UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
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LISLE, IL 60532-4352

August 12, 2015

EA-15-130

Mr. Peter A. Gardner
Monticello Nuclear Generating Plant
Northern States Power Company, Minnesota
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT NRC INTEGRATED AND POWER UPRATE INSPECTION REPORT AND EXERCISE OF ENFORCEMENT DISCRETION 05000263/2015002

Dear Mr. Gardner:

On June 30, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Monticello Nuclear Generating Plant. The enclosed report documents the inspection findings, which were discussed on July 21, 2015, with you and other members of your staff.

Based on the results of this inspection, four NRC-identified and two self-revealed findings of very low safety significance were identified. The findings involved violations of NRC requirements. However, because of their very low safety significance, and because the issues were entered into your corrective action program (CAP), the NRC is treating the issues as non-cited violations (NCVs) in accordance with Section 2.3.2 of the NRC Enforcement Policy.

A violation involving a failure to have secondary containment operable during Operations with the Potential to Drain the Reactor Vessel (OPDRV) was identified. Specifically, from April 23, 2015 through May 8, 2015, Monticello Nuclear Generating Plant performed a total of three activities within two work windows without setting secondary containment, which is a violation of Technical Specification (TS) 3.6.4.1. The NRC issued Enforcement Guide Memorandum (EGM) 11-003, "Enforcement Guidance Memorandum on Dispositioning Boiling Water Reactor Licensee Noncompliance with Technical Specification Containment Requirements During Operations with a Potential for Draining the Reactor Vessel," Revision 2, on December 13, 2013, allowing for the exercise of enforcement discretion for such OPDRV-related TS violations, when certain criteria are met. The NRC concluded that Monticello Nuclear Generating Plant met these criteria during the activities for which the EGM was invoked. Therefore, I have been authorized, after consultation with the Director, Office of Enforcement, and the Regional Administrator, to exercise enforcement discretion and refrain from issuing enforcement for the violation.

If you contest these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555–0001, with copies to the Regional Administrator–Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555–0001; and the NRC Resident Inspector at Monticello

Nuclear Generating Plant. In addition, if you disagree with a cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Monticello Nuclear Generating Plant.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390, “Public Inspections, Exemptions, Requests for Withholding,” of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC’s Public Document Room or from the Publicly Available Records (PARS) component of the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Kenneth Riemer, Branch Chief
Branch 2
Division of Reactor Projects

Docket No. 50–263
License No. DPR–22

Enclosure:
Inspection Report 05000263/2015002;
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50–263
License No:	DPR–22
Report No:	05000263/2015002
Licensee:	Northern States Power Company, Minnesota
Facility:	Monticello Nuclear Generating Plant
Location:	Monticello, MN
Dates:	April 1, 2015, through June 30, 2015
Inspectors:	P. Zurawski, Senior Resident Inspector P. Voss, Resident Inspector R. Elliott, Reactor Engineer S. Bell, Health Physicist T. Bilik, Senior Reactor Inspector (Lead) M. Ziolkowski, Reactor Engineer N. McMurray, NSPDP (Observer) J. Park, Reactor Inspector (Observer)
Approved by:	K. Riemer, Branch Chief Branch 2 Division of Reactor Projects

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SUMMARY OF FINDINGS

Inspection Report (IR) 05000263/2015002; 04/01/2015–06/30/2015; Monticello Nuclear Generating Plant; Fire Protection, Inservice Inspection Activities, Maintenance Risk Assessments and Emergent Work Control, Operability Evaluations, Drill Evaluations, and Follow-Up of Events and Notices of Enforcement Discretion.

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Six Green findings were identified by the inspectors. Each finding was considered a non-cited violation (NCV) of U.S. Nuclear Regulatory Commission (NRC) regulations. The significance of inspection findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" dated June 2, 2011. Cross-cutting aspects are determined using IMC 0310, "Aspects Within the Cross-Cutting Areas" effective date December 4, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy dated February 4, 2015. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG–1649, "Reactor Oversight Process" Revision 5, dated February 2014.

Cornerstone: Initiating Events

Green. A self-revealed 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 due to the failure to properly implement Procedure 0304-01, "Safeguard Bus Loss of Voltage Protection Relay Unit Calibration – Safeguards Bus No. 15." Specifically, electrical maintenance workers failed to comply with Step 20 which directed the installation of a jumper between terminals ZX10 and ZX11 in an electrical panel, when they incorrectly installed the electrical jumper between terminals ZX11 and ZX12. This resulted in the loss of the Division I safety related 4160 Volts Alternating Current (Vac), 480 Vac, and 125 Volts Direct Current (Vdc) electrical buses, which subsequently led to the loss of shutdown cooling (SDC) for approximately 3 hours and 15 minutes. Initial corrective actions for this issue included immediately invoking strict plant status controls to focus efforts on recovery, restoring the electrical buses and SDC to operation, and reinforcing risk recognition and human performance tools. This issue was entered into the licensee's CAP (CAP 1477351) and a root cause evaluation (RCE) was in progress at the time this inspection period concluded.

The inspectors determined that the issue was more than minor because it adversely impacted the Initiating Events Cornerstone attribute of Human Performance and Configuration Control, and affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors utilized IMC 0609, Appendix G for shutdown operations and determined that the issue was 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, Avoid Complacency aspect because of the failure of licensee individuals to implement error reduction tools and the failure of the organization to plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes [H.12]. (Section 4OA3)

Green. A self-revealed finding of very low safety significance and an associated NCV of technical specification (TS) 5.4.1, "Procedures," was identified on May 16, 2015, when the licensee failed to implement procedure FP-OP-TAG-01, "Fleet Tagging," for equipment control activities associated with the Scram Discharge Volume (SDV). Specifically, the licensee failed to ensure that clearance order checklist 58972-03 restored valve I-CRD-R-26, an SDV instrument vent valve, to its normal position prior to returning the SDV system to service. As a result, during subsequent reactor coolant system (RCS) pressure boundary testing, RCS water leaked out onto the reactor building floor through the open vent line, creating an unplanned operation with a potential for draining the reactor vessel (OPDRV). This issue was entered into the licensee's CAP (CAP 1479307). Immediate corrective actions included termination of the leakage by closing and capping the SDV vent line and resetting the scram. The site initiated an apparent cause evaluation (ACE), which was in progress at the end of the inspection period.

The inspectors determined that the failure to adequately restore the SDV system to service in accordance with fleet tagging requirements was a performance deficiency requiring evaluation. The inspectors determined that the finding was more than minor in accordance with IMC 0612, Appendix B, because it adversely impacted the Initiating Events Cornerstone attributes of Configuration Control and Procedure Quality, and affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors evaluated the finding using IMC 0609, Attachment 4, Phase 1 – Initial Screening and Characterization of Findings, which required an analysis using IMC 0609 Appendix G, the Shutdown Operations significance determination process (SDP) since the reactor was in Mode 4 (cold shutdown). The finding was assessed in accordance with IMC 0609 Appendix G, Attachment 1, Exhibit 2 for Initiating Events. Using IMC 0609 Appendix G, Attachment 3, for a Phase 2 analysis, the inspectors determined it to have very low safety significance. The inspectors concluded that this finding was cross-cutting in the Human Performance, Challenge the Unknown aspect because of the failure of individuals to stop when faced with uncertain conditions, and the failure to ensure that risks are evaluated and managed before proceeding [H.11]. (Section 1R15)

Cornerstone: Mitigating Systems

Green. The inspectors identified a finding of very low safety significance and an NCV of TS 5.4.1.d when the licensee failed to implement procedures associated with Fire Protection Program Implementation to ensure that portable fire extinguishers were maintained in accordance with the fire strategy. Specifically, on May 1, 2015, the licensee failed to implement fire protection plan procedures when they failed to control three portable fire extinguishers in the condenser room, a room housing safe shutdown cabling, in accordance with Fire Strategy A.3-12-C. In this case, inspectors found that of the four dry chemical extinguishers required to be stationed in the condenser room, two indicated that they were partially depleted and needed to be recharged, and a third extinguisher was missing entirely. Immediate corrective actions included recharging the partially depleted extinguishers and procuring a portable extinguisher to replace the missing one. This issue was entered into the licensee's CAP (CAP 1477246).

The inspectors determined that the failure to implement the fire strategy procedure to ensure that condenser room portable fire extinguishers were maintained was a

performance deficiency requiring evaluation. The inspectors determined the issue was more than minor in accordance with IMC 0612 Appendix B because it was associated with the Mitigating Systems Cornerstone attribute of Protection Against External Factors—including fire, and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Because the plant was shut down, the inspectors assessed the significance of this finding in accordance with IMC 0609, Appendix G, the Shutdown Operations SDP, and determined that it had 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 Problem Identification and Resolution, Identification aspect because of the failure to implement a CAP with a low threshold for identifying issues, and failure to ensure that individuals identify issues completely, accurately, and in a timely manner in accordance with the program [P.1]. (Section 1R05)

Green. The inspectors identified a Green NCV of Title 10 CFR Part 50, Appendix B, Criterion IX, "Control of Special Processes," for a failure to measure the interpass temperature while performing welding on diesel generator fuel oil modification supports. Consequently, welding was performed without the Code and Procedure required interpass temperature being monitored on a number of welds, a parameter which can affect the mechanical properties of the material being welded. To restore compliance, the welder proceeded to measure the interpass temperatures on the balance of the welds and verified that the interpass temperature did not exceed that allowed by procedure. The licensee entered this issue into its CAP (CAP 1475767).

The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because the inspectors answered "yes" to the more than minor question, "If left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC intervention, the welder would have completed all of the welds without having measured the interpass temperature, a welding parameter which can affect the mechanical properties (e.g., impact properties) of some materials being welded, and if left uncorrected could lead to a potential failure of the weld in service.

In accordance with Table 2, "Cornerstones Affected by Degraded Condition or Programmatic Weakness," of IMC 0609, Attachment 4, "Initial Characterization of Findings," issued June 19, 2012, the inspectors checked the box under the Mitigating Systems Cornerstone because leakage on the Emergency Diesel Generator (EDG) fuel oil system could cause core decay heat removal to be degraded. The inspectors determined this finding was of very-low safety significance (Green) based on answering "yes" to the question in Part A of Exhibit 2, "Mitigating Systems Screening Questions," in IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power," issued on June 19, 2012. Specifically, the inspectors answered "yes" to the screening question "If the finding is a deficiency affecting the design or qualification of a mitigating Structure, System, or Component (SSC), does the SSC maintain its operability or functionality?" The welder proceeded to measure the interpass temperatures on the balance of the welds and verified that the interpass temperature did not exceed that allowed by procedure, and the issue did not result in the actual loss of the operability or functionality of a safety system.

The inspectors determined that the primary cause of the failure to monitor the interpass temperature procedure was related to the cross-cutting component of Problem Identification and Resolution, Operating Experience (P.5). Specifically, the organization failed to effectively implement external operating experience in a timely manner. (Section 1R08)

Cornerstone: Barrier Integrity

Green. The inspectors identified a finding of very low safety significance and an associated NCV of TS 3.6.4.1, Secondary Containment and TS 3.6.4.3, Standby Gas Treatment System (SBGT) because the licensee did not maintain secondary containment and the SBGT system operable as required during activities considered OPDRVs. Specifically, on April 14, 2015, and again on May 13, 2015, the licensee failed to classify activities associated with draining reactor inventory as OPDRVs while relying on an automatic isolation function for the drain path, and as a result failed to maintain required equipment operable during these activities. Once questioned by the inspectors, the licensee took action to control other outage related draining activities as OPDRVs and placed this issue into its CAP (CAP 1479284).

The inspectors determined that the failure to maintain secondary containment and SBGT operable while an OPDRV was in progress was a performance deficiency. The performance deficiency was more than minor because it was associated with the configuration control attribute of the Barrier Integrity Cornerstone, and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, RCS, and containment) protect the public from radionuclide releases caused by accidents or events because the secondary containment boundary and the SBGT were not maintained operable during an OPDRV activity. The inspectors evaluated the finding using IMC 0609, Attachment 4, Phase 1 – Initial Screening and Characterization of Findings, which required an analysis using IMC 0609 Appendix G, the Shutdown Operations SDP since the reactor was shut down. The finding was assessed in accordance with IMC 0609 Appendix G, Attachment 1, Exhibit 4 and Appendix H for containment integrity findings. Using Appendix H, the inspectors concluded the finding had very low safety significance (Green) because decay heat was low and containment was deinerted. 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, Documentation aspect because of the failure of the licensee to create and maintain complete, accurate and up-to-date documentation [H.7]. (Section 1R13)

Green. The inspectors identified a finding of very low safety significance and an associated NCV of TS 5.4.1, “Procedures,” on April 15, 2015, when the licensee failed to implement procedure 9001, “Reactor Well & Dryer-Separator Storage Pool Filling Procedure,” for refueling preparation activities. Specifically, when faced with indications that the condensate storage tanks (CSTs) did not contain enough water inventory to complete outage critical path reactor pressure vessel (RPV) flooding activities, the licensee failed to implement 9001 procedure steps for using prescribed equipment and methods to fill the reactor cavity. With the proceduralized methods unavailable, operators used the site decision-making process to utilize demineralizer water hoses to fill the cavity rather than processing required 9001 procedure changes. This issue was entered into the licensee’s CAP (CAP 1474891). Immediate corrective actions included action to initiate the procedure change process for 9001 and department communication

to Operations regarding the incident, emphasizing that the decision making process is not a substitute for the procedure change process.

The inspectors determined that the failure to fill the reactor cavity in accordance with the 9001 reactor well filling procedure was a performance deficiency requiring evaluation. The inspectors evaluated IMC 0612, Appendix E, and did not find any similar examples of minor issues. The inspectors determined that the finding was more than minor in accordance with IMC 0612, Appendix B, because if left uncorrected, the performance deficiency had the potential to lead to a more significant safety concern. Specifically, the operations crew's use of the decision-making process to support outage critical path by bypassing proceduralized steps and performing activities using methods contrary to the procedure could lead to a more significant safety concern. In addition, if performed incorrectly (i.e. without flushing the hoses prior to use), the use of demineralizer hoses could introduce foreign material into the core and challenge the integrity of the fuel cladding barrier. The inspectors evaluated the finding using IMC 0609, Attachment 4, Phase 1 – Initial Screening and Characterization of Findings, which required an analysis using IMC 0609 Appendix G, the Shutdown Operations SDP since the reactor was in Mode 5 (refueling). The finding was assessed in accordance with IMC 0609 Appendix G, Attachment 1, Exhibit 4 for Barrier Integrity and determined to have very low safety significance.

The inspectors concluded that this finding was cross-cutting in the Human Performance, Conservative Bias aspect because of the failure of the individuals to use decision-making practices that emphasize prudent choices over those that are simply allowable, and the failure to ensure that proposed actions are determined to be safe in order to proceed, rather than unsafe in order to stop [H.14]. (Section 1R13)

REPORT DETAILS

Summary of Plant Status

Monticello began the inspection period operating at 95 percent (1908 MWt) of its newly licensed extended power uprate (EPU) power of 2004 MWt. On April 12, 2015, the licensee shut the reactor down for planned refueling outage (RFO) RFO27. The outage ended on May 28, 2015, and the unit was returned to 95 percent power on June 4, 2015. The licensee resumed EPU testing activities on June 10, 2015 raising power to 97.5 percent (1953 MWt). After completing the 97.5 percent power EPU data evaluation, the licensee raised power on June 30, 2015 to 100 percent (2004 MWt). At the end of the inspection period, Monticello was holding power at 97.5 percent (1953 MWt) pending review of 100 percent power data which had been acquired.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness of Offsite and Alternate AC Power Systems

a. Inspection Scope

The inspectors verified that plant features and procedures for operation and continued availability of offsite and alternate alternating current (AC) power systems during adverse weather were appropriate. The inspectors reviewed the licensee's procedures affecting these areas and the communications protocols between the transmission system operator (TSO) and the plant to verify that the appropriate information was being exchanged when issues arose that could impact the offsite power system. Examples of aspects considered in the inspectors' review included:

- coordination between the TSO and the plant during off-normal or emergency events;
- explanations for the events;
- estimates of when the offsite power system would be returned to a normal state; and
- notifications from the TSO to the plant when the offsite power system was returned to normal.

The inspectors also verified that plant procedures addressed measures to monitor and maintain availability and reliability of both the offsite AC power system and the onsite alternate AC power system prior to or during adverse weather conditions. Specifically, the inspectors verified that the procedures addressed the following:

- actions to be taken when notified by the TSO that the post-trip voltage of the offsite power system at the plant would not be acceptable to assure the continued operation of the safety-related loads without transferring to the onsite power supply;
- compensatory actions identified to be performed if it would not be possible to predict the post-trip voltage at the plant for the current grid conditions;

- re-assessment of plant risk based on maintenance activities which could affect grid reliability, or the ability of the transmission system to provide offsite power; and
- communications between the plant and the TSO when changes at the plant could impact the transmission system, or when the capability of the transmission system to provide adequate offsite power was challenged.

Documents reviewed are listed in the Attachment to this report. The inspectors also reviewed CAP items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their corrective action plan (CAP) in accordance with station corrective action procedures.

This inspection constituted one readiness of offsite and alternate AC power systems sample as defined in Inspection Procedure (IP) 71111.01–05.

b. Findings

No findings were identified.

.2 External Flooding

a. Inspection Scope

The inspectors evaluated the design, material condition, and procedures for coping with the design basis probable maximum flood. The evaluation included a review to check for deviations from the descriptions provided in the Updated Safety Analysis Report (USAR) for features intended to mitigate the potential for flooding from external factors. As part of this evaluation, the inspectors checked for obstructions that could prevent draining, checked that the roofs did not contain obvious loose items that could clog drains in the event of heavy precipitation, and determined that barriers required to mitigate the flood were in place and operable. Additionally, the inspectors performed a walkdown of the protected area to identify any modification to the site which would inhibit site drainage during a probable maximum precipitation event or allow water ingress past a barrier. The inspectors also walked down underground bunkers/manholes subject to flooding that contained multiple train or multiple function risk-significant cables. The inspectors also reviewed the abnormal operating procedure (AOP) for mitigating the design basis flood to ensure it could be implemented as written. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one external flooding sample as defined in IP 71111.01–05.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

- 1R and 2R Transformers;
- Reactor Coolant System (RCS);
- 11 EDG during 12 EDG maintenance; and
- Division II Residual Heat Removal (RHR)/Service Water.

The inspectors selected these systems based on their risk significance relative to the Reactor Safety Cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, USAR, technical specification (TS) requirements, outstanding work orders (WOs), condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

These activities constituted four partial system walkdown samples as defined in IP 71111.04–05.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

The inspectors conducted fire protection walkdowns which were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- Fire Zone 14-B: Isophase Bus Area;
- Fire Zone 12-E: Steam Jet Air Ejector Room;
- Fire Zone 12-C: Condenser Area;
- Fire Zone 2-F: Main Steam Chase; and

- Fire Zone 33: Emergency Filtration Train (EFT) Building Third Floor.

The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and implemented adequate compensatory measures for out-of-service, degraded or inoperable fire protection equipment, systems, or features in accordance with the licensee's fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events with later additional insights, their potential to impact equipment which could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Using the documents listed in the Attachment to this report, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's CAP. Documents reviewed are listed in the Attachment to this report.

These activities constituted five quarterly fire protection inspection samples as defined in IP 71111.05–05.

b. Findings

1) Failure to Maintain Portable Fire Extinguishers in Accordance with Fire Strategy

Introduction: The inspectors identified a finding of very low safety significance and a non-cited violation (NCV) of TS 5.4.1.d when the licensee failed to implement procedures associated with Fire Protection Program Implementation to ensure that portable fire extinguishers were maintained in accordance with the fire strategy. Specifically, on May 1, 2015, the licensee failed to implement fire protection plan procedures when they failed to control three portable fire extinguishers in the condenser room, a room housing safe shutdown cabling, in accordance with Fire Strategy A.3-12-C. In this case, inspectors found that of the four dry chemical extinguishers required to be stationed in the condenser room, two indicated that they were partially depleted and needed to be recharged, and a third extinguisher was missing entirely.

Description: On May 1, 2015, the inspectors performed a fire protection walk-down of the condenser room fire zone. This was normally a posted high radiation area, but it had been down-posted due to the fact that the plant entered a RFO on April 12, 2015. The inspectors walked down the condenser basement area as well as the second floor (the mezzanine level). As part of the walk-down, the inspectors reviewed Fire Strategy A.3-12-C, which was the fire strategy procedure for the condenser room fire zone. This procedure informs the Fire Brigade on the location of available local equipment for fire mitigation if a fire were to occur in the condenser room. During the inspectors' walk-down, several issues were identified. The most notable of these issues included the fact that of the four dry chemical extinguishers required by Fire Strategy A.3-12-C to be stationed and available in the condenser room, two indicated that they were partially depleted and needed to be recharged, and a third extinguisher was missing entirely.

There were also indications that the fire equipment stationed in the area had not been inspected since 2013 (hose stations) and 2014 (portable fire extinguishers), which occurred during previous outages and potentially meant that the equipment had not yet been inspected during the current RFO despite the fact that three weeks had elapsed into a planned month long outage. The inspectors engaged the licensee on these issues, and the licensee initiated a CAP to document these and other issues. The licensee also initiated corrective actions to recharge the partially depleted extinguishers and procure a portable extinguisher to replace the missing one.

While the licensee took action to correct the conditions, the inspectors also noted that until prompted with NRC questions following corrective actions for the issue, the licensee failed to identify that the fourth missing fire extinguisher was not listed in the licensee's database for inspection and hadn't been included on the database for several years. As a result, the extinguisher found previously missing which was replaced, had not been added to the database and could be vulnerable to missing required future inspections. The licensee initiated action to investigate the discrepancy and determine compliance with the fire protection code for the extinguisher. The licensee's evaluation determined that the missing fire extinguisher was required to be in place in accordance with National Fire Protection Association (NFPA) fire protection requirements.

Analysis: The inspectors determined that the failure to implement the fire strategy procedure to ensure that condenser room portable fire extinguishers were maintained was a performance deficiency because it represented a failure to meet TS 5.4.1.d, the cause was reasonably within the licensee's ability to foresee/correct, and it should have been prevented. The inspectors evaluated the finding in accordance with Inspection Manual Chapter (IMC) 0612 Appendix B and determined the issue was more than minor because it was associated with the Mitigating Systems Cornerstone attribute of Protection Against External Factors—including fire, and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the finding degraded the fire strategy for the condenser room, which required portable fire extinguishers to be maintained at designated locations in order to be available to assist with mitigation of initiating events and protection of safe shutdown equipment. In this case, the safe shutdown equipment included cabling associated with the Division II Suppression Pool Temperature monitoring instrumentation.

The inspectors assessed the significance of this finding using IMC 0609, Attachment 4, "Initial Characterization of Findings," and determined that the findings should be evaluated using 0609 Appendix IMC 0609, Appendix G, "Shutdown Operations SDP," due to the fact that the plant was in an outage at the time of the finding. The inspectors assessed the finding under the Mitigating Systems Cornerstone using Exhibit 3. The inspector assessed whether the finding met either of the criteria, "1. There was no degraded fire barrier and the fire scenario did not require the use of water to extinguish the fire," OR "2. The missing fire extinguisher or fire hose was missing for a short time and other extinguishers or hose stations were in the vicinity." While the inspectors answered No to Criterion 2, because the fire extinguisher was missing for a period of years, the inspectors were able to answer Yes to Criterion 1, and as a result they determined that the finding had very low safety significance (Green). The inspectors also noted that there were fire hoses stationed in the area, and the room was equipped with an automatic fire suppression system, which further supported the significance determination. The inspectors also consulted IMC 0609, Appendix F, Fire Protection

SDP since the finding also existed while the reactor was at power, and determined that the issue had very low safety significance because the finding was associated with portable fire extinguishers not used for hot work fire watches and it was associated with pre-fire plans. 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, Identification aspect because of the failure to implement a CAP with a low threshold for identifying issues and failure to ensure that individuals identify issues completely, accurately, and in a timely manner in accordance with the program [P.1].

Enforcement: TS 5.4.1 stated, in part, "Written procedures shall be established, implemented, and maintained covering the following activities: (d) Fire Protection Program Implementation." Strategy A.3-12-C, the Fire Strategy for the Condenser Area Fire Zone 12-C, included specific locations where local fire equipment, including portable fire extinguishers, was required to be stationed. Specifically, A.3-12-C informed the Fire Brigade on the location of available local equipment for fire mitigation and states, "Reels and portable extinguishers located in area (Ref. map)." The maps (FP12C-2 and FP12C-1) for the Condenser Room included three available dry extinguishers at specific locations on the condenser basement level, and one available dry extinguisher at the condenser mezzanine level.

Contrary to the above, on May 1, 2015, the licensee failed to implement procedures associated with Fire Protection Program Implementation to ensure that portable fire extinguishers were maintained to mitigate potential fires in accordance with the fire strategy. Specifically, the licensee failed to implement fire protection plan procedures when they failed to control three portable fire extinguishers in the condenser room, a room housing safe shutdown cabling, in accordance with Fire Strategy A.3-12-C. In this case, inspectors found that of the four dry chemical extinguishers required to be stationed in the condenser room, two indicated that they were partially depleted and needed to be recharged, and a third extinguisher was missing entirely. Because this violation was of very low safety significance and it was entered into the CAP (CAP 1477246), this issue is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy **(NCV 05000263/2015002–01: Failure to Maintain Portable Fire Extinguishers in Accordance with Fire Strategy)**. Immediate corrective actions included recharging the partially depleted extinguishers and procuring a portable extinguisher to replace the missing one.

1R06 Flood Protection Measures (71111.06)

.1 Underground Vaults

a. Inspection Scope

The inspectors selected underground bunkers/manholes subject to flooding that contained cables whose failure could disable risk-significant equipment. The inspectors determined that the cables were not submerged, that splices were intact, and that appropriate cable support structures were in place. In those areas where dewatering devices were used, such as a sump pump, the device was operable and level alarm circuits were set appropriately to ensure that the cables would not be submerged. In those areas without dewatering devices, the inspectors verified that drainage of the area was available, or that the cables were qualified for submergence conditions. The

inspectors also reviewed the licensee's corrective action documents with respect to past submerged cable issues identified in the CAP to verify the adequacy of the corrective actions. The inspectors performed a walkdown of the following underground bunkers/manholes subject to flooding:

- 2MH04 (RHR, CS, SRV & ASDS);
- CP102 (1AR feeder and control); and
- NMH305 (OG dilution fans, MCC 115 & 124 feeders, Substation A feeder).

Specific documents reviewed during this inspection are listed in the Attachment to this report. This inspection constituted one underground vaults sample as defined in IP 71111.06-05.

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08G)

From March 20, 2015 through April 24, 2015, the inspector conducted a review of the implementation of the licensee's inservice inspection (ISI) program for monitoring degradation of the RCS, risk-significant piping and components, and containment systems.

The inservice inspections described in Sections 1R08.1 and 1R08.5 below constituted one inspection sample as defined in IP 71111.08-05.

.1 Piping Systems Inservice Inspection

a. Inspection Scope

The inspectors either observed or reviewed the following Non-Destructive Examinations mandated by the American Society of Mechanical Engineers (ASME) Section XI Code to evaluate compliance with the ASME Code Section XI and Section V requirements, and if any indications and defects were detected, to determine if these were dispositioned in accordance with the ASME Code or an NRC-approved alternative requirement:

- Ultrasonic Examination of core spray B, bent pipe-to-bent pipe, W-18;
- Ultrasonic Examination of main steam A, B, C, D, pipe-to-valve welds, W-32 thru W-35;
- Visual Examination (VT) - 3, examination of recirculation manifold B, H-6 and H-10;
- VT-3 examination of main-steam C snubber/clamp, H-3;
- VT-3 examination of feedwater snubber/clamp, H-5;
- VT-3 examination of main-steam condensate leakoff linear support, H-3; and
- VT-3 examination of head vent hanger, H-2.

The inspectors reviewed the following examination completed during the previous outage with relevant/recordable conditions/indications accepted for continued service to determine whether acceptance was in accordance with the ASME Code Section XI, or an NRC-approved alternative.

The inspector reviewed records for the following pressure boundary weld repairs completed for risk-significant systems during the last outage to determine if the licensee applied the pre-service Non-Destructive Examinations and acceptance criteria required by the Construction Code, and/or the NRC-approved Code relief request. Additionally, the inspectors reviewed the welding procedure specifications (WPSs) and supporting weld procedure qualification records to determine whether the weld procedures were qualified in accordance with the requirements of the Construction Code and the ASME Code, Section IX.

- Install piping and valve on P200B Service Water (WO 496046-19);
- Install seal water vent line assembly pipe and cap, 12 Rx recirculation pump (WO 496046-22); and
- Install diesel generator fuel oil separation modification supports (WO 505386-07).

b. Findings

1) Failure to Measure Interpass Temperature

Introduction: The inspectors identified a finding of very low safety significance and NCV of Title 10 CFR Part 50, Appendix B, Criterion IX, "Control of Special Processes," for a failure to measure the interpass temperature while performing welding on the emergency diesel generator (EDG) fuel oil modification supports. Consequently, welding was performed without the Code and Procedure required interpass temperature being monitored, a parameter which can potentially affect the mechanical properties of materials being welded.

Description: The inspectors observed that a welder had failed to measure the interpass temperature while performing shielded metal arc welding on EDG fuel oil piping supports. The inspectors also noted that there were no temperature-measuring devices readily available in the area.

The welder was to perform the welding activities in accordance with WPS FP-PE-pAWS-I-II-SM-001, which specified an interpass temperature to ensure that temperature was not exceeded on the workpiece between passes. Furthermore, Procedure 4 AWI-07.02.03; "Fleet Welding Program Site-Specific Requirements," used in conjunction with the WPS, required, in part, that, "verification of preheat and interpass temperature SHALL be verified."

Multiple passes had already been performed on a number of welds as part of the modification of the EDG fuel oil train separation before the inspectors observed the in-process welding and noted the failure to measure the interpass temperature. The inspectors were concerned that failing to follow procedures as required by the code and procedures, could impact the quality of the welds and lead to susceptible material failing while in service, and thereby adversely affect the integrity of the associated systems. As a result of the inspectors' concern, the welder measured the interpass temperatures on the balance of the welding and verified that the interpass temperatures did not exceed that allowed by procedure. Since the measured interpass temperatures were well below that permitted by procedure, the inspectors concluded that there was reasonable assurance that the previous weld passes would not have exceeded the interpass temperature. The issue was entered into the licensee's CAP (CAP 1475767).

Analysis: The inspectors determined that the failure to measure the weld interpass temperature as required by the ASME Code and site procedure was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because the inspectors answered "yes" to the more than minor question, "If left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC intervention, the welder would have completed all of the welds without having measured the interpass temperature, a welding parameter which can affect the mechanical properties (e.g., impact properties) of some materials being welded, and if not corrected, could lead to a potential failure of welds in service.

In accordance with Table 2, "Cornerstones Affected by Degraded Condition or Programmatic Weakness," of IMC 0609, Attachment 4, "Initial Characterization of Findings," issued June 19, 2012, the inspectors checked the box under the Mitigating Systems Cornerstone because leakage on the EDG fuel oil system could cause core decay heat removal to be degraded. The inspectors determined this finding was of very-low safety significance (Green) based on answering "yes" to the questions in Part A of Exhibit 2, "Mitigating Systems Screening Questions," in IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power," issued on June 19, 2012. Specifically, the inspectors answered "yes" to the screening question "If the finding is a deficiency affecting the design or qualification of a mitigating structure, system, and component (SSC), does the SSC maintain its operability or functionality?" The welder subsequently performed interpass temperature measurements and demonstrated that the temperature would remain below the required temperature of the welds in question, and the issue did not result in the actual loss of the operability or functionality of a safety system.

The inspectors determined that the primary cause of the failure to measure the interpass temperature while performing a manual welding process was related to the cross-cutting component of Problem Identification and Resolution, "Operating Experience" [P.5]. Both the licensee and the vendor performing the welding were aware of similar findings recently identified at other plants, but failed to exercise that operating experience while welding the EDG fuel oil piping. Specifically, the organization failed to effectively implement external operating experience in a timely manner.

Enforcement: The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion IX, for a welder's failure to measure interpass temperature while performing welding on the EDG fuel oil piping supports contrary to procedures.

Title 10 CFR Part 50, Appendix B, Criterion IX, "Control of Special Processes", states that "Measures shall be established to assure that special processes, including welding, heat treating, and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements."

The WPS FP-PE-pAWS-I-II-SM-001, used to perform welding on EDG fuel oil piping support welds, includes an interpass temperature.

Welding procedure 4 AWI-07.02.03; "Fleet Welding Program Site-Specific Requirements," used in conjunction with the WPS, required, in part, that "verification of preheat and interpass temperature SHALL be verified."

Contrary to the above, while performing welding on the EDG fuel oil piping supports welds, the welder did not accomplish the welding in accordance with the WPS in that the welder failed to measure the interpass temperature. After identification by the inspectors, the welder proceeded to measure the interpass temperature on the balance of the weld passes, thereby demonstrating that interpass temperatures had not been exceeded.

Because of the very-low safety significance, and because the licensee entered this issue into its CAP (CAP 1475767), it is being treated as an NCV consistent with Section 2.3.2 of the Enforcement Policy **(NCV 05000263/2015002-02: Failure to Measure Interpass Temperature)**.

.2 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI-related problems entered into the licensee's CAP, and conducted interviews with licensee staff to determine if:

- the licensee had established an appropriate threshold for identifying ISI-related problems;
- the licensee had performed a root cause (if applicable) and taken appropriate corrective actions; and
- the licensee had evaluated operating experience and industry generic issues related to ISI and pressure boundary integrity.

The inspectors performed these reviews to evaluate compliance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the Attachment to this report.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review of Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

On June 15, 2015, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification training to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;

- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of abnormal and emergency procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly licensed operator requalification program simulator sample as defined in IP 71111.11–05.

b. Findings

No findings were identified.

.2 Resident Inspector Quarterly Observation During Periods of Heightened Activity or Risk (71111.11Q)

a. Inspection Scope

On April 12, 2015, the inspectors observed reactor shutdown activities associated with RFO27. This was an activity that required heightened awareness or was related to increased risk. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of procedures;
- control board manipulations; and
- oversight and direction from supervisors.

The performance in these areas was compared to pre-established operator action expectations, procedural compliance and task completion requirements. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly licensed operator heightened activity/risk sample as defined in IP 71111.11–05.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk-significant systems:

- Plant Communications System;
- Fire System; and
- RHR System.

The inspectors reviewed events such as where ineffective equipment maintenance had resulted in valid or invalid automatic actuations of engineered safeguards systems and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- implementing appropriate work practices;
- identifying and addressing common cause failures;
- scoping of systems in accordance with 10 CFR 50.65(b) of the maintenance rule;
- characterizing system reliability issues for performance;
- charging unavailability for performance;
- trending key parameters for condition monitoring;
- ensuring 10 CFR 50.65(a)(1) or (a)(2) classification or re-classification; and
- verifying appropriate performance criteria for structures, systems, and components (SSCs)/functions classified as (a)(2), or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

This inspection constituted three quarterly maintenance effectiveness samples as defined in IP 71111.12–05.

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

.1 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- Drywell (DW) Head Lift during Reactor Disassembly;
- Reactor Cavity Flood Up in Preparation for Refueling;
- Recirculation System modification OPDRV;
- Reactor cavity drain down from flooded up to RPV flange;
- 11 Recirculation Motor-Generator Vibrations; and
- Power increase from 1953MWt to 2004MWt (100% power) first time evolution.

These activities were selected based on their potential risk significance relative to the Reactor Safety Cornerstones. As applicable for each activity, the inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4) and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met. Documents reviewed during this inspection are listed in the Attachment to this report.

These maintenance risk assessments and emergent work control activities constituted six samples as defined in IP 71111.13–05.

b. <u>Findings</u>

1) <u>Failure to Maintain Secondary Containment and Standby Gas Treatment System Operable During OPDRV Activities</u>

<u>Introduction</u>: The inspectors identified a finding of very low safety significance and an associated NCV of TS 3.6.4.1, Secondary Containment and TS 3.6.4.3, SBGT System because the licensee did not maintain secondary containment and the SBGT system operable as required during activities considered operations with the potential to drain the reactor vessel (OPDRVs). Specifically, on April 14, 2015, and again on May 13, 2015, the licensee failed to classify activities associated with draining reactor inventory as OPDRVs while relying on an automatic isolation function for the drain path, and as a result failed to maintain required equipment operable during these activities.

<u>Description</u>: On April 14 and May 13, 2015, the licensee performed actions to drain reactor vessel inventory as part of outage related activities. On April 14, 2015, the licensee drained approximately 32 inches of vessel inventory, by using the reactor water cleanup system (RWCU), which takes suction from the vessel at locations below the reactor, to dump inventory to tanks and other locations external to the reactor pressure vessel (RPV). On May 13, 2015, the licensee drained RPV inventory from being fully flooded up in the refueling cavity to the reactor vessel flange. This was a planned activity where operators drained approximately 22 feet of vessel inventory, relying in part on RWCU, dumping at a rate of approximately 100 gallons per minute. In both cases, the licensee did not classify the activities as OPDRVs, due to the fact that the TS automatic isolation function for RWCU was maintained operable, and would initiate to mitigate the loss of inventory (LOI), if required.

On May 13, 2015, the inspectors observed that the licensee had performed the draining activities that morning and challenged the licensee’s failure to control the activities as

OPDRVs, and maintain the required equipment operable for that mode of applicability. An OPDRV is described in the licensee's TS as an operation with a potential for draining the reactor vessel. The licensee also adopted a definition of an OPDRV in procedure 2208 "OPDRV Checklist" as provided in EGM 11-003 as any activity that could potentially result in draining or siphoning the RPV water level below the top of the fuel, without taking credit for mitigating measures. The inspectors observed that the May 13, 2015 activities appeared to meet the definition of OPDRV, and challenged whether the site was attempting to credit mitigating actions in order to avoid classifying the activity as an OPDRV.

During discussions with the licensee, operators noted that procedure OWI-03.03, Operation with the Potential to Drain the Vessel included the same EGM 11-003 OPDRV definition that was provided in procedure 2208. However, they also pointed to procedure allowances in OWI-03.03 which stated, "Use of the RWCU system for cleanup and inventory control activities per the design is not considered an OPDRV." This procedure also stated, "Plant system operation in accordance with its design and TS is not considered an OPDRV (e.g., operation of SDC or other closed loop systems that remove water from the reactor/cavity) provided the TS required automatic isolation features for the current mode are operable." The inspectors noted that this information provided inaccurate and misleading guidance that allowed for confusion on the part of operators when classifying the drain down activities. Separately, the inspectors noted that this same procedure provided appropriate guidance that should have been observed. Specifically, the procedure stated, "cavity drain down (to the top of the flange using systems that take suction above the top of active fuel) would not typically be considered an OPDRV." The inspectors noted that this portion of the procedure only permitted excluding drain down activities with suction taken above the fuel from OPDRV classification, and did not exempt activities with suction below the fuel from OPDRV classification.

Following a review of the extent of condition, the inspectors identified the April 14, 2015, example of the performance deficiency. On April 14, 2015, SBGT and secondary containment were inoperable during the draining activities. On May 13, 2015, both SBGT and secondary containment were inoperable during draining of the RPV from the flooded up condition to the RPV flange. This resulted in the failure to maintain secondary containment and SBGT operable during the OPDRV mode of applicability. TS Limiting Condition for Operation (LCO) 3.0.2 stated, "Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met." The inspectors noted that upon entry into the OPDRV mode of applicability, the licensee was required to declare the LCOs for the inoperable SSCs not met, and was required to take the required actions. Entering TS 3.6.4.1 and TS 3.6.4.3 would have both required that the OPDRVs be suspended immediately.

<u>Analysis</u>: The inspectors determined that the failure to maintain secondary containment and SBGT operable while an OPDRV was in progress was a performance deficiency because it represented a failure to meet TS requirements, the cause was reasonably within the licensee's ability to foresee/correct, and it should have been prevented. The performance deficiency was more than minor because it was associated with the configuration control attribute of the Barrier Integrity Cornerstone, and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, RCS, and containment) protect the public from radionuclide releases

caused by accidents or events. Specifically, the secondary containment boundary and the SBGT system were not maintained operable during an OPDRV activity.

The inspectors evaluated the finding using IMC 0609, Attachment 4, Phase 1 – Initial Screening and Characterization of Findings, which required an analysis using IMC 0609 Appendix G, the Shutdown Operations SDP since the reactor was shut down. The finding was assessed in accordance with IMC 0609 Appendix G, Attachment 1, Exhibit 4 for the Barrier Integrity cornerstone. Because primary containment was deinerted and open to secondary containment at the time of the finding, the inspectors conservatively evaluated the finding under IMC 0609 Appendix H for containment integrity findings. Using appendix H, the inspectors determined that the finding initially occurred during plant operational state (POS) 1, or in cold shutdown (Mode 4) with the RCS intact and RHR in service. This occurred before the reactor cavity was flooded up for refueling, and within eight days of the start of the outage. Using Table 6.3, the inspectors determined that this configuration had very low safety significance because containment was deinerted. The inspectors determined that the performance deficiency occurred again in POS 2, after refueling had been completed, and did not occur within eight days of the start of the outage, which was a configuration having very low safety significance in accordance with Section 6.2 of Appendix H, due to the low decay heat load associated with this configuration. Additionally, the inspectors determined that the licensee maintained adequate mitigation capability for reactor vessel water level inventory and an event did not occur that could be characterized as a loss of control. As a result of these considerations, the finding screened as having very low safety significance (green). 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, Documentation aspect because of the failure of the licensee to create and maintain complete, accurate and up-to-date documentation [H.7]. Specifically, procedure OWI-03.03 for OPDRVs contained inaccurate and misleading guidance that allowed for confusion on the part of operators when classifying the drain down activities.

Enforcement: TS 3.6.4.1, Secondary Containment, required in part, "secondary containment shall be operable during modes 1, 2, 3, during movement of recently irradiated fuel assemblies in the secondary containment and during OPDRVs." TS 3.6.4.3, Standby Gas Treatment System, required in part, "Two trains of SBGT shall be operable during modes 1, 2, 3, during movement of recently irradiated fuel assemblies in the secondary containment and during OPDRVs." With secondary containment inoperable, TS 3.6.4.1 required that the OPDRVs be suspended immediately. With both SBGT trains inoperable, Tech Spec 3.6.4.3 required that the OPDRVs be suspended immediately.

Contrary to the above, on April 14 and May 13, 2015, the licensee failed to maintain secondary containment and the SBGT system operable as required during activities considered OPDRVs. Specifically, the licensee failed to classify activities associated with draining reactor inventory as OPDRVs while relying on an automatic isolation function for the drain path, and as a result failed to maintain required equipment operable during these activities, or take action to immediately suspend the OPDRVs. Once questioned by the inspectors, the licensee took action to control other outage related draining activities as OPDRVs and placed this issue into its CAP (CAP 1479284). Because the licensee entered the issue into its CAP and the finding is of very low safety significance (Green), this violation is being treated as an NCV, consistent with

Section 2.3.2 of the NRC's Enforcement Policy **(NCV 05000263/2015002-03: Failure to Maintain Secondary Containment and Standby Gas Treatment System Operable During OPDRV Activities)**.

The inspectors determined that this OPDRV was not eligible for enforcement discretion in accordance with EGM 11-003 criterion 2(a), because at the time of the OPDRV evolution, the reactor cavity was not in a flooded condition where the reactor vessel water level was maintained at least 21 feet and 11 inches over the top of the RPV flange as required by TS 3.9.6. As a result of not meeting this criterion of the EGM, the inspectors did not recommend enforcement discretion for this violation.

2) Failure to Fill the Reactor Cavity in Accordance with Refueling Preparation Procedure

Introduction: The inspectors identified a finding of very low safety significance and an associated NCV of TS 5.4.1, "Procedures," on April 15, 2015, when the licensee failed to implement procedure 9001, "Reactor Well & Dryer-Separator Storage Pool Filling Procedure," for refueling preparation activities. Specifically, when faced with indications that the CSTs did not contain enough water inventory to complete outage critical path RPV flooding activities, the licensee failed to implement 9001 procedure steps for using prescribed equipment and methods to fill the reactor cavity. With the proceduralized methods unavailable, operators used the site decision-making process to utilize demineralizer water hoses to fill the cavity rather than processing required 9001 procedure changes.

Description: On April 16, 2015, while gathering plant status, the inspectors discovered that the licensee had been challenged with water inventory issues during activities associated with flooding the reactor cavity in preparation for refueling the reactor vessel. Specifically, the inspectors noted that the control room had received indications that they had exhausted the water inventory in the condensate storage tanks (CSTs) during the flood up activities. This presented a challenge because this was the source of clean, quality water that the licensee's reactor cavity filling procedures relied upon. In discussions with the control room, the inspectors noted that rather than wait to process more water to replace the CST inventory, operators chose to use demineralizer water hoses available on the refuel floor to continue the flood up. The inspectors questioned whether this was a method prescribed by the procedure for filling the reactor well in preparation for refueling, and learned that it was not. The inspectors also questioned whether the procedure change process was utilized to evaluate and incorporate this method into the procedure, and learned that it was not.

The inspectors reviewed Procedure 9001, "Reactor Well & Dryer-Separator Storage Pool Filling Procedure." The inspectors noted that there were specific methods prescribed by the procedure for filling the reactor cavity during the process for flooding up in preparation for refueling. The inspectors found that Step 11 stated, "Fill the reactor well and dryer separator storage pool by performing one or more of the following:" Step 11 then provided detailed instructions for use of the condensate/feedwater system, the 11 core spray system, and the 12 core spray system. When flood up level begins to approach the required maximum level, Step 25 stated, "IF continued filling reactor well and dryer-separator storage pool is needed, THEN one or both of the following may be used:" Step 25 provided instructions on use of a CST pressurizing station via core spray, or the control rod drive (CRD) system. The inspectors determined that instructions for use of the demineralizer water hoses were not included in this procedure. The

inspectors concluded that use of the demineralizer water hoses to fill the reactor cavity in support of outage critical path was contrary to procedure 9001.

The inspectors reviewed the station logs and noted an entry on April 15, 2015, which stated:

> "Per FP-PA-HU-05 (Decision making) attachments 4 and 5, the tier II decision making process was implemented to determine required actions to add water to the reactor cavity using demineralizer water hoses on the refuel floor as part of flood up action in 9001 Rx Well Dryer-Separator Storage Pool Filling Procedure. Adding water with the demineralizer system is not currently a method listed in the 9001 procedure. The CST tank levels were approaching level at which injection with the condensate service system was in jeopardy due to low level trips. The CRD pump suction pressure had lowered to 0 psig and was in jeopardy of tripping.

This decision was based on the following:

> 1. Demineralized water is considered a high quality source of water;
> 2. Chemistry supervision concurred with adding demineralized water directly; and
> 3. Adding water would support critical path outage work.
>
> The Shift Manager (SM), Control Room Supervisor, Shift Technical Advisor, and duty operators were involved in the decision making process. Engineering was informed. Tier II was determined to be appropriate because personnel are dedicated to the demineralizer water hoses and are in direct contact with the control room."

The inspectors noted that it appeared that the operations crew had utilized the decision-making process to decide to perform activities contrary to the prescribed methods in the 9001 procedure. The 9001 procedure was the procedure designated to precisely and directly control the reactor cavity filling process in preparation for refueling activities. The inspectors discussed this decision making with operations management and staff. In discussions with one of the individuals involved in the decision making, it was noted that an additional reason for wanting to expeditiously fill the reactor cavity, aside from the outage critical path reason, was the fact that the spent fuel pool cooling (FPC) system had been removed from service to support the flood up process. As a result, operators wanted to minimize time in this configuration. The inspectors noted that while a valid concern, the fuel pool time to boil was long (several days) and there were several options the operations crew could have chosen to avoid actions contrary to the 9001 procedure, including restoring FPC to service while new water was processed into the CST, or initiating the procedure change process to evaluate including the demineralizer hose method for filling the cavity.

<u>Analysis</u>: The inspectors determined that the failure to fill the reactor cavity in accordance with the 9001 reactor well filling procedure was a performance deficiency because it represented a failure to meet TS requirement 5.4.1; the cause was reasonably within the licensee's ability to foresee and correct; and should have been prevented. The inspectors evaluated IMC 0612, Appendix E, and did not find any similar examples of minor issues. The inspectors determined that the finding was more than

minor in accordance with IMC 0612, Appendix B, because if left uncorrected, the performance deficiency had the potential to lead to a more significant safety concern. Specifically, the operations crew's use of the decision-making process to support outage critical path by bypassing proceduralized steps and performing activities using methods contrary to the procedure could lead to a more significant safety concern. In addition, if performed incorrectly, i.e. without flushing the hoses prior to use, the use of demineralizer water hoses could introduce foreign material into the core and challenge the integrity of the fuel cladding barrier.

The inspectors evaluated the finding using IMC 0609, Attachment 4, Phase 1 – Initial Screening and Characterization of Findings, which required an analysis using IMC 0609 Appendix G, the Shutdown Operations SDP since the reactor was in Mode 5 (refueling). The finding was assessed in accordance with IMC 0609 Appendix G, Attachment 1, Exhibit 4 for Barrier Integrity. The inspectors answered "No" to all the Barrier Integrity screening questions and determined that the finding had very low safety significance (Green). The inspectors concluded that this finding was cross-cutting in the Human Performance, Conservative Bias aspect, because of the failure of the individuals to use decision-making practices that emphasize prudent choices over those that are simply allowable, and the failure to ensure that proposed actions are determined to be safe in order to proceed, rather than unsafe in order to stop. [H.14] Specifically, operations personnel failed to follow the 9001 procedure because they used a decision making process that allowed use of demineralizer water hoses, which carried additional foreign material and coordination risks, rather than the proceduralized methods, without evaluating the change through the procedure change process.

<u>Enforcement</u>: TS 5.4.1 required that written procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide (RG) 1.33, Revision 2, Appendix A, February 1978. Section 2.k of RG 1.33, Revision 2, included Preparation for Refueling procedures. Procedure 9001, "Reactor Well & Dryer-Separator Storage Pool Filling Procedure," Step 11 stated, "Fill the reactor well and dryer separator storage pool by performing one or more of the following:" Step 11 provided detailed instructions for use of the condensate/feedwater system, the 11 core spray system, and the 12 core spray system. When flood up level begins to approach the required maximum level, Step 25 stated, "IF continued filling reactor well and dryer-separator storage pool is needed, THEN one or both of the following may be used:" Step 25 provided instructions on use of a CST pressurizing station via core spray, or the CRD system.

Contrary to these requirements, on April 15, 2015, the licensee failed to implement procedure 9001, "Reactor Well & Dryer-Separator Storage Pool Filling Procedure," for refueling preparation activities. Specifically, when faced with indications that the CSTs did not contain enough water inventory to complete outage critical path RPV flooding activities, the licensee failed to implement 9001 procedure steps for using prescribed equipment and methods to fill the reactor cavity. With the proceduralized methods unavailable, operators used the site decision-making process to utilize demineralizer water hoses to fill the cavity rather than processing required 9001 procedure changes. Because this violation was of very low safety significance and it was entered into the CAP (CAP 1474891), this issue is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy **(NCV 05000263/2015002–04: Failure to Fill the Reactor Cavity in Accordance with Refueling Preparation Procedure)**. Immediate corrective actions included action to initiate the procedure change process for 9001 and

department communication to Operations regarding the incident, emphasizing that the decision making process is not a substitute for the procedure change process.

1R15 Operability Determinations and Functional Assessments (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- Vapor Phase Corrosion Inhibitor challenges operability of Equipment Qualification required Motor Operated Valves;
- EDG Fuel Oil System support missing;
- Unplanned entry into OPDRV mode of applicability due to clearance order; and
- 11 EDG due to decreased fuel oil levels from 12 EDG maintenance run.

The inspectors selected these potential operability issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TS and USAR to the licensee's evaluations to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment to this report.

This operability inspection constituted four samples as defined in IP 71111.15–05.

b. Findings

1) Inadequate Clearance Order results in unplanned OPDRV

Introduction: A finding of very low safety significance and an associated NCV of TS 5.4.1, "Procedures," was self-revealed on May 16, 2015, when the licensee failed to implement procedure FP-OP-TAG-01, "Fleet Tagging," for equipment control activities associated with the scram discharge volume (SDV). Specifically, the licensee failed to ensure that clearance order checklist 58972-03 restored valve I-CRD-R-26, an SDV instrument vent valve, to its normal position prior to returning the SDV system to service. As a result, during subsequent RCS pressure boundary testing, RCS water leaked out onto the Reactor Building floor through the open vent line, creating an unplanned OPDRV.

Description: On May 16, 2015, while in cold shutdown for a RFO, the licensee inserted a planned reactor scram in order to support RCS pressure boundary testing. The licensee had planned to be in this configuration with a scram inserted for several days. However, when the scram was initiated, RCS water was discovered leaking from an

open and uncapped ¾ inch SDV instrument vent line. Due to the direct path from the RPV to the floor of the reactor building main level, this represented an unplanned OPDRV. Licensee procedures and EGM 11-003 define an OPDRV as any activity that could potentially result in draining or siphoning the RPV water level below the top of the fuel, without taking credit for mitigating measures.

Upon discovery of this condition, immediate action was taken to secure the leakage. Specifically, operators took action to close and cap the open vent valve, I-CRD-R-26, and reset the scram in order to terminate the loss of RCS inventory. As a result of this condition, a significant area of the 935' Reactor Building was left highly contaminated. During the event, reactor vessel level dropped a maximum of 2 inches, which was mostly attributed to the characteristics of inserting the scram, but was partially due to the leakage through the open vent line.

Licensee investigation revealed that Clearance Order checklist 58972-03 failed to restore valve I-CRD-R-26, to its normal position prior to returning the SDV system to service. Specifically, when the tags were lifted for the valve, the vent line was left open and uncapped, which should have been identified by individuals preparing and utilizing the checklist. Further investigation determined that the checklist preparer relied on a previous checklist used for a different work activity and did not thoroughly challenge the final configuration by consulting plant drawings. Discussion with operations staff also revealed that the operations crew that was restoring the SDV to service using the checklist questioned the final configuration of the vent line, noting that it should normally be left closed and capped. During crew discussions, the operators concluded that the vent line was supposed to be left open for some additional instrumentation and controls testing planned for later in the schedule. Ultimately, no action was taken to validate those assumptions, and no action was taken to transfer this abnormal status to another process for configuration control.

Analysis: The inspectors determined that the failure to adequately restore the SDV system to service in accordance with fleet tagging requirements was a performance deficiency because it represented a failure to meet TS requirement 5.4.1; the cause was reasonably within the licensee's ability to foresee and correct; and should have been prevented. The inspectors determined that the finding was more than minor in accordance with IMC 0612, Appendix B, because it adversely impacted the Initiating Events Cornerstone attributes of Configuration Control and Procedure Quality, and affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the clearance order checklist improperly returned the SDV system to service with a vent line open and uncapped, resulting in a small LOI event.

The inspectors evaluated the finding using IMC 0609, Attachment 4, Phase 1 – Initial Screening and Characterization of Findings, which required an analysis using IMC 0609, Appendix G, the Shutdown Operations SDP since the reactor was in Mode 4 (cold shutdown). The finding was assessed in accordance with IMC 0609 Appendix G, Attachment 1, Exhibit 2 for Initiating Events. For LOI initiating events, Question B.2 of Exhibit 2 asked "Did a LOI event result in a leakage such that if the leakage were undetected and/or unmitigated in 24 hours or less it would cause the currently operating decay heat removal method to fail (e.g., level would drop to the SDC isolation Level 3 set point (BWR))?" This question was answered "YES," since the RCS leakage path through the open vent line on the SDV would have impacted decay heat removal if no

action had been taken to isolate or mitigate the leakage for 24 hours. As a result, the analysis of the event transitioned to IMC 0609 Appendix G, Attachment 3, "Phase 2 Significance Determination Process Template for BWR During Shutdown" for a Phase 2 evaluation. The Senior Reactor Analyst (SRA) determined this to be a precursor to an initiating event for LOI. The POS was determined to be "POS 1," since the reactor head was on with the RCS closed and with RHR in service. The initiating event likelihood (IEL) for LOI using Table 2, "Initiating Event Likelihoods (IELs) for LOI precursors" was determined to be "4." The RHR system remained operating during this event. The time to RCS boil given a loss of decay heat removal was on the order of 6.2 hours. Both trains of RHR and the CRD pumps were available during the event.

Using IMC 0609 Appendix G, Attachment 3, Worksheet 1, the SRA evaluated the equipment mitigative capability and operator recovery credit. The combined sequences from Worksheet 1 had a risk significance of less than 1E-6/yr. Therefore, the SRA determined that this issue was best characterized as a finding of very low safety significance (Green).

The inspectors concluded that this finding was cross-cutting in the Human Performance, Challenge the Unknown aspect because of the failure of individuals to stop when faced with uncertain conditions, and the failure to ensure that risks are evaluated and managed before proceeding [H.11]. Specifically, operators raised questions about the abnormal configuration during system restoration, but failed to stop and adequately assess the uncertain conditions, and instead proceeded with restoration activities based on assumptions.

Enforcement: TS 5.4.1 requires that written procedures be established, implemented, and maintained covering the applicable procedures recommended in RG 1.33, Revision 2, Appendix A, February 1978. Section 1.c of RG 1.33, Revision 2, includes Equipment Control (e.g., locking and tagging) procedures. FP–OP–TAG–01, "Fleet Tagging" states, under the section for Tag Removal, "Operations Supervision SHALL…Verify the tag sequencing of the Lift C/L is correct. Transfer status of components not restored to the normal position according to site specific procedures to another process for configuration control." Under the section for review and approval of a clearance order checklist, the procedure states, "The Operations approver SHALL…verify adverse plant impacts caused by the tag activity are identified and contingency actions are established when necessary."

Contrary to these requirements, on May 16, 2015, the licensee failed to implement procedure FP-OP-TAG-01, "Fleet Tagging," for equipment control activities associated with the SDV. Specifically, the licensee failed to ensure that Clearance Order checklist 58972-03 restored valve I-CRD-R-26, an SDV instrument vent valve, to its normal position prior to returning the SDV system to service, and alternatively failed to transfer the abnormal configuration to another process for configuration control. Additionally, Operations personnel failed to identify the adverse plant impacts associated with leaving the vent line open and uncapped when restoring the SDV system to service. As a result, during subsequent RCS pressure boundary testing, RCS water leaked out onto the Reactor Building floor through the open vent line, creating an unplanned OPDRV. Because this violation was of very low safety significance and it was entered into the CAP (CAP 1479307), this issue is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy **(NCV 05000263/2015002-05: Inadequate Clearance Order Results in unplanned OPDRV).** Immediate corrective actions

included termination of the leakage by closing and capping the SDV vent line and resetting the scram. The site initiated an ACE, which was in progress at the end of the inspection period.

The inspectors determined that this OPDRV was not eligible for enforcement discretion in accordance with EGM 11-003 because at the time of the OPDRV evolution, the reactor was in Mode 4, Cold Shutdown. The inspectors noted that EGM discretion criteria could only be applied while the plant was in mode 5, Refueling. As a result of not meeting this requirement of the EGM, the inspectors did not recommend enforcement discretion for this violation.

1R18 Plant Modifications (71111.18)

a. Inspection Scope

The inspectors reviewed the following modification(s):

- Refuel Floor Loading Reanalysis – RFO27 (Temporary Modification); and
- EDG Fuel Oil Separation (Permanent Modification).

The inspectors reviewed the configuration changes and associated 10 CFR 50.59 safety evaluation screening against the design basis, the USAR, and the TS, as applicable, to verify that the modification did not affect the operability or availability of the affected system(s). The inspectors, as applicable, observed ongoing and completed work activities to ensure that the modifications were installed as directed and consistent with the design control documents; the modifications operated as expected; post-modification testing adequately demonstrated continued system operability, availability, and reliability; and that operation of the modifications did not impact the operability of any interfacing systems. As applicable, the inspectors verified that relevant procedure, design, and licensing documents were properly updated. Lastly, the inspectors discussed the plant modification with operations, engineering, and training personnel to ensure that the individuals were aware of how the operation with the plant modification in place could impact overall plant performance. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one temporary modification sample and one permanent plant modification sample as defined in IP 71111.18–05.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the following post-maintenance (PM) activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- RWCU plant protection system relay replacement;
- CV-1729 Major Overhaul of Valve and Actuator;
- Transversing Incore Probe Channel 3 Isolation Capability Issue;
- Reactor Coolant Pressure Boundary Leakage Test;
- 12 EDG Post Maintenance Test;
- DW to Torus Vacuum Breakers; and
- DW Closeout.

These activities were selected based upon the SSC's ability to impact risk. The inspectors evaluated these activities for the following (as applicable): the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was returned to its operational status following testing (temporary modifications or jumpers required for test performance were properly removed after test completion); and test documentation was properly evaluated. The inspectors evaluated the activities against TSs, the USAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with post-maintenance tests to determine whether the licensee was identifying problems and entering them in the CAP and that the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the Attachment to this report.

This inspection constituted seven post-maintenance testing samples as defined in IP 71111.19–05.

b. Findings

No findings were identified.

1R20 Outage Activities (71111.20)

.1 Refueling Outage Activities

a. Inspection Scope

The inspectors reviewed the Outage Safety Plan (OSP) and contingency plans for the Unit 1 RFO, conducted April 12, 2015 through May 30, 2015, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the RFO, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below:

- licensee configuration management, including maintenance of defense-in-depth commensurate with the OSP for key safety functions and compliance with the applicable TS when taking equipment out of service;

- implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing;
- installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error;
- controls over the status and configuration of electrical systems to ensure that TS and OSP requirements were met, and controls over switchyard activities;
- monitoring of decay heat removal processes, systems, and components;
- controls to ensure that outage work was not impacting the ability of the operators to operate the spent FPC system;
- reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss;
- controls over activities that could affect reactivity;
- maintenance of secondary containment as required by TS;
- licensee fatigue management, as required by 10 CFR 26, Subpart I;
- refueling activities, including fuel handling and sipping to detect fuel assembly leakage;
- startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the DW (primary containment) to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing; and
- licensee identification and resolution of problems related to RFO activities.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted one RFO sample as defined in IP 71111.20–05.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- Reactor Building to Torus Vacuum Breaker Mechanical Exercise Test (Routine);
- Drywell-Torus Monthly Vacuum Breaker Check (IST);
- Primary Containment Purge and Vacuum Breaker, Pressure and Isolation Valve LLRT Test (CIV);
- OSP-ECC-0566: Low Pressure Emergency Core Cooling System (ECCS) Automatic Initiation and Loss of Auxiliary Power Test (Routine);
- 0255-20-ID-1: Excess Flow Check Valve Test (Routine);
- 0255-20-IIC-1: Reactor Coolant Pressure Boundary Leakage Test (RCS Leakage);

- 1056: Reactor Core Isolation Cooling (RCIC) Turbine Overspeed Trip Test (Routine);
- 0163: Stack Wide Range Gas Monitor Calibration (Routine);
- 0054-B: MSL Low Pressure 6P1 Isolation Insufficient, Test & Calibration (Routine); and
- 0127: Drywell to Torus Vacuum Breaker 24 Month Surveillance (Routine).

The inspectors observed in-plant activities and reviewed procedures and associated records to determine the following:

- did preconditioning occur;
- the effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis;
- plant equipment calibration was correct, accurate, and properly documented;
- as-left setpoints were within required ranges; and the calibration frequency was in accordance with TSs, the USAR, procedures, and applicable commitments;
- measuring and test equipment calibration was current;
- test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied;
- test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used;
- test data and results were accurate, complete, within limits, and valid;
- test equipment was removed after testing;
- where applicable for inservice testing activities, testing was performed in accordance with the applicable version of Section XI, American Society of Mechanical Engineers code, and reference values were consistent with the system design basis;
- where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure;
- where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;
- prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;
- equipment was returned to a position or status required to support the performance of its safety functions; and
- all problems identified during the testing were appropriately documented and dispositioned in the CAP.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted seven routine surveillance testing samples, one inservice testing sample, one reactor coolant system leak detection inspection sample, and one containment isolation valve sample as defined in IP 71111.22, Sections–02 and–05.

b. Findings

No findings were identified.

1EP6 Drill Evaluation (71114.06)

.1 Training Observation

a. Inspection Scope

The inspector reviewed an issue associated with a first quarter 2015 observation of a simulator training evolution for licensed operators on March 16, 2015, which required emergency plan implementation by a licensee operations crew. This evolution was planned to be evaluated and included in performance indicator (PI) data regarding drill and exercise performance. The inspectors observed event classification and notification activities performed by the crew. The inspectors also attended the post-evolution critique for the scenario. The focus of the inspectors' activities was to note any weaknesses and deficiencies in the crew's performance and ensure that the licensee evaluators noted the same issues and entered them into the CAP. The inspectors reviewed documents listed in the Attachment to this report.

This review was performed in order to close an Unresolved Item (URI) opened during the first quarter inspection period, and as a result, this inspection did not constitute a sample as defined in IP 71114.06–06.

b. Findings

1) (Closed) Unresolved Item 05000263/2015001–04: Inadequate Evaluation of Operating Crew During Simulator Assessment

Introduction: The inspectors closed an unresolved item (URI) which was identified on March 16, 2015, due to the licensee's potential failure to properly assess and critique a senior reactor operator's (SRO's) performance during a simulator self-assessment in accordance with Procedure MTCP-03.49, "Conduct of Training Cycle Self-Assessments." During the first quarter of 2015, the inspectors determined that this issue represented a URI because more information was required to determine if a violation existed and if the performance deficiency was more than minor. This URI was initially opened in Integrated IR 05000263/2015001. The inspectors assessed additional information and determined that this URI is closed.

Description: On March 16, 2015, the NRC inspectors observed that the licensee failed to properly assess and critique a SRO's performance during a simulator self-assessment in accordance with Procedure MTCP-03.49, "Conduct of Training Cycle Self-Assessments." Specifically, during an NRC observation of a Licensed Operator Training self-assessment and emergency preparedness objective demonstration, the inspector observed that the evaluators did not adequately critique a knowledge deficiency in the Interpreting and Diagnosing Events competency area when evaluating an SM's performance. The SM's performance, if not corrected by an independent peer checker would have resulted in an emergency action level (EAL) under-classification during a graded self-assessment. This assessment included an evaluated Drill/Exercise Performance opportunity for the EAL classification in question.

The inspectors observed that following the simulator exercise, when discussing the near miss for the EAL classification, the critique failed to discuss a key deficiency associated with the SM's misinterpretation of safety parameters display system (SPDS) during his efforts to try and validate radiation monitor indications. Rather, the evaluators focused their critique on the fact that the lead reactor operator did not adequately monitor and communicate radiation levels, and linked this performance as the most prominent cause of the initial under-classification of the event. Although event classification is one of the SM's primary duties during an event, the critique failed to analyze the SM's critical role in the initial under-classification. The critique failed to probe why these performance weaknesses occurred, and failed to determine a course of specific actions for the crew to take to improve individual performance relative to the SM's role in the EAL classification.

As a result of the failure to probe the SM's performance weaknesses and the associated causes, the critique assessed the individual's performance as satisfactory, rather than as being disqualified and needing remediation. The inspectors noted that at the end of the critique, this item was not discussed as an item needing resolution, nor was it discussed that the SM had a challenge to his qualifications and needed potential remediation, which was contrary to the site's MTCP-0349 procedure. These discussions and follow-up actions did not take place until after the critique had concluded and the NRC inspectors raised questions about the SM's misinterpretation of SPDS and his overall performance. Following inspector questions regarding the performance issues on the part of the SRO, the licensee took action to disqualify the individual, develop a remediation plan, and initiate procedure changes to improve the critique process. The licensee also instituted additional management observations of the requalification training process.

Under the Section entitled "Conduct of Crew Critique," Procedure MTCP-03.49, "Conduct of Training Cycle Self-Assessments" stated in Step 4.4.1, "The Shift Manager, self-assessment team and crew participate in the scenario critique. The purpose of the Crew Critique is for the crew to understand why weaknesses occurred and determine a course of specific actions for the crew to take to improve individual and team performance." In addition, Step 4.4.5 stated, "at the end of the critique, the SM and Lead Observer should summarize the crew's identified improvement strategies and any items identified as needing resolution including simulator deficiencies and responsible individuals." Based on NRC observation, the critique did not appear to meet these standards. Corrective actions for this issue included disqualifying the individual, developing a remediation plan, and initiating procedure changes to improve the critique process. This issue was entered into the CAP (CAP 1470975).

In follow-up to this URI, the inspectors reviewed guidance contained in the SDP for the performance deficiency in question. Specifically, the inspectors reviewed IMC 0609 Appendix I and IP 71111.11 for Operator Licensing issues. The inspectors reviewed guidance in IP 71111.11, which stated, "Note that 10 CFR 55.59 requires only a requalification operating test to be conducted annually, and any additional evaluation of licensed operators is up to the facility licensee, in accordance with their accredited systems approach to training program. Given the optional nature of additional licensed operator evaluations, issues identified by the resident inspector during this activity typically will not result in documented findings, except, for example, if the issue is associated with conformance with operator license conditions or simulator performance. In particular, the checklists contained in Appendices B, C, D, E, and F concerning

examination quality, administration, security, and remedial training and re-examinations are not applicable for examinations that are not required per 10 CFR 55.59. However, observations made during this inspection activity should be provided to the facility licensee as feedback." The inspectors concluded that IP 71111.11 Appendix D, "Administration of Annual Requalification Operating Tests" would normally be used to disposition issues similar to these and was deemed not applicable in this case.

The inspectors also consulted with NRC regional operator licensing and emergency preparedness inspectors and management to seek additional guidance on whether the performance deficiency was a violation of NRC requirements and whether the issue was greater than minor. As a result of this review, the inspectors determined that this issue represented a minor performance deficiency which did not constitute a violation of NRC requirements. Specifically, the licensee failed to critique the operators' performance in accordance with MTCP-0349 which represented a performance deficiency, but because the issue occurred during a requalification training session and not during the required Annual Operating Exam, it did not represent a violation of NRC requirements and was not greater than minor. The inspectors provided this issue as an observation to licensee management as documented in CAP 1470975. As a result of this issue, this URI is closed. **(URI 05000263/2015001—04: Inadequate Evaluation of Operating Crew During Simulator Assessment)**.

2. RADIATION SAFETY

Cornerstones: Occupational and Public Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed all licensee performance indicators (PIs) for the Occupational Exposure Cornerstone for follow-up. The inspectors reviewed the results of Radiation Protection (RP) Program audits (e.g., licensee's quality assurance audits or other independent audits). The inspectors reviewed any reports of operational occurrences related to occupational radiation safety since the last inspection. The inspectors reviewed the results of the audit and operational report reviews to gain insights into overall licensee performance.

This inspection constituted a partial sample as defined in IP 71124.01-05.

b. Findings

No findings were identified.

.2 Radiological Hazard Assessment (02.02)

a. Inspection Scope

The inspectors determined if there had been changes to plant operations since the last inspection that may result in a significant new radiological hazard for onsite workers or members of the public. The inspectors evaluated whether the licensee assessed the

potential impact of these changes and has implemented periodic monitoring, as appropriate, to detect and quantify the radiological hazard.

The inspectors reviewed the last two radiological surveys from selected plant areas, and evaluated whether the thoroughness and frequency of the surveys were appropriate for the given radiological hazard.

The inspectors conducted walkdowns of the facility, including radioactive waste processing, storage, and handling areas to evaluate material conditions and performed independent radiation measurements to verify conditions.

The inspectors selected the following radiologically risk-significant work activities that involved exposure to radiation:

- Radiation Work Permit (RWP) 155507 1R27 Drywell (DW) Undervessel General Activities;
- RWP 155109 1R27 SC-Main Steam Isolation Valve Work Activities;
- RWP 155515 1R27 DW Nozzle/General Inservice Inspection Activities; and
- RWP 155511 1R27 DW CRD Changeout.

For these work activities, the inspectors assessed whether the pre-work surveys performed were appropriate to identify and quantify the radiological hazard, and to establish adequate protective measures. The inspectors evaluated the Radiological Survey Program to determine if hazards were properly identified, including the following:

- identification of hot particles;
- the presence of alpha emitters;
- the potential for airborne radioactive materials, including the potential presence of transuranics and/or other hard-to-detect radioactive materials (This evaluation may include licensee planned entry into non-routinely entered areas subject to previous contamination from failed fuel.);
- the hazards associated with work activities that could suddenly and severely increase radiological conditions and that the licensee has established a means to inform workers of changes that could significantly impact their occupational dose; and
- severe radiation field dose gradients that can result in non-uniform exposures of the body.

The inspectors observed work in potential airborne areas, and evaluated whether the air samples were representative of the breathing air zone. The inspectors evaluated whether continuous air monitors were located in areas with low background to minimize false alarms and were representative of actual work areas. The inspectors evaluated the licensee's program for monitoring levels of loose surface contamination in areas of the plant with the potential for the contamination to become airborne.

b. Findings

No findings were identified.

.3 Instructions to Workers (02.03)

a. Inspection Scope

The inspectors selected various containers holding non-exempt licensed radioactive materials that may cause unplanned or inadvertent exposure of workers, and assessed whether the containers were labeled and controlled in accordance with Title 10 CFR 20.1904, "Labeling Containers," or met the requirements of 10 CFR 20.1905(g), "Exemptions To Labeling Requirements."

The inspectors reviewed the following RWPs used to access high-radiation areas, and evaluated the specified work control instructions or control barriers:

- RWP 155507 1R27 DW Undervessel General Activities;
- RWP 155109 1R27 SC-MSIV Work Activities;
- RWP 155515 1R27 DW Nozzle/General ISI Activities; and
- RWP 155511 1R27 DW CRD Changeout.

For these RWPs, the inspectors assessed whether allowable stay times or permissible dose (including from the intake of radioactive material) for radiologically significant work under each RWP were clearly identified. The inspectors evaluated whether electronic personal dosimeter alarm set-points were in conformance with survey indications and plant policy.

The inspectors reviewed selected occurrences where a worker's electronic personal dosimeter noticeably malfunctioned or alarmed. The inspectors evaluated whether workers responded appropriately to the off-normal condition. The inspectors assessed whether the issue was included in the CAP, and dose evaluations were conducted as appropriate.

For work activities that could suddenly and severely increase radiological conditions, the inspectors assessed the licensee's means to inform workers of changes that could significantly impact their occupational dose.

b. Findings

No findings were identified.

.4 Contamination and Radioactive Material Control (02.04)

a. Inspection Scope

The inspectors observed locations where the licensee monitors potentially contaminated material leaving the radiological control area, and inspected the methods used for control, survey, and release from these areas. The inspectors observed the performance of personnel surveying and releasing material for unrestricted use and evaluated whether the work was performed in accordance with plant procedures and whether the procedures were sufficient to control the spread of contamination and prevent unintended release of radioactive materials from the site. The inspectors assessed whether the radiation monitoring instrumentation had appropriate sensitivity for the type(s) of radiation present.

The inspectors evaluated whether there was guidance on how to respond to an alarm that indicates the presence of licensed radioactive material.

The inspectors reviewed the licensee's procedures and records to verify that the radiation detection instrumentation was used at its typical sensitivity level based on appropriate counting parameters. The inspectors assessed whether or not the licensee has established a *de facto* "release limit" by altering the instrument's typical sensitivity through such methods as raising the energy discriminator level or locating the instrument in a high-radiation background area.

The inspectors selected several sealed sources from the licensee's inventory records and assessed whether the sources were accounted for and verified to be intact.

The inspectors evaluated whether any transactions, since the last inspection, involving nationally tracked sources were reported in accordance with 10 CFR 20.2207.

b. Findings

No findings were identified.

.5 Radiological Hazards Control and Work Coverage (02.05)

a. Inspection Scope

The inspectors evaluated ambient radiological conditions (e.g., radiation levels or potential radiation levels) during tours of the facility. The inspectors assessed whether the conditions were consistent with applicable posted surveys, RWPs, and worker briefings.

The inspectors evaluated the adequacy of radiological controls, such as required surveys, RP job coverage (including audio and visual surveillance for remote job coverage), and contamination controls. The inspectors evaluated the licensee's use of electronic personal dosimeters in high-noise areas as high-radiation area monitoring devices.

The inspectors assessed whether radiation monitoring devices were placed on the individual's body consistent with licensee procedures. The inspectors assessed whether the dosimeter was placed in the location of highest expected dose or that the licensee properly employed an NRC approved method of determining effective dose equivalent.

The inspectors reviewed the application of dosimetry to effectively monitor exposure to personnel in high-radiation work areas with significant dose rate gradients.

The inspectors reviewed the following RWPs for work within airborne radioactivity areas with the potential for individual worker internal exposures:

- RWP 155507 1R27 DW Undervessel General Activities;
- RWP 155515 1R27 DW Nozzle/General ISI Activities; and
- RWP 155511 1R27 DW CRD Changeout.

For these RWPs, the inspectors evaluated airborne radioactive controls and monitoring, including potential for significant airborne levels (e.g., grinding, grit blasting, system

breaches, entry into tanks, cubicles, and reactor cavities). The inspectors assessed barrier (e.g., tent or glove box) integrity and temporary high-efficiency particulate air ventilation system operation.

The inspectors examined the licensee's physical and programmatic controls for highly activated or contaminated materials (i.e., nonfuel) stored within spent fuel and other storage pools. The inspectors assessed whether appropriate controls (i.e., administrative and physical controls) were in place to preclude inadvertent removal of these materials from the pool.

The inspectors examined the posting and physical controls for selected high-radiation areas and very-high radiation areas to verify conformance with the occupational PI.

b. Findings

No findings were identified.

.6 Risk-Significant High-Radiation Area and Very-High Radiation Area Controls (02.06)

a. Inspection Scope

The inspectors discussed with the RP manager the controls and procedures for high-risk, high radiation areas and very-high radiation areas. The inspectors discussed methods employed by the licensee to provide stricter control of very-high radiation area access as specified in 10 CFR 20.1602, "Control of Access to Very-High Radiation Areas," and RG 8.38, "Control of Access to High and Very-High Radiation Areas of Nuclear Plants." The inspectors assessed whether any changes to licensee procedures substantially reduce the effectiveness and level of worker protection.

The inspectors discussed the controls in place for special areas that have the potential to become very-high radiation areas during certain plant operations with first-line health physics supervisors (or equivalent positions having backshift health physics oversight authority). The inspectors assessed whether these plant operations require communication beforehand with the health physics group, so as to allow corresponding timely actions to properly post, control, and monitor the radiation hazards including re-access authorization.

b. Findings

No findings were identified.

.7 Radiation Worker Performance (02.07)

a. Inspection Scope

The inspectors observed radiation worker performance with respect to stated RP work requirements. The inspectors assessed whether workers were aware of the radiological conditions in their workplace and the RWP controls/limits in place, and whether their performance reflected the level of radiological hazards present.

The inspectors reviewed radiological problem reports since the last inspection that found the cause of the event to be human performance errors. The inspectors evaluated

whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. The inspectors discussed with the RP manager any problems with the corrective actions planned or taken.

b. Findings

No findings were identified.

.8 Radiation Protection Technician Proficiency (02.08)

a. Inspection Scope

The inspectors observed the performance of the RP technicians with respect to all RP work requirements. The inspectors evaluated whether technicians were aware of the radiological conditions in their workplace and the RWP controls/limits, and whether their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities.

The inspectors reviewed radiological problem reports since the last inspection that found the cause of the event to be RP technician error. The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems.

b. Findings

No findings were identified.

.9 Problem Identification and Resolution (02.09)

a. Inspection Scope

The inspectors evaluated whether problems associated with radiation monitoring and exposure control were being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee's CAP. The inspectors assessed the appropriateness of the corrective actions for a selected sample of problems documented by the licensee that involve radiation monitoring and exposure controls. The inspectors assessed the licensee's process for applying operating experience to their plant.

b. Findings

No findings were identified.

2RS2 Occupational As-Low-As-Reasonably-Achievable Planning and Controls (71124.02)

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed pertinent information regarding plant collective exposure history, current exposure trends, and ongoing or planned activities in order to assess

current performance and exposure challenges. The inspectors reviewed the plant's 3-year rolling average collective exposure.

The inspectors reviewed the site-specific trends in collective exposures and source term measurements.

The inspectors reviewed site-specific procedures associated with maintaining occupational exposures as-low-as-reasonably-achievable (ALARA), which included a review of processes used to estimate and track exposures from specific work activities.

This inspection constituted a partial sample as defined in IP 71124.02-05.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Security

4OA1 Performance Indicator Verification (71151)

.1 Safety System Functional Failures

a. Inspection Scope

The inspectors sampled licensee submittals for the safety system functional failures PI for the period from the second quarter 2014 through the first quarter 2015. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013, and NUREG–1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73" definitions and guidance, were used. The inspectors reviewed the licensee's operator narrative logs, operability assessments, maintenance rule records, maintenance WOs, issue reports, event reports and NRC Integrated IRs for the period of April 2014 through March 2015 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one safety system functional failures sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.2 Reactor Coolant System Leakage

a. Inspection Scope

The inspectors sampled licensee submittals for the RCS Leakage PI for the period from the second quarter 2014 through the first quarter 2015 and guidance contained in the NEI Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013, were used. The inspectors reviewed the licensee's operator logs, RCS leakage tracking data, issue reports, event reports and NRC Integrated IRs for the period of April 2014 through March 2015 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one reactor coolant system leakage sample as defined in IP 71151–05.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Security

.1 Routine Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As part of the various baseline IPs discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify they were being entered into the licensee's CAP at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Attributes reviewed included: identification of the problem was complete and accurate; timeliness was commensurate with the safety significance; evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent-of-condition reviews, and previous occurrences reviews were proper and adequate; and that the classification, prioritization, focus, and timeliness of corrective actions were commensurate with safety and sufficient to prevent recurrence of the issue. Minor issues entered into the licensee's CAP as a result of the inspectors' observations are included in the Attachment to this report.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead, by procedure they were considered an integral part of the inspections performed during the quarter and documented in Section 1 of this report.

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's CAP. This review was accomplished through inspection of the station's daily condition report packages.

These daily reviews were performed by procedure as part of the inspectors' daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. Findings

No findings were identified.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.2 above, licensee trending efforts, and licensee human performance results. The inspectors' review nominally considered the 6-month period of January 1, 2015 through June 30, 2015 although some examples expanded beyond those dates where the scope of the trend warranted.

The review also included issues documented outside the normal CAP in major equipment problem lists, repetitive and/or rework maintenance lists, departmental problem/challenges lists, system health reports, quality assurance audit/surveillance reports, self-assessment reports, and Maintenance Rule assessments. The inspectors compared and contrasted their results with the results contained in the licensee's CAP trending reports. Corrective actions associated with a sample of the issues identified in the licensee's trending reports were reviewed for adequacy.

This review constituted one semi-annual trend inspection sample as defined in IP 71152–05.

b. Findings

No findings were identified.

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153)

.1 Loss of Shutdown Cooling (SDC) and Division I Safety Related Electrical Buses During Surveillance Test

a. Inspection Scope

The inspectors reviewed the plant's response to a loss of SDC and Division I safety related electrical buses during the plant's RFO. On May 2, 2015, while performing a surveillance test on the Division 1 4160V bus, plant workers installed an electrical bypass jumper in the wrong location, contrary to procedural direction. As a result, the plant experienced a Loss of Bus 15 (the Division I - 4160V ECCS Bus) and Load Center 103 (a Division I - 480 Vac Safety Related Load Center). This led to a trip of the running Division II SDC Pump when position indication that was powered by Division I electrical buses was deenergized for the RHR SDC inboard suction isolation valve. The Division I EDG, which would normally start to power the deenergized Division I safety related buses, was out of service due to the ongoing test. Power was restored to the 480 Vac and 125 Vdc buses approximately 2 hours and 15 minutes after the event occurred, by cross-tying the essential buses to the available Division 2 power. During the event as a result of the unplanned loss of SDC, temperature in the reactor vessel rose approximately 10 degrees F.

The inspectors responded to the Control Room during the event and verified appropriate actions were taken to address the situation, operators maintained awareness of risks associated with restoration, the crew adequately evaluated EALs, and that the operators identified and entered the appropriate TS action statements. The inspectors observed that the workers in the field had not performed an adequate concurrent verification, which led to the loss of the electrical buses. The inspectors also noted that plant personnel did not adequately identify the risk to losing SDC that was associated with the activity. The inspectors also noted that during the event, the licensee had failed to enter tech spec 3.8.8 "Electrical Distribution" action statements for loss of a required 480V bus. The issue was entered into the licensee's CAP and corrected. The inspectors determined that this issue was minor due to the fact that the crew ultimately followed all required actions of TS 3.8.8 as part of their entries into various other tech spec action statements associated with the event. Documents reviewed are listed in the Attachment to this report.

This event follow-up review constituted one sample as defined in IP 71153–05.

b. Findings

1) Loss of Electrical Buses and Shutdown Cooling (SDC) Due to Inadequate Procedure Adherence

Introduction: A self-revealed finding of very low safety significance and an NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," was identified due to the failure to properly implement Procedure 0304-01, "Safeguard Bus Loss of Voltage Protection Relay Unit Calibration – Safeguards Bus No. 15." Specifically, electrical maintenance workers failed to comply with Step 20 which directed the installation of a jumper between terminals ZX10 and ZX11 in an electrical panel, when they incorrectly installed the electrical jumper between terminals ZX11 and ZX12.

This resulted in the loss of the Division I safety related 4160 Vac, 480 Vac, and 125 Vdc electrical buses, which subsequently led to the loss of SDC for approximately 3 hours and 15 minutes.

Description: On May 2, 2015, while performing a surveillance test on the Division 1 4160V bus, plant workers installed an electrical bypass jumper in the wrong location, contrary to procedural direction. The plant was in the middle of a RFO, and the reactor cavity was fully flooded at the time of the event. As a result of the error, the plant experienced a Loss of Bus 15 (Division I - 4160V ECCS Bus) and Load Center 103 (Division I - 480 Vac Safety Related Load Center). This subsequently resulted in a trip of the running Division II SDC Pump when position indication that was powered by Division I electrical buses was deenergized for the RHR SDC inboard suction isolation valve. The Division I EDG, which would normally start to power the deenergized safety related buses, was out of service due to the ongoing test.

The event was complicated by the fact that a capacity test of the #11 Battery (Division I 125 VDC) was in progress at the time of the event and the battery's normal Division I DC loads were powered from a temporary battery with limited capacity. The loss of Load Center 103 removed power to the charger supplying this temporary battery. The battery discharged throughout the event to the point that it was unable to provide power to associated loads. As a result, the control room lost the ability to remotely operate Division I breakers and annunciators were lost for some control boards in the control room. This complicated actions to recover power to the deenergized 480 Vac bus and the 125 Vdc bus because when operators attempted to close the required breaker, it failed to close due to the loss of the temporary battery during the event. Ultimately operators were required to manually close the breaker to restore power.

Power was restored to the 480 Vac and 125 Vdc buses approximately 2 hours and 15 minutes after the event occurred, by cross-tying the essential buses to the available Division 2 power. This allowed for restoration of SDC within approximately 3 hours and 15 minutes. The Division 1 4160 V bus was restored several hours later, but the loss of this bus alone had minimal impact due to the fact that the licensee was in a Division I work window at the time of the event, and the bus was not required. The event was mitigated by the fact that Division II Fuel Pool Cooling remained in service providing cooling to the reactor during the event, and was able to handle the entire decay heat load of the core at the time the event occurred. However, during the event as a result of the unplanned loss of SDC, temperature in the reactor vessel rose approximately 10 degrees Fahrenheit.

As a result of this event, the plant entered an orange outage risk condition, the second highest risk category for the decay heat removal and electrical system critical safety functions. The plant also entered into strict plant status controls, preventing workers from entering the plant to perform work that was not directly in support of recovery from the event. The licensee made an 8 hour report of 10CFR 50.72 for a potential loss of the decay heat removal safety function. This event resulted in a site clock reset, and the licensee initiated a RCE to determine the cause of the event.

The inspectors determined that the workers in the field had not performed an adequate concurrent verification, which led to the loss of the electrical buses. The inspectors also noted that plant personnel did not adequately identify the risk of losing SDC that was associated with the work activity. As a result, the inspectors determined that the

licensee failed to adequately avoid complacency. Specifically, the plant electrical workers did not adequately implement error reduction tools and the organization failed to plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes.

Analysis: The inspectors determined that the failure to correctly perform step 20 of procedure 0304-01 was a performance deficiency because it represented a failure to meet 10 CFR Part 50 Appendix B Criterion V, the cause was reasonably within the licensee's ability to foresee/correct, and it should have been prevented. The inspectors evaluated the issue using the SDP and determined that it was more than minor because it adversely impacted the Initiating Events Cornerstone attribute of Human Performance and Configuration Control, and affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the event caused the loss of Division I safety related buses and SDC, which resulted in the plant entering orange risk due to the challenge the event posed to the decay heat removal and electrical system critical safety functions.

The inspectors assessed the significance of this finding using IMC 0609, Attachment 4, "Initial Characterization of Findings," and determined that the findings should be evaluated under the Initiating Events Cornerstone. Because the plant was in a RFO with the reactor cavity fully flooded at the time of the event, the inspectors assessed the performance deficiency using IMC 0609 Appendix G, "Shutdown Operations Significance Determination Process Phase 1 Initial Screening and Characterization of Findings." Using "Exhibit 2 - Initiating Events Screening Questions," the inspectors determined that this issue was of very low safety significance (Green) because they answered 'no' to all of the screening questions. 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, Avoid Complacency aspect because of the failure of licensee individuals to implement error reduction tools and the failure of the organization to plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes [H.12]. Specifically, the workers in the field had not performed an adequate concurrent verification, which led to the loss of the electrical buses, and the organization did not adequately identify the risk of losing SDC that was associated with the activity.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," required, in part, that activities affecting quality be prescribed by documented procedures of a type appropriate to the circumstances and be accomplished in accordance with these procedures. The licensee established Step 20 of Procedure 0304-01, "Safeguard Bus Loss of Voltage Protection Relay Unit Calibration – Safeguards Bus No. 15," Revision 13, as the implementing procedure for a periodic surveillance test of Bus No. 15. Step 20 stated: "(CONCURRENT VERIFICATION) IF Bus 15 is energized, THEN install jumper in 152-505, between terminals ZX10 and ZX11 (to bypass 127-5 and 127-5X, contacts 3 and 7)."

Contrary to the above, on May 2, 2015, electrical maintenance workers failed to correctly perform Step 20 of Procedure 0304-01 when they inappropriately installed the electrical jumper between terminals ZX11 and ZX12. This resulted in the loss of the Division 1 safety related 4160 Vac, 480 Vac, and 125 Vdc electrical buses, which subsequently led to the loss of SDC for approximately 3 hours and 15 minutes. Because this violation

was of very low safety significance and entered into the licensee's CAP (1477351), this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000263/2015002–06: Loss of Electrical Buses and Shutdown Cooling (SDC) Due to Inadequate Procedure Adherence)**. Initial corrective actions for this issue included immediately invoking strict plant status controls to focus efforts on recovery, restoring the electrical buses and SDC to operation, and reinforcing risk recognition and human performance tools. A RCE was in progress at the time this inspection period concluded.

.2 <u>(Closed) LER 05000263/2015-001-00, "Operations with a Potential to Drain the Reactor Vessel (OPDRV) Without Secondary Containment Operable" and Exercise of Enforcement Discretion (EA-15-130)</u>

Between April 23, 2015 and May 1, 2015 and again between May 2, 2015 and May 8, 2015, the Monticello Nuclear Generating Plant (MNGP) performed OPDRV activities while in Mode 5 without an operable secondary containment. An OPDRV is an activity that could result in the draining or siphoning of the RPV water level below the top of fuel, without crediting the use of mitigating measures to terminate the uncovering of fuel. Secondary containment is required by TS 3.6.4.1 to be operable during OPDRV activities. The required action for this specification is to suspend OPDRV operations. Therefore, entering the OPDRV without establishing secondary containment integrity was considered a condition prohibited by TS as defined by 10 CFR 50.73(a)(2)(i)(B).

The NRC issued EGM 11-003, Revision 2, on December 13, 2013, to provide guidance on how to disposition boiling water reactor licensee noncompliance with TS containment requirements during OPDRV operations. The NRC considers enforcement discretion related to secondary containment operability during Mode 5 OPDRV activities appropriate because the associated interim actions necessary to receive the discretion ensure an adequate level of safety by requiring licensees' immediate actions to (1) adhere to the NRC plain language meaning of OPDRV activities, (2) meet the requirements which specify the minimum makeup flow rate and water inventory based on OPDRV activities with long drain down times, (3) ensure that adequate defense in depth is maintained to minimize the potential for the release of fission products with secondary containment not operable by (a) monitoring RPV level to identify the onset of a LOI event, (b) maintaining level monitoring to ensure secondary containment can be closed before inventory is drained to the RPV flange, (c) maintaining the capability to isolate the potential leakage paths, (d) prohibiting Mode 4 (cold shutdown) OPDRV activities, and (e) prohibiting movement of recently irradiated fuel with the spent fuel storage pool gates removed in Mode 5, and (4) ensure that licensees follow all other Mode 5 TS requirements for OPDRV activities.

The inspectors reviewed this licensee event report (LER) for potential performance deficiencies and/or violations of regulatory requirements. The inspectors also reviewed the station's implementation of the EGM during the OPDRVs for which the EGM was invoked. Based on review of the following items, the inspectors determined that the licensee met the EGM requirements for discretion:

1. The inspectors observed that the OPDRV activities were logged in the control room narrative logs and that the log entry appropriately recorded that the standby source of makeup designated for the evolutions.

2. The inspectors noted that the reactor vessel water level was maintained at least 21 feet and 11 inches over the top of the RPV flange as required by TS 3.9.6. The inspectors also verified that at least one safety-related pump was available as the standby source of makeup designated in the control room narrative logs for the evolutions. The inspectors confirmed that the worst case estimated time to drain the reactor cavity to the RPV flange was greater than 24 hours.

3. The inspectors reviewed Engineering Change documents which calculated the time to drain down during these activities and the feasibility of pre-planned actions the station would take to isolate potential leakage paths during these periods of time.

4. The inspectors verified that the OPDRVs were not conducted in Mode 4 and that the licensee did not move recently irradiated fuel during the OPDRVs. The inspectors noted that MNGP had in place a contingency plan for isolating the potential leakage path and verified that two independent means of measuring RPV water level were available for identifying the onset of LOI events.

TS 3.6.4.1 required, in part, that secondary containment shall be operable during OPDRV. TS 3.6.4.1, Condition C, required the licensee to initiate action to suspend OPDRV immediately when secondary containment is inoperable. Contrary to the above, between April 23, 2015 and May 1, 2015 and again between May 2, 2015 and May 8, 2015, MNGP performed OPDRV activities while in Mode 5 without an operable secondary containment. Specifically, the station performed the following OPDRV activities without an operable secondary containment:

- 12 Recirculation System pump upper seal replacement;
- 12 Recirculation System modifications to add and replace valves; and
- 11 Recirculation System modifications to add and replace valves.

Because the violation occurred during the discretion period described in EGM 11-003, Revision 2, the NRC is exercising enforcement discretion in accordance with Section 3.5, “Violations Involving Special Circumstances,” of the NRC Enforcement Policy and, therefore, will not issue enforcement action for this violation (EA-15-130).

In accordance with EGM 11-003, Revision 2, each licensee that receives discretion must submit a license amendment request within 12 months of the NRC staff’s publication in the Federal Register of the notice of availability for a generic change to the standard TS to provide more clarity to the term OPDRV. The inspectors observed that Monticello is tracking the need to submit a license amendment request in its CAP (CAP 1476012). LER 05000263/2015-001-00 is now closed.

This event follow-up review constituted one sample as defined in IP 71153-05.

4OA5 Other Activities

.1 Power Uprate Related Inspection Activities (71004)

a. Inspection Scope

During this inspection period, the inspectors observed activities related to the EPU license amendment which authorized the licensee to operate at 2004 MWth. Specific activities are documented below, and as referenced:

- Section 1R13–This section documents specific inspector reviews of EPU activities associated with the increasing of reactor power from 1953 MWth to 2004 MWth (100% power), a first time evolution.

b. Findings

No findings were identified.

.2 Correction to the Cover Letter of Integrated and Power Uprate Inspection Report 05000263/2015001

On May 8, 2015, the NRC issued MNGP Integrated and Power Uprate Inspection Report 05000263/2015001. This report included a cover letter which stated, "Additionally, five licensee-identified findings are listed in Section 4OA7 of this report." However, as reflected in Section 4OA7 of that report, only four licensee-identified findings were identified during the 1st Quarter of 2015. These findings were correctly documented in Section 4OA7. The cover letter, which stated there were five licensee-identified findings, was incorrect.

4OA6 Management Meetings

.1 Exit Meeting Summary

On July 21, 2015, the inspectors presented the inspection results to Mr. Peter Gardner and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

Interim exits were conducted for:

- The inspection results for the areas of radiological hazard assessment and exposure controls; and occupational ALARA planning and controls with Mr. Peter Gardner, Vice President, on April 17, 2015.
- The results of the ISI with the Site Vice-President, Mr. Peter Gardner, on April 23, 2015.

The inspectors confirmed that none of the potential report input discussed was considered proprietary. Proprietary material received during the inspection was returned to the licensee.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

P. Gardner, Site Vice President
K. Scott, Director of Site Operations
H. Hanson, Jr., Plant Manager
T. Witschen, Operations Manager
M. Lingenfelter, Director of Engineering
K. Jepson, Organizational Effectives and Human Performance Manager
B. Olson, Maintenance Manager
S. Quiggle, Chemistry Manager
C. England, Radiation Protection Manager
A. Ward, Regulatory Affairs Manager
T. Hedges, Radiation Protection General Supervisor
P. Young, Program Engineering Supervisor
T. Jones, Non-Destructive Examination Coordinator
R. Deopere, Inservice Inspection Program Owner
S. Sollom, Regulatory Affairs

Nuclear Regulatory Commission

K. Riemer, Chief, Reactor Projects Branch 2

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000263/2015002-01	NCV	Failure to Maintain Portable Fire Extinguishers in Accordance with Fire Strategy (Section 1R05)
05000263/2015002-02	NCV	Failure to Measure Interpass Temperature (Section 1R08)
05000263/2015002-03	NCV	Failure to Maintain Secondary Containment and Standby Gas Treatment System Operable During OPDRV Activities (Section 1R13)
05000263/2015002-04	NCV	Failure to Fill the Reactor Cavity in Accordance with Refueling Preparation Procedure (Section 1R13)
05000263/2015002-05	NCV	Inadequate Clearance Order Results in Unplanned OPDRV (Section 1R15)
05000263/2015002-06	NCV	Loss of Electrical Buses and Shutdown Cooling (SDC) Due to Inadequate Procedure Adherence (Section 4OA3)

Closed

05000263/2015002-01	NCV	Failure to Maintain Portable Fire Extinguishers in Accordance with Fire Strategy (Section 1R05)
05000263/2015002-02	NCV	Failure to Measure Interpass Temperature (Section 1R08)
05000263/2015002-03	NCV	Failure to Maintain Secondary Containment and Standby Gas Treatment System Operable During OPDRV Activities (Section 1R13)
05000263/2015002-04	NCV	Failure to Fill the Reactor Cavity in Accordance with Refueling Preparation Procedure (Section 1R13)
05000263/2015002-05	NCV	Inadequate Clearance Order Results in Unplanned OPDRV (Section 1R15)
05000263/2015002-06	NCV	Loss of Electrical Buses and Shutdown Cooling (SDC) Due to Inadequate Procedure Adherence (Section 4OA3)
05000263/2015001-04	URI	Inadequate Evaluation of Operating Crew During Simulator Assessment (Section 1EP6)
05000263/2015-001-00	LER	Operations with a Potential to Drain the Reactor Vessel (OPDRV) Without Secondary Containment Operable (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspector reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

Section 1R01

- 1478; External Flood Surveillance; Revision 14
- 4 AWI-04.02.01; Revision 24
- A.6; Acts of Nature; Revision 52

Section 1R04

- 1487; Site Housekeeping Quarterly Inspection; Revision 7
- 2154-22; EDG Emergency Service Water System Prestart Valve Checklist; Revision 24
- 2154-23; RHR Service Water Restart Checklist; Revision 33
- 2154-34; Emergency Service Water System Prestart Valve Checklist; Revision 29
- 2206; Plant Prestart Checklist EDG – Emergency Service Water System; Revision 4
- A.6; Acts of Nature Revision 51
- AR 727626; Further Review of NRC Questions Concerning Wind Generated Missiles
- AWI-04.02.01; Housekeeping; Revision 24
- B.08.01.02-05; EDG Emergency Service Water; Revision 18
- CAP 1474448; Small area article procedure not initially followed—NRC
- CAP 1474485; Issues Identified During DAC Walkdown
- CAP 1476696; Questions/comments from NRC Resident plant walkdown
- DBD-S.06; Turbine Building; Revision 5
- DBD-T.05; External Flooding Topic; Revision 5
- NH-36664 P&ID RHR Service Water & Emergency Service Water Systems; Revision 87
- NH-36664; P&ID RHR Service Water & Emergency Service Water Systems; Revision 87
- NH-36665; P&ID Service Water System & Make-Up Intake Structure; Revision 97
- NH-36665; P&ID Service Water System & Make-Up Intake Structure; Revision 97
- USAR-02; Site and Environs; Revision 30

Section 1R05

- 0274; Fire Hose Hydrostatic Test – Interior Hose Stations; Revision 30
- 1123; Portable Fire Extinguishers; Revision 63
- 4 AWI-08.01.01; Fire Prevention Practices; Revision 47
- CA-04-204; NFPA 10 Installation of Portable Fire Extinguisher Code Evaluation; Revision 2
- CAP 1475406; Control of leak / potential leak areas in the SJAE Room
- CAP 1477246; NRC Questions Raised in Condenser Bay
- CAP 1477456; Two Fire Extinguishers in Hotside Found With Low Pressure
- CAP 1477743; NRC Identified: Inadequate Hand Protection After Coaching
- CAP 1477765; Scaffold Danger Tag was Found Hung on a Permanent Ladder
- CAP 1484142; A.3 Procedures are Inconsistent identifying SSD Equipment
- CAP 1484163; Fire Extinguishers in Fire Zone 12C/TB Condenser Area
- CAP 1484257; 1123-01 Fire Extinguisher PM Not Performed as Written
- Strategy A.3-12-E; Fire Zone 12-C—Condenser Area; Revision 8

- Strategy A.3-12-E; Fire Zone 12-E—Steam Jet Air Ejector Room; Revision 1
- Strategy A.3-14-B; Fire Zone 14-B—Isophase Bus Area; Revision 8
- Strategy A.3-2-F; Fire Zone 2-F—Main Steam Chase; Revision 3
- Strategy A.3-33; Fire Zone 33 – EFT Building Third Floor; Revision 7
- WO 441327; 1123-01 Port Fire Extinguisher Ck Hi Rad Areas; April 1, 2013
- WO 465763; VNDR-FIR, 1135 Portable Fire Extinguishers Annual Test; December 3, 2013
- WO 468436; MECH-FIR 1123 Portable Fire Extinguishers Mnthly Test; July 18, 2013
- WO 491815; VNDR – FIR, 1135 Portable Fire Extinguishers Annual Test; December 3, 2014
- WO 520101; 1123 Portable Fire Extinguisher RS Monthly Test; May 7, 2015

Section 1R06

NF-74413-4; Underground Services Electrical Power; Revision 93
NF-74413-6; Underground Services of Division II Cable Raceway System; Revision 80
WO 503242; Underground Vaults for Water Inspection; Revision 0

Section 1R08 Inservice Inspection

- 2013MT002; MT of RHR Discharge B Double Spring/4 Lugs; dated April 25, 2013
- 2015VT008; VT-3 Main Steam A, H-7; dated April 21, 2015
- 4 AWI-07.02.03; Fleet Welding Program Site Specific Requirements; Revision 16
- AR 01373691; RF026 ISI – Thread Damage on Stud; dated March 8, 2013
- CAP 1371305; RHRSW Piping Near TMin Found During SW-MIC Examination
- CAP 1377629; Coppery Material Fused or Imbedded in Torus Shell
- CAP 1378157; Small Random Weld Bead on Torus Shell
- CAP 1378342; IWE Inspection Results
- CAP 1385358; Error Identified on UT Calibration Report
- CAP 1387838; Corrosion Observed at Base of Head Vent Pipe during RX Hydro
- CAP 1388615; NRC Question Prompted Review of Eval for Went Line Corrosion
- CAP 1417186; The NRC Asked a Question About Vent Valve Configuration
- FP-PE-NDE-425; Ultrasonic Thickness Examination – Localized Corrosion; Revision 1
- FP-PE-NDE-530; Visual Examination, VT-3; Revision 7
- M1_I5-P1_RF26; MT of RHR Discharge
- Memorandum; "Owner" as Defined by ASME; dated April 11, 2002
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- PQR NSP-PQR-1270/1271; Revision 0
- PQR PAL-SM-8-8(2); Revision 0
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- CAP 1476870; NRC OPDRV Question: SBGT Operability Covered by EGM 11-003
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- CAP 1479284; NRC RI OPDRV Comment
- CAP 1482991; Increase in vibration levels on 11 RX MG Set tachometer
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- CAP 1479416; RCS Leakage Test, 1R27, 2015 – Documented Leakage
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- CAP 1479436; 1R27 RCS Leakage Test – XR-10-2A
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- CAP 1479438; 1R27 RCs Leakage Test – RHR-73-2
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- CAP 1479440; 1R27 RCS Leakage Test – FW-96-01
- CAP 1479441; 1R27 RCS Leakage Test – MO-2076
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- CAP 1479450; 1R27 RCS Leakage Test – CRD-101/30-07
- CAP 1479451; 1R27 RCS Leakage Test – CRD-105/38-23
- CAP 1479452; 1R27 RCS Leakage Test – CRD-105/34-23
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- WO 490763-01; 0255-20-IIC-1 RX Coolant Pressure Boundary Leakage Test; March 19, 2015

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- WO 507436

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- CAP 1474585; NRC Question – Should VHRA Controls to be Applied to LPRM Keys
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- Nuclear Oversight 3rd Trimester 2014 Executive Summary; dated January 16, 2015
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- Report # 2014-02-034; NOS Observation Report; dated July 7-25, 2014
- RWP 155109 1R27 SC-MSIV Work Activities; Revision 0
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- Snapshot Self-Assessment; Assess the Radiological Hazards Program Utilizing the NRC Inspection Manual; dated February 21, 2015

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- QF0445; NRC Data Collection and Submittal (RCS Total Leakage) – 3rd Quarter 2014; October 13, 2014
- QF0445; NRC Data Collection and Submittal (RCS Total Leakage) – 4th Quarter 2014; January 9, 2015
- QF0445; NRC Data Collection and Submittal (Safety System Functional Failure) – 2nd Quarter 2014; July 1, 2014
- QF0445; NRC Data Collection and Submittal (Safety System Functional Failure) – 3rd Quarter 2014; October 2, 2014
- QF0445; NRC Data Collection and Submittal (Safety System Functional Failure) – 4th Quarter 2014; January 12, 2015
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- Monticello RF27 Post Outage Critique; June 23, 2015
- Monticello RF27 Post Outage Critique; June 26, 2015
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- SAR 1429618/1462861; Department/Functional Area DRUM Report 4th Quarter 2014—Operations Department; March 13, 2015

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- 2208; Operations with Potential to Drain the Reactor Vessel (OPDRV) Checklist; Revision 3
- 9006; Reactor Well and Dryer-Separator Storage Pool Draining Procedure; Revision 27
- A.2-101; Classification of Emergencies; Revision 49
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- Fuel Pool Heat Exchanger Decay Heat Removal Capability Proto-HX Evaluation; May 2, 2015
- LER 2015-001-00; Operations with a Potential to Drain the Reactor Vessel (OPDRV) Without Secondary Containment Operable; June 16, 2015
- MNGP Control Room Log Entries for Loss of SDC; May 2, 2015
- Monticello Site Clock Reset-Red Sheet—Loss of SDC; May 2, 2015
- Monticello Station Log Entries regarding OPDRV Activities; April 11, 2015 through May 23, 2015
- NE-36404-4A; MNGP RHR Pump P-202B ACB 152-604 Schematic Diagram; Revision 76
- NF-36298-2; DC Electrical Load Distribution One Line Diagram; Revision F
- NF-36388; Reactor Building Spent Fuel & Dryer Separator Storage Pool—Pool Location Plan; Revision 4
- NF-36392; Rev A-Reactor Building Spent Fuel & Dryer Separator Storage Pool Shield Plug Detail; Revision 75

- NX-7823-4-11C; RHR Shutdown Cooling Supply Inboard Isolation Valve MO-2029 – Scheme B3333; Revision 76
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- Operations Manual C.4-B.09.10.A; Abnormal Procedure—Loss of a 125V DC Bus; Revision 14
- OWI-03.03; Operations with Potential to Drain the Reactor; Revision 9 & 10
- WO 498933; 12 Recirc Upper Seal Oscillations; April 23, 2015
- WO 505374; 11 and 12 Recirc Pump Add/Replace Valves; April 25, 2015

LIST OF ACRONYMS USED

AC	Alternating Current
ACE	Apparent Cause Evaluation
ADAMS	Agencywide Document Access Management System
ALARA	As-Low-As-Is-Reasonably-Achievable
AOP	Abnormal Operating Procedure
ASME	American Society of Mechanical Engineers
BWR	Boiling-Water Reactor
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CRD	Control Rod Drive
CST	Condensate Storage Tank
DW	Drywell
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EFT	Emergency Filtration Train
EGM	Enforcement Guide Memorandum
EPU	Extended Power Uprate
FPC	Fuel Pool Cooling
IEL	Initiating Event Likelihood
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
ISI	Inservice Inspection
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LOI	Loss of Inventory
MNGP	Monticello Nuclear Generating Plant
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NRC	U.S. Nuclear Regulatory Commission
OPDRV	Operations with the Potential to Drain the Reactor Vessel
OSP	Outage Safety Plan
PARS	Publicly Available Records
PI	Performance Indicator
POS	Plant Operational State
psig	Pounds Per Square Inch Gauge
RCE	Root Cause Evaluation
RCS	Reactor Coolant System
RFO	Refueling Outage
RG	Regulatory Guide
RHR	Residual Heat Removal
RP	Radiation Protection
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Cleanup
RWP	Radiation Work Permit
SBGT	Standby Gas Treatment

SDC	Shutdown Cooling
SDP	Significance Determination Process
SDV	Scram Discharge Volume
SM	Shift Manager
SPDS	Safety Parameters Display System
SRA	Senior Reactor Analyst
SRO	Senior Reactor Operator
SSC	Structure, System, and Component
TS	Technical Specification
TSO	Transmission System Operator
USAR	Updated Safety Analysis Report
URI	Unresolved Item
Vac	Volts Alternating Current
VCI	Vapor Phase Corrosion Inhibitor
Vdc	Volts Direct Current
VT	Visual Examination
WO	Work Order
WPS	Welding Procedure Specification

Nuclear Generating Plant. In addition, if you disagree with a cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Monticello Nuclear Generating Plant.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390, "Public Inspections, Exemptions, Requests for Withholding," of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC's Public Document Room or from the Publicly Available Records (PARS) component of the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Kenneth Riemer, Branch Chief
Branch 2
Division of Reactor Projects

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