ‘No’ to all five questions contained in Column 2 of the Table 4a worksheet. As a result, the inspectors determined that the finding had very low safety significance (Green). In accordance with Section 6.9.d.9 and 6.9.d.10 of the NRC Enforcement Policy, this violation was categorized as Severity Level IV because it was an example where the licensee failed to make a report required by 10 CFR 50.73; it represented a failure to identify all applicable reporting codes on an LER that may impact the completeness or accuracy of other information submitted to the NRC; and the underlying technical issue was evaluated by the SDP and determined to be of very low safety significance. The inspectors determined that the contributing cause that provided the most insight into the performance deficiency affected the cross cutting area of Problem Identification and Resolution, having corrective action program components, and involving aspects associated with properly classifying and evaluating for reportability conditions adverse to quality [P.1(c)].

The associated Traditional Enforcement item is Non-Cited Violaton (NCV) 05000263/2011005-05.
Inspection Report# : 2011005 *(pdf)*

Significance: G Dec 15, 2011
Identified By: NRC
Item Type: FIN Finding
Failure to Follow Fire Water Aging Management Program Implementing Procedure
The inspectors identified a finding of very low safety significance (Green) involving the licensee’s failure to accomplish activities affecting quality in accordance with procedures. Specifically, the licensee failed to incorporate operating experience in accordance with procedures. This impacted the licensee’s ability to implement an effective aging management program for the fire protection system. No violation of NRC requirements was identified.

The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of Protection Against External Factors (Fire) and 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). Using IMC 0609, Appendix F, Fire Protection SDP, and the Monticello SPAR model, the inspectors determined that this finding had very low safety significance. The inspectors did not identify an associated crosscutting aspect for this finding. (Section 4OA5.7b.(1))
Inspection Report# : 2011010 *(pdf)*

Significance: G Sep 30, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO FOLLOW EMERGENCY DIESEL GENERATOR QUARTERLY SURVEILLANCE PROCEDURE.
The inspectors identified a finding of very low safety significance and an associated non cited violation (NCV) of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” when the licensee failed to follow the quarterly emergency diesel generator (EDG) surveillance procedure during testing of the EDG air start system. Specifically, the licensee failed to follow a procedural step that involved in service testing of a check valve in the EDG air start system that, if degraded, could allow air to bleed out of the starting air tanks which are required for diesel generator operability. The licensee entered this issue into their corrective action program (CAP), and corrective actions for this issue included suspension of the test, performance of a Human Performance Investigation Team review, and disqualification of the individual performing the test. 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 using human error

prevention techniques during performance of work activities. [H.4(a)]

The inspectors determined that the licensee's failure to follow their EDG surveillance procedure 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 Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, and determined that the issue was more than minor because the performance deficiency was associated with the Human Performance attribute of the Mitigating Systems Cornerstone and affected the cornerstone's objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). As a result, this finding was evaluated under the Mitigating Systems Cornerstone. 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. The finding was determined to have very low safety significance because the inspectors answered "No" to all five questions. (Section 1R22)
Inspection Report# : 2011004 *(pdf)*

Significance: G Jul 15, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO INSPECT AND TEST THE MCR AIR INTAKE SMOKE DETECTOR.
The inspectors identified a finding of very low safety significance and associated NCV of License Condition 2.C.4 for the licensee's failure to inspect and test the main control room (MCR) air intake smoke detector. Specifically, the licensee failed to inspect and test the smoke detector between 2006 and 2011 as required by the preventative maintenance program. The licensee successfully tested the detector once the performance deficiency was identified and entered this issue into their corrective action program to evaluate the status of the detector.

The inspectors determined that the finding was more than minor because if left uncorrected, the failure to inspect and test the MCR air intake smoke detector would become a more significant safety concern. Specifically, if the licensee continued not testing and maintaining the detector it would eventually fail to respond properly and result in a delayed notification to control room operators of a fire that could result in smoke entering the control room. This finding was of very low safety significance because the licensee successfully tested the detector. This finding did not have a cross-cutting aspect because the finding was not representative of current performance.

Inspection Report# : 2011008 *(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)*

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

Significance: G Dec 31, 2011
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
FAILURE TO PROPERLY BLOCK AND BRACE A RADIOACTIVE PACKAGE FOR TRANSPORT.
The inspectors reviewed a self revealed finding of very low safety significance and an associated NCV of 10 CFR 71.5. Specifically, the licensee failed to appropriately block and brace a radioactively contaminated condensate demineralizer vessel within a transport package, such that, the package contents would not compromise and penetrate the transport package. The issue has been entered into the licensee's corrective action program as CR [condition report] 01294652. Corrective actions were implemented to address supervision's responsibilities during shipment preparation regarding appropriate blocking and bracing of package contents.
The finding was more than minor because the performance deficiency could be reasonably viewed as a precursor to a significant event, in that, the penetration of the transportation package by its contents could lead to the inadvertent spread of radioactive contamination in the public domain. Using IMC 0609, Attachment D, for the Public Radiation Safety SDP, the inspectors determined the finding to be of very low safety significance. The inspectors also determined that this finding had a cross cutting aspect in the area of problem identification and resolution (operating experience) [P.2(b)].
Inspection Report# : 2011005 *(pdf)*

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 : March 02, 2012