UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 E. LAMAR BLVD.
ARLINGTON, TX 76011-4511

January 26, 2017

EA-16-226

Mr. Ken Higginbotham
Vice President-Nuclear and CNO
Nebraska Public Power District
Cooper Nuclear Station
72676 648A Avenue
P.O. Box 98
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION – NRC INTEGRATED INSPECTION REPORT 05000298/2016004 AND EXERCISE OF ENFORCEMENT DISCRETION

Dear Mr. Higginbotham:

On December 31, 2016, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Cooper Nuclear Station. On January 11, 2017, the NRC inspectors discussed the results of this inspection with you and other members of your staff. The results of this inspection are documented in the enclosed report.

NRC inspectors documented four findings of very low safety significance (Green) in this report. All of these findings involved violations of NRC requirements. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the Enforcement Policy.

A violation involving the failure to maintain the operability of secondary containment during operations with the potential to drain the reactor vessel (OPDRVs) was identified. Specifically, from October 1, 2016, through October 25, 2016, Cooper Nuclear Station performed a total of four OPDRV activities with secondary containment inoperable in violation of Technical Specification 3.6.4.1, “Secondary Containment.” The NRC issued Enforcement Guidance Memorandum 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 3, on January 15, 2016, allowing for the exercise of enforcement discretion for such OPDRV-related technical specification violations, when certain criteria are met. The NRC concluded that Cooper Nuclear Station met these criteria. Therefore, the NRC is exercising enforcement discretion in accordance with Section 3.5, “Violations Involving Special Circumstances,” of the NRC Enforcement Policy and will not issue enforcement action for this violation, subject to a timely license amendment request being submitted.

If you contest the violations or significance of 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 U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region IV; the Director, Office of Enforcement; and the NRC resident inspector at the Cooper Nuclear Station.

If you disagree with a cross-cutting aspect assignment 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 U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region IV; and the NRC resident inspector at the Cooper Nuclear Station.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with 10 CFR 2.390, "Public Inspections, Exemptions, and Requests for Withholding."

Sincerely,

/RA/

Gregory G. Warnick, Branch Chief
Project Branch C
Division of Reactor Projects

Docket No. 50-298
License No. DPR-46

Enclosure:
Inspection Report 05000298/2016004
w/ Attachments:
1. Supplemental Information
2. Request for Information for the
   Occupational Radiation Safety Inspection

COOPER NUCLEAR STATION – NRC INTEGRATED INSPECTION REPORT 05000298/2016004 AND EXERCISE OF ENFORCEMENT DISCRETION

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-298
License:	DPR-46
Report:	05000298/2016004
Licensee:	Nebraska Public Power District
Facility:	Cooper Nuclear Station
Location:	72676 648A Ave Brownville, NE
Dates:	October 1 through December 31, 2016
Inspectors:	P. Voss, Senior Resident Inspector C. Henderson, Resident Inspector P. Elkmann, Sr. Emergency Preparedness Inspector J. O'Donnell, CHP, Health Physicist M. Phalen, Senior Health Physicist
Approved By:	Gregory G. Warnick Chief, Project Branch C Division of Reactor Projects

Enclosure

SUMMARY

IR 05000298/2016004; 10/01/2016 – 12/31/2016; Cooper Nuclear Station; Maint. Risk Asses. & Emergent Work Control, Emergency Action Level & Emergency Plan Changes, and Follow-up of Events & Notices of Enforcement Discretion.

The inspection activities described in this report were performed between October 1 and December 31, 2016, by the resident inspectors at the Cooper Nuclear Station and inspectors from the NRC's Region IV office. Four findings of very low safety significance (Green) are documented in this report. All of these findings involved violations of NRC requirements. The significance of inspection findings is indicated by their color (Green, White, Yellow, or Red), which is determined using Inspection Manual Chapter 0609, "Significance Determination Process," dated April 29, 2015. Their cross-cutting aspects are determined using Inspection Manual Chapter 0310, "Aspects within the Cross-Cutting Areas," dated December 4, 2014. Violations of NRC requirements are dispositioned in accordance with the NRC Enforcement Policy. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," dated July 2016.

Cornerstone: Initiating Events

- Green. The inspectors reviewed a self-revealed, non-cited violation of Technical Specification 5.4.1.a for the licensee's failure to maintain Station Procedure 2.2.56, "Main Steam System," Revision 49, to prevent a main steam line high flow Group 1 primary containment isolation signal when opening an inboard main steam isolation valve. Specifically, the licensee failed to maintain Station Procedure 2.2.56 with adequate differential pressure limits for reopening closed main steam isolation valves during plant shutdown, which caused the unexpected closure of all the open main steam isolation and drain valves during the plant cooldown process. This resulted in a loss of the main steam line decay heat removal path, which caused reactor coolant system pressure and temperature to increase by approximately 13 psig and 3 degrees Fahrenheit, respectively, during the event. The immediate corrective actions were to reset the Group 1 isolation signal and open the main steam line drain valves to recommence plant cooldown. The licensee entered this deficiency into the corrective action program as Condition Report CR-CNS-2016-05835, and the licensee initiated an apparent cause evaluation to investigate this condition.

  The licensee's failure to maintain Station Procedure 2.2.56 to prevent a main steam line high flow Group 1 isolation signal when opening an inboard main steam isolation valve, in violation of Technical Specification 5.4.1.a, was a performance deficiency. The performance deficiency was determined to be more than minor, and therefore a finding, because it was associated with the procedural quality attribute of the Initiating Events Cornerstone and affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown. Specifically, the Group 1 isolation signal closed the main steam line drain valves, which resulted in a loss of the main steam line decay heat removal path and caused reactor coolant system pressure and temperature to increase. The inspectors determined Inspection Manual Chapter 0609, Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process Phase 1 Initial Screening and Characterization of Findings," dated May 9, 2014, was not applicable because plant temperature and pressure were not within the normal residual heat removal/decay heat removal system operating parameters. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings

At-Power," dated June 19, 2012, the inspectors determined that the finding screened as having very low safety significance (Green) because it did not cause both a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of a trip to a stable shutdown condition. A cross-cutting aspect was not assigned to this finding because the performance deficiency occurred in 1988 when the licensee changed the procedural limits for differential pressure across the main steam isolation valves when reopening them, and therefore, was not indicative of current licensee performance. (Section 4OA3)

Cornerstone: Barrier Integrity

- Green. The inspectors reviewed a self-revealed, non-cited violation of Technical Specification 3.6.4.2, "Secondary Containment Isolation Valves," for the licensee's failure to maintain secondary containment isolation valve HV-AOV-265 operable as a result of erecting scaffolding that interfered with valve operation. Specifically, between June 29, 2016, and September 14, 2016, the licensee erected scaffolding in close proximity of valve HV-AOV-265, such that, during valve stroking, the scaffolding would pinch the actuator air line and prevent the valve from closing, rendering the valve inoperable for approximately 10 weeks. This resulted in the licensee's need to reduce power to approximately 50 percent in order to comply with technical specifications upon discovery. Immediate corrective actions included removal of the scaffolding, replacement of the pinched air line, and restoration of the valve to operable status. The licensee entered this deficiency into the corrective action program as Condition Report CR-CNS-2016-05608 and initiated a root cause evaluation to investigate this condition.

  The licensee's failure to implement Procedure 7.0.7, "Scaffolding Construction and Control," Revision 34, to ensure scaffolding did not adversely affect plant equipment, in violation of Technical Specification 3.6.4.2, was a performance deficiency. The performance deficiency was determined to be more than minor, and therefore a finding, because it was associated with the structure, system, and component and barrier performance attribute of the Barrier Integrity Cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (secondary containment) protect the public from radionuclide releases caused by accidents or events. Specifically, the improperly erected scaffolding prevented the operation of a secondary containment isolation valve, rendering it inoperable for approximately 10 weeks. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, the inspectors determined that the finding had very low safety significance (Green) because it only represented a degradation of the radiological barrier function provided for the control room, reactor building, spent fuel pool building, or standby gas treatment system. The finding had a cross-cutting aspect in the area of human performance associated with resources. Specifically, the licensee failed to ensure that personnel, equipment, procedures, and other resources were available and adequate to support nuclear safety [H.1]. (Section 4OA3)

Cornerstone: Occupational Radiation Safety

- Green. The inspectors reviewed a self-revealed, non-cited violation of Technical Specification 5.4.1.a for the licensee's failure to ensure sufficient radiological work controls were in place when the reactor pressure vessel moisture separator was installed during vessel reassembly. Specifically, the licensee failed to maintain sufficient detail in Station Procedure 7.4Reassembly, "Reactor Vessel Reassembly," Revision 13, to ensure that the

moisture separator had adequate water shielding during lifts, such that radiation fields were appropriately controlled. The licensee took immediate corrective action to ensure re-submergence of the radiologically significant sections of the moisture separator and restore the requisite water shielding, thereby restoring ambient refuel floor radiological conditions. The licensee entered this deficiency into the corrective action program as Condition Report CR-CNS-2016-07552.

The licensee's failure to ensure sufficient radiological work controls were in place when the reactor pressure vessel moisture separator was lifted during vessel reassembly, in violation of Technical Specification 5.4.1.a, was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it was associated with the program and process attribute of the Occupational Radiation Safety Cornerstone and adversely affected the cornerstone objective to ensure adequate protection of worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. Specifically, the failure to have sufficient procedural guidance to maintain adequate water shielding on the moisture separator resulted in unanticipated elevated dose rates on the refuel floor and unplanned radiological exposures to workers in the immediate work area. Using Inspection Manual Chapter 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," dated August 19, 2008, the inspectors determined that the violation had very low safety significance (Green) because: (1) it was not an as low as reasonably achievable (ALARA) finding; (2) there was no overexposure; (3) there was no substantial potential for an overexposure; and (4) the ability to assess dose was not compromised. The inspectors determined that the finding had a cross-cutting aspect in the area of human performance associated with avoiding complacency. Specifically, the licensee failed to recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes and failed to implement appropriate error reduction tools [H.12]. (Section 1R13)

Cornerstone: Emergency Preparedness

- Green. The inspectors identified a non-cited violation of 10 CFR 50.54(q)(3) for the licensee's failure to perform an analysis demonstrating that proposed emergency plan implementing procedure changes did not reduce the effectiveness of the emergency plan. Specifically, the licensee's 50.54(q) evaluation failed to demonstrate that Emergency Plan Implementing Procedure 5.7.1, "Emergency Classification," Revision 54, changes, associated with Emergency Action Level SG2.1 and the fission product barrier matrix, did not result in a reduction in effectiveness. The corrective action was to revise 10 CFR 50.54(q) Evaluation 2016-011 to provide additional information about the ability of emergency coordinators in the Technical Support Center and Emergency Operations Facility to classify using the revised emergency action levels. The licensee entered this deficiency into the corrective action program as Condition Report CR-CNS-2016-05697.

  The licensee's failure to perform an analysis demonstrating that proposed changes to Emergency Plan Implementing Procedure 5.7.1 did not reduce the effectiveness of the emergency plan, in violation of 10 CFR 50.54(q)(3), was a performance deficiency. The finding was more than minor, and therefore a finding, because it was associated with the procedure quality attribute (emergency action level changes) of the Emergency Preparedness Cornerstone and adversely affected the cornerstone objective to ensure that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. Specifically, the licensee's

ability to ensure that adequate measures are taken to protect the health and safety of the public is degraded if the licensee performs inadequate analyses of the effects of changes to the emergency plan. Using Inspection Manual Chapter 0609, Appendix B, “Emergency Preparedness Significance Determination Process,” Attachment 2, dated September 22, 2015, the inspectors determined that the finding was of very low safety significance (Green) because it was not associated with a risk-significant planning standard function or a planning standard function. This finding had a cross-cutting aspect in the area of human performance, associated with change management, because the licensee failed to use a systematic process for evaluating and implementing changes so that nuclear safety remains the overriding priority. Specifically, the licensee did not have an adequate understanding of the licensing basis for making changes to emergency action levels [H.3]. (Section 1EP4)

PLANT STATUS

The Cooper Nuclear Station began the inspection period in a shutdown status for Refueling Outage 29, which started on September 24, 2016. On November 6, 2016, the station commenced reactor startup and the reactor was made critical. On November 8, 2016, the station synchronized the main generator to the grid and began power ascension. The plant returned to full power on November 14, 2016, and remained there for the rest of the inspection period.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment (71111.04)

Partial Walk-Down

a. Inspection Scope

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

- November 21, 2016, spent fuel pool cooling system
- November 28, 2016, reactor core isolation cooling
- November 30, 2016, main steam isolation valves

The inspectors reviewed the licensee’s procedures and system design information to determine the correct lineup for the systems. They visually verified that critical portions of the systems were correctly aligned for the existing plant configuration.

These activities constituted three partial system walk-down samples, as defined in Inspection Procedure 71111.04.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

Quarterly Inspection

a. Inspection Scope

The inspectors evaluated the licensee’s fire protection program for operational status and material condition. The inspectors focused their inspection on four plant areas important to safety:

- October 15, 2016, reactor building 931 feet, reactor equipment cooling heat exchangers, Fire Area RB-M, Zone 3C

- October 31, 2016, fire impairment for battery room A, door BLDG-DOOR-H111, Fire Area CB-A-1, Zone 8E

- October 31, 2016, fire impairment for cable spreading room, door BLDG-DOOR-H200, Fire Area CB-D, Zone 9A

- November 5, 2016, drywell, Fire Area Drywell

For each area, the inspectors evaluated the fire plan against defined hazards and defense-in-depth features in the licensee's fire protection program. The inspectors evaluated control of transient combustibles and ignition sources, fire detection and suppression systems, manual firefighting equipment and capability, passive fire protection features, and compensatory measures for degraded conditions.

These activities constituted four quarterly inspection samples, as defined in Inspection Procedure 71111.05.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program and Licensed Operator Performance (71111.11)

.1 Review of Licensed Operator Requalification

a. Inspection Scope

On November 22, 2016, the inspectors observed a simulator training for an operating crew. The inspectors assessed the performance of the operators and the evaluators' critique of their performance. The inspectors also assessed the modeling and performance of the simulator during the training activity.

These activities constituted completion of one quarterly licensed operator requalification program sample, as defined in Inspection Procedure 71111.11.

b. Findings

No findings were identified.

.2 Review of Licensed Operator Performance

a. Inspection Scope

On November 6 and November 8, 2016, the inspectors observed the performance of on-shift licensed operators in the plant's main control room. At the time of the observations, the plant was in a period of heightened risk due to reactor plant startup from Refueling Outage 29.

In addition, the inspectors assessed the operators' adherence to plant procedures, including conduct of operations procedure and other operations department policies.

These activities constituted completion of one quarterly licensed operator performance sample, as defined in Inspection Procedure 71111.11.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed two instances of degraded performance or condition of safety-related structures, systems, and components (SSCs):

- November 17, 2016, reactor instrument excess flow check valves and primary containment isolation valves
- November 30, 2016, main steam isolation valves exceeded local leak-rate test limits

The inspectors reviewed the extent of condition of possible common cause SSC failures and evaluated the adequacy of the licensee's corrective actions. The inspectors reviewed the licensee's work practices to evaluate whether these may have played a role in the degradation of the SSCs. The inspectors assessed the licensee's characterization of the degradation in accordance with 10 CFR 50.65 (the Maintenance Rule), and verified that the licensee was appropriately tracking degraded performance and conditions in accordance with the Maintenance Rule.

These activities constituted completion of two maintenance effectiveness samples, as defined in Inspection Procedure 71111.12.

b. Findings

No findings were identified.

.2 Quality Control

a. Inspection Scope

On December 16, 2016, the inspectors reviewed the licensee's quality control activities through: (1) a review of parts installed in the residual heat removal system that were purchased as commercial-grade parts but were dedicated prior to installation in a quality-grade application; (2) a review of the licensee's control of quality parts during maintenance; (3) a review of whether quality control verifications were properly specified in accordance with the licensee's Quality Assurance Program, and were implemented as specified, during work associated with work activities.

These activities constituted completion of one quality control sample, as defined in Inspection Procedure 71111.12.

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors reviewed six risk assessments performed by the licensee prior to changes in plant configuration and the risk management actions taken by the licensee in response to elevated risk:

- October 21, 2016, outage risk control, 4160 V bus out of service, Division 1
- October 27, 2016, backfeeding of the normal station transformer
- November 3, 2016, operations with a potential for draining the reactor vessel
- November 22, 2016, service water and emergency diesel generator availability during service water emergency sparger flow testing, Division 2
- November 22, 2016, primary containment isolation system relay work and loss of shutdown cooling
- November 22, 2016, moisture separator lift during reactor pressure vessel reassembly

The inspectors verified that these risk assessments were performed timely and in accordance with the requirements of 10 CFR 50.65 (the Maintenance Rule) and plant procedures. The inspectors reviewed the accuracy and completeness of the licensee's risk assessments and verified that the licensee implemented appropriate risk management actions based on the result of the assessments.

These activities constituted completion of six maintenance risk assessment inspection samples, as defined in Inspection Procedure 71111.13.

b. Findings

Introduction. The inspectors reviewed a self-revealed, Green, non-cited violation of Technical Specification 5.4.1.a for the licensee's failure to ensure sufficient radiological work controls were in place when the reactor pressure vessel moisture separator was installed during vessel reassembly. Specifically, the licensee failed to maintain sufficient detail in Station Procedure 7.4Reassembly, "Reactor Vessel Reassembly," Revision 13, to ensure that the moisture separator had adequate water shielding during lifts, such that radiation fields were appropriately controlled.

Description. On October 26, 2016, during a planned refueling outage, the inspectors observed portions of the licensee's reactor pressure vessel (RPV) reassembly work activities on the refuel floor. The refuel floor was controlled as a high radiation area during the RPV reassembly process. Radiological control of work on the refuel floor was maintained in accordance with radiation work permit (RWP) 2016-545 and station specific work management procedures. During reassembly, the licensee performed a lift

of the moisture separator (MS). The MS is a highly irradiated RPV internal component. Prior to the lift, the MS was stored completely submerged in water in an equipment storage pit, next to the RPV cavity.

The licensee performed the MS lift in accordance with Station Procedure 7.4Reassembly, "Reactor Vessel Reassembly," Revision 13, which provided instructions for lifting the MS from the equipment storage pit and placing it into the RPV. This procedure contained several cautions for performing the lift while minimizing radiological doses to workers performing the lift. Caution 1, for Step 7.2.2, stated, "Steam separator lifted only high enough to facilitate transportation to reactor vessel. Underwater camera system shall be used to confirm height of separator in relation to weir wall and RPV studs." Caution 2 stated, "Ensure rate of lift continuously monitored by radiation protection personnel and sufficient water level maintained in cavity as high radiation levels could result." In addition, a warning for the same section stated, "Steam separator is highly irradiated, top of steam separator shall not breach water surface. When separator near surface of pool, high radiation levels may be encountered."

Additional radiological controls were specified in the RWP. However, these controls were based on historical data from previous successful MS transfers and not specific MS survey data. The licensee established stop work criteria for work area dose rates at 900 mrem/hr. Stop work criteria for individual workers' electronic dosimeters (EDs) were set at 500 mrem/hr for dose rate and 150 mrem for accumulated dose.

The licensee briefed workers in accordance with technical specification requirements for high radiation area access control requirements with expected dose rates of approximately 600 mrem/hr during movement of the MS. The workers were also informed of contingencies for the possibility of dose rates of up to 900 mrem/hr (i.e. the RWP stop work criteria). Actual dose rates encountered during the MS moves were higher than expected or briefed. Radiological surveys indicated dose rates of 3500 mrem/hr and 1500 mrem/hr at the two dryer separator pit railings and two workers received ED dose rate alarms during the lift. Upon receipt of ED dose rate alarms and indications of elevated dose rates on radiation survey meters, radiation protection personnel notified workers to stop the lift. The MS was then lowered back to its stand in the dryer separator pit, which caused dose rates to return to normal levels. The inspectors determined that appropriate high radiation area controls remained in place in the work area during this event.

The inspectors determined that the MS was lifted higher in the pit than was necessary or intended, decreasing the water shielding the MS during the lift. The reduced shielding resulted in increased radiation fields and elevated work area dose rates in the vicinity of the MS on the refuel floor. The two individuals that received dose rate alarms each received approximately 10 mrem for the period of time that the MS was not appropriately shielded. The inspectors concluded that the intensity of the radiation fields created during this work evolution and the resultant workers' radiological exposures were primarily a consequence of the radiation emitting from the unshielded MS and not the result of a planned work sequence. The inspectors also observed that there was no overexposure in excess of NRC limits, and there was no substantial potential for an overexposure.

During review of Station Procedure 7.4Reassembly, the inspectors determined that the procedure contained insufficient guidance to ensure that controls were in place to

maintain adequate water shielding above the MS during the move. Specifically, the procedure did not specify limitations on the height of the MS during the lift, nor did it ensure that a specific volume or height of water was maintained above the MS. Procedural guidance focused the use of underwater cameras on ensuring that the MS was lifted high enough to clear the weir wall at the bottom of the pit. As a result, during the lift, workers placed significant focus on lifting the MS high enough to clear the weir wall and placed insufficient focus on maintaining an appropriate depth of water above the MS for shielding. The inspectors noted that given the radiological controls in place, coupled with the lack of specificity on water shielding levels for the MS lift, the licensee should have created more precise controls through revision of Procedure 7.4Reassembly or use of detailed work order instructions. Additionally, the licensee should have highlighted these controls during the pre-job briefing process and incorporated other human error prevention techniques. The inspectors concluded that the licensee had failed to recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes and failed to implement appropriate error reduction tools. The licensee initiated action to revise the procedures for vessel disassembly and reassembly to include more precise MS lift controls.

Analysis. The licensee's failure to ensure sufficient radiological work controls were in place when the reactor pressure vessel moisture separator was installed during vessel reassembly, in violation of Technical Specification 5.4.1.a, was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it was associated with the program and process attribute of the Occupational Radiation Safety Cornerstone and adversely affected the cornerstone objective to ensure adequate protection of worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. Specifically, the failure to have sufficient procedural guidance to maintain adequate water shielding on the moisture separator resulted in unanticipated elevated dose rates on the refuel floor and unplanned radiological exposures to workers in the immediate work area. Using Inspection Manual Chapter 0609, Appendix C, “Occupational Radiation Safety Significance Determination Process,” dated August 19, 2008, the inspectors determined that the violation was of very low safety significance (Green) because: (1) it was not an as low as reasonably achievable (ALARA) finding; (2) there was no overexposure; (3) there was no substantial potential for an overexposure; and (4) the ability to assess dose was not compromised. The inspectors determined that the finding had a cross-cutting aspect in the area of human performance associated with avoiding complacency. Specifically, the licensee failed to recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes and failed to implement appropriate error reduction tools [H.12].

Enforcement. Technical Specification 5.4.1.a requires, in part, that written procedures be established, implemented, and maintained covering the applicable procedures recommended in Appendix A to Regulatory Guide 1.33, “Quality Assurance Program Requirements,” Revision 2, February 1978. Regulatory Guide 1.33, Appendix A, Section 9.d.6, requires procedures for performing maintenance associated with removal of the reactor head, including reactor vessel disassembly and reassembly. The licensee established Procedure 7.4Reassembly, “Reactor Vessel Reassembly,” Revision 13, for performing maintenance associated with removal of the reactor head, including reactor vessel disassembly and reassembly. Contrary to the above, until October 26, 2016, the licensee failed to maintain a procedure for performing maintenance associated with removal of the reactor head, including reactor vessel disassembly and reassembly.

Specifically, the licensee failed to maintain Procedure 7.4Reassembly with sufficient detail to ensure the moisture separator had an adequate water shield during lifts, such that radiation fields were appropriately controlled. This resulted in unanticipated elevated dose rates on the refuel floor during a lift and unplanned radiological exposures to workers in the immediate work area. The licensee implemented immediate corrective action to ensure re-submergence of the radiologically significant sections of the moisture separator and restore the requisite water shielding, thereby restoring ambient refuel floor radiological conditions. The licensee also initiated actions to revise the procedures for vessel disassembly and reassembly to include more precise MS lift controls. Because this violation was of very low safety significance (Green) and was entered into the licensee's corrective action program as Condition Report CR-CNS-2016-07552, this violation is being treated as a non-cited violation in accordance with Section 2.3.2.a of the NRC Enforcement Policy. (NCV 05000298/2016004-01, "Failure to Maintain Reactor Vessel Assembly Procedure to Ensure Adequate Moisture Separator Shielding")

1R15 Operability Determinations and Functionality Assessments (71111.15)

a. Inspection Scope

The inspectors reviewed four operability determinations that the licensee performed for degraded or nonconforming structures, systems, or components (SSCs):

- November 2, 2016, operability determination of reactor equipment cooling pump B trip
- November 10, 2016, operability determination of incorrect test resistors installed for reactor core isolation cooling beginning of cycle test
- December 12, 2016, operability determination of control room emergency filtration system fan SF-C-1B
- December 31, 2016, operability determination of the residual heat removal service water piping and reactor equipment cooling heat exchanger A service water backwash line below design minimum wall thickness

The inspectors reviewed the timeliness and technical adequacy of the licensee's evaluations. Where the licensee determined the degraded SSC to be operable, the inspectors verified that the licensee's compensatory measures were appropriate to provide reasonable assurance of operability. The inspectors verified that the licensee had considered the effect of other degraded conditions on the operability of the degraded SSC.

These activities constituted completion of four operability review samples, as defined in Inspection Procedure 71111.15.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

a. Inspection Scope

On December 2, 2016, the inspectors reviewed a permanent plant modification to the startup station service transformer.

The inspectors reviewed the design and implementation of the modification. The inspectors verified that work activities involved in implementing the modification did not adversely impact operator actions that may be required in response to an emergency or other unplanned event. The inspectors verified that post-modification testing was adequate to establish the operability and functionality of the structure, system, or component as modified.

These activities constituted completion of one sample of permanent modifications, as defined in Inspection Procedure 71111.18.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed eight post-maintenance testing activities that affected risk-significant structures, systems, or components (SSCs):

- October 20, 2016, residual heat removal shutdown cooling, Loop A, primary containment isolation relay K59 relay coil replacement
- October 21, 2016, steam dryer repair
- November 3, 2016, reactor pressure vessel/recirculation pump replacement system pressure test
- November 8, 2016, startup station service transformer replacement
- November 8, 2016, main steam isolation valve re-packing and repairs
- November 8, 2016, service water maintenance and repairs, Division II
- November 17, 2016, primary containment isolation relay replacements
- November 22, 2016, reactor core isolation cooling auxiliary lube oil cooler pressure control valve RCIC-AO-PCV32 replacement

The inspectors reviewed licensing- and design-basis documents for the SSCs and the maintenance and post-maintenance test procedures. The inspectors observed the performance of the post-maintenance tests to verify that the licensee performed the tests in accordance with approved procedures, satisfied the established acceptance criteria, and restored the operability of the affected SSCs.

These activities constituted completion of eight post-maintenance testing inspection samples, as defined in Inspection Procedure 71111.19.

b. Findings

No findings were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

During the station's refueling outage that concluded on November 8, 2016, the inspectors evaluated the licensee's outage activities. The inspectors verified that the licensee considered risk in developing and implementing the outage plan, appropriately managed personnel fatigue, and developed mitigation strategies for losses of key safety functions. This verification included the following:

- Review of the licensee's outage plan prior to the outage
- Review and verification of the licensee's fatigue management activities
- Monitoring of shut-down and cool-down activities
- Verification that the licensee maintained defense-in-depth during outage activities
- Observation and review of reduced-inventory
- Observation and review of fuel handling activities
- Monitoring of heat-up and startup activities

These activities constituted completion of one refueling outage sample, as defined in Inspection Procedure 71111.20.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed three risk-significant surveillance tests and reviewed test results to verify that these tests adequately demonstrated that the structures, systems, and components (SSCs) were capable of performing their safety functions:

- October 27, 2016, emergency diesel generator undervoltage logic functional, load shedding, and sequential loading test, Division I

- November 10, 2016, secondary containment leak test

- November 16, 2016, Group 6 functional surveillance testing for reactor building ventilation radiation monitors

The inspectors verified that these tests met technical specification requirements, that the licensee performed the tests in accordance with their procedures, and that the results of

the test satisfied appropriate acceptance criteria. The inspectors verified that the licensee restored the operability of the affected SSCs following testing.

These activities constituted completion of three surveillance testing inspection samples, as defined in Inspection Procedure 71111.22.

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level (EAL) and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspector performed an in-office and on-site review of Emergency Plan Implementing Procedure 5.7.1, "Emergency Classification," Revision 54, submitted to the NRC on April 27, 2016. This revision revised EAL SG2.1 and fission product barrier matrix thresholds, 8 and 10, by changing the technical basis document. Specifically, the revision changed the basis for determining when reactor vessel water level could not be restored and maintained.

This revision was compared to its previous revision, to the criteria of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, to Nuclear Energy Institute Report 99-01, "Emergency Action Level Methodology," Revision 5, and to the standards in 10 CFR 50.47(b) to determine if the revision adequately implemented the requirements of 10 CFR 50.54(q)(3) and 50.54(q)(4).

These activities constituted completion of one EAL and emergency plan changes sample, as defined in Inspection Procedure 71114.04.

b. Findings

Introduction. The inspectors identified a Green, non-cited violation of 10 CFR 50.54(q)(3) for the licensee's failure to perform an analysis demonstrating that proposed emergency plan implementing procedure changes did not reduce the effectiveness of the emergency plan. Specifically, the licensee's 50.54(q) evaluation failed to demonstrate that Emergency Plan Implementing Procedure 5.7.1, "Emergency Classification," Revision 54, changes, associated with EAL SG2.1 and the fission product barrier matrix, did not result in a reduction in effectiveness.

Description. On March 28, 2016, the licensee implemented Revision 54 to Emergency Plan Implementing Procedure (EPIP) 5.7.1, "Emergency Classification," using their authority to make changes to the site emergency plan based on determining that the changes did not reduce the effectiveness of the site emergency plan and did not require prior approval from the NRC. The licensee submitted a summary of their regulatory analysis of the impact of changes to EPIP 5.7.1 in a letter to the NRC, dated April 27, 2016. The inspectors reviewed the changes implemented by EPIP 5.7.1, Revision 54, and the associated 50.54(q) Evaluation 2016-011, dated January 27, 2016.

During this review, the inspectors noted that EPIP 5.7.1, Revision 54, revised EAL SG2.1 and fission product barrier matrix thresholds, 8 and 10, by changing the technical basis document. Specifically, the revision changed the basis for determining when reactor pressure vessel (RPV) water level could not be restored and maintained. In the previous revision of the basis, implemented in Revision 46 in 2012, the licensee changed EAL SG2.1 to read: “reactor vessel level cannot be restored or maintained when level is less than -158 inches for greater than or equal to 15 minutes.” The licensee’s regulatory impact analysis provided the following basis for implementing the 15-minute requirement:

> “This change provides clarification for conditions that are evaluated to determine when the "cannot be restored and maintained" threshold criteria are violated. The criteria are afforded a great deal of latitude when evaluating plant conditions and do not require immediate declaration just because the parameter being evaluated (RPV level) has crossed the threshold value. This criteria requires an evaluation of injection systems and their present and future capability including conditions following pressure reduction…This change adds amplifying discussion of the assessment of injection system capability that provides insight to the ultimate ability to restore and maintain these values…<u>A time limit to be considered for the restoration of the threshold values is also being incorporated to further restrict the restoration effort prior to the decision or declaration</u>…Because these changes provide more insight into the base concern, adequate core cooling, these changes do not reduce the effectiveness of Emergency Preparedness Function 7…”

In contrast, EPIP 5.7.1, Revision 54, updated the basis document to remove the 15-minute time clock for restoring and maintaining RPV level and replaced it with the statement: “Determination of inability to restore and maintain RPV level is based upon actions driven by EOPs to restore level.”

The inspector reviewed the licensee’s regulatory impact analysis for Revision 54 to EPIP 5.7.1. The regulatory impact analysis provided the following basis for removing the 15-minute requirement:

> “Revision 46 of Procedure 5.7.1 included clarification for conditions that are to be evaluated when determining when “cannot be restored and maintained” threshold criteria are violated…However, a 15-minute time clock was also added to define what a reasonable amount of time would be for determining that the limit cannot be attained. Fifteen minutes was chosen due its consistency with other delay times incorporated into EAL criteria…<u>Neither the EOPs or NEI 99-01 Technical Basis, include a time clock philosophy associated with EALs or Fission Product Barrier criteria that rely on the criteria “cannot be restored or maintained.” Due to having no solid technical basis for the time clock</u>,…Operations crews train utilizing EOPs for determining “restore and maintain,” and the previously added clarification for conditions to be evaluated for determining “cannot be restored and maintained” threshold will remain in place, <u>it has been concluded that the 15-minute clock should be removed from CNS Technical Basis in Procedure 5.7.1</u>…Based on this 10 CFR 50.54(q) Evaluation, it is concluded that the proposed change does not reduce the effectiveness of the CNS Emergency Plan…”

The inspector determined that the regulatory impact analysis for EPIP 5.7.1, Revision 54, was accurate in stating that the analysis for Revision 46 did not provide a technical basis for a 15-minute requirement to determine whether RPV level could be restored and maintained above -158 inches, and that the time requirement did not derive from NEI 99-01, "Emergency Action Level Methodology," Revision 5, dated February 2008. However, the inspector concluded that the 50.54(q) analysis did not address how the removal of the time requirement previously determined to be necessary would ensure that a classification decision was made in a timely manner. The Revision 54 analysis based the removal of the time requirement on emergency operating procedure training for licensed operators located in the control room, but the inspector concluded that the analysis failed to consider that the EALs are also implemented by emergency directors not located in the control room nor trained on the emergency operating procedures.

The inspector concluded that the lack of a NEI 99-01 technical basis for the previous changes made to EPIP 5.7.1 was not a sufficient justification for removing those changes without consideration of the licensing basis for the previous changes. Therefore, the inspector concluded that the analysis in 50.54(q) Evaluation 2016-011 did not demonstrate that removing the time clock added in Revision 46 was not a reduction in effectiveness.

<u>Analysis</u>. The licensee's failure to perform an analysis demonstrating that proposed changes to EPIP 5.7.1 did not reduce the effectiveness of the emergency plan, in violation of 10 CFR 50.54(q)(3), was a performance deficiency. The finding was more than minor, and therefore a finding, because it was associated with the procedure quality attribute (emergency action level changes) of the Emergency Preparedness Cornerstone and adversely affected the cornerstone objective to ensure that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. Specifically, the licensee's ability to ensure that adequate measures are taken to protect the health and safety of the public is degraded if the licensee performs inadequate analyses of the effects of changes to the emergency plan. Using Inspection Manual Chapter 0609, Appendix B, "Emergency Preparedness Significance Determination Process," Attachment 2, dated September 22, 2015, the inspectors determined that the finding was of very low safety significance (Green) because it was not associated with a risk-significant planning standard function or a planning standard function. This finding had a cross-cutting aspect in the area of human performance, associated with change management, because the licensee failed to use a systematic process for evaluating and implementing changes so that nuclear safety remains the overriding priority. Specifically, the licensee did not have an adequate understanding of the licensing basis for making changes to EALs [H.3].

<u>Enforcement</u>. Title 10 CFR 50.54(q)(3) requires, in part, that a licensee may make changes to its emergency plan without NRC approval only if the licensee performs an analysis demonstrating that the changes do not reduce the effectiveness of the plan. Contrary to the above, from March 28, 2016, to October 19, 2016, the licensee made changes to its emergency plan without NRC approval, but failed to perform an analysis demonstrating that the changes did not reduce the effectiveness of the plan. Specifically, the licensee's 10 CFR 50.54(q) Evaluation 2016-011 failed to demonstrate that the EPIP 5.7.1, "Emergency Classification," Revision 54, changes associated with EAL SG2.1 and the fission product barrier matrix did not result in a reduction in

effectiveness. The corrective action was to revise the 10 CFR 50.54(q) Evaluation 2016-011 to provide additional information about the ability of emergency coordinators in the Technical Support Center and Emergency Operations Facility to classify using the revised emergency action levels. Because this violation was of very low safety significance (Green) and was entered into the licensee's corrective action program as Condition Report CR-CNS-2016-05697, this violation is being treated as a non-cited violation in accordance with Section 2.3.2.a of the NRC Enforcement Policy. (NCV 05000298/2016004-02, "Failure of an Analysis to Demonstrate that Changes Did Not Reduce the Effectiveness of the Emergency Plan")

2. RADIATION SAFETY

Cornerstones: Public Radiation Safety and Occupational Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

a. Inspection Scope

The inspectors evaluated the licensee's performance in assessing the radiological hazards in the workplace associated with licensed activities. The inspectors assessed the licensee's implementation of appropriate radiation monitoring and exposure control measures for both individual and collective exposures. During the inspection, the inspectors interviewed licensee personnel, walked down various areas in the plant, performed independent radiation dose rate measurements, and observed postings and physical controls. The inspectors reviewed licensee performance in the following areas:

- Radiological hazard assessment, including a review of the plant's radiological source terms and associated radiological hazards. The inspectors also reviewed the licensee's radiological survey program to determine whether radiological hazards were properly identified for routine and nonroutine activities and assessed for changes in plant operations.

- Instructions to workers, including radiation work permit requirements and restrictions, actions for electronic dosimeter alarms, changing radiological condition, and radioactive material container labeling.

- Contamination and radioactive material control, including release of potentially contaminated material from the radiologically controlled area, radiological survey performance, radiation instrument sensitivities, material control and release criteria, and control and accountability of sealed radioactive sources.

- Radiological hazards control and work coverage. During walk-downs of the facility and job performance observations, the inspectors evaluated ambient radiological conditions, radiological postings, adequacy of radiological controls, radiation protection job coverage, and contamination controls. The inspectors also evaluated dosimetry selection and placement as well as the use of dosimetry in areas with significant dose rate gradients. The inspectors examined the licensee's controls for items stored in the spent fuel pool and evaluated airborne radioactivity controls and monitoring.

- High radiation area and very high radiation area controls. During plant walk-downs, the inspectors verified the adequacy of posting and physical controls including areas of the plant with the potential to become risk-significant high radiation areas.

- Radiation worker performance and radiation protection technician proficiency with respect to radiation protection work requirements. The inspectors determined if workers were aware of significant radiological conditions in their workplace, radiation work permit controls/limits in place, and electronic dosimeter dose and dose rate set points. The inspectors observed radiation protection technician job performance including the performance of radiation surveys.

- Problem identification and resolution for radiological hazard assessment and exposure controls. The inspectors reviewed audits, self-assessments, and corrective action program documents to verify problems were being identified and properly addressed for resolution.

These activities constituted completion of the seven required samples of the radiological hazard assessment and exposure control program, as defined in Inspection Procedure 71124.01.

b. Findings

No findings were identified.

2RS2 Occupational ALARA Planning and Controls (71124.02)

a. Inspection Scope

The inspectors assessed licensee performance with respect to maintaining individual and collective radiation exposures as low as is reasonably achievable (ALARA). The inspectors performed this portion of the attachment during the refueling outage, in order to directly observe the licensee’s ALARA process activities including planning, implementation of radiological work controls, execution of work activities, and ALARA review of work-in-progress. During the inspection, the inspectors interviewed licensee personnel, reviewed licensee documents, and evaluated licensee performance in the following areas:

- Implementation of ALARA and radiological work controls. The inspectors observed pre-job briefings, reviewed planned radiological administrative, operational, and engineering controls, and compared the planned controls to field activities.

- Radiation worker and radiation protection technician performance during work activities performed in radiation areas, airborne radioactivity areas, or high radiation areas.

- Problem identification and resolution for ALARA and radiological work controls. The inspectors reviewed audits, self-assessments, and corrective action program documents to verify problems were being identified and properly addressed for resolution.

These activities constituted completion of three of the five required samples of the occupational ALARA planning and controls program, as defined in Inspection Procedure 71124.02.

b. <u>Findings</u>

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 <u>Mitigating System Performance Index: Emergency AC Power Systems (MS06), High Pressure Injection Systems (MS07), Heat Removal Systems (MS08), Residual Heat Removal Systems (MS09), and Cooling Water Systems (MS10)</u>

a. <u>Inspection Scope</u>

The inspectors reviewed the licensee's mitigating system performance index data for the period of October 1, 2015, through September 30, 2016, to verify the accuracy and completeness of the reported data. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, to determine the accuracy of the reported data.

These activities constituted verification of the mitigating system performance index for emergency ac power systems, high pressure injection systems, heat removal systems, residual heat removal systems, and cooling water systems, as defined in Inspection Procedure 71151.

b. <u>Findings</u>

No findings were identified.

.2 <u>Occupational Exposure Control Effectiveness (OR01)</u>

a. <u>Inspection Scope</u>

The inspectors verified that there were no unplanned exposures or losses of radiological control over locked high radiation areas and very high radiation areas during the period of April 1, 2015, through September 30, 2016. The inspectors reviewed a sample of radiologically controlled area exit transactions showing exposures greater than 100 millirem. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, to determine the accuracy of the reported data.

These activities constituted verification of the occupational exposure control effectiveness performance indicator, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.3 Radiological Effluent Technical Specifications (RETS)/Offsite Dose Calculation Manual (ODCM) Radiological Effluent Occurrences (PR01)

a. Inspection Scope

The inspectors reviewed corrective action program records for liquid or gaseous effluent releases that occurred between April 1, 2015, and September 30, 2016, and were reported to the NRC to verify the performance indicator data. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, to determine the accuracy of the reported data.

These activities constituted verification of the radiological effluent technical specifications (RETS)/offsite dose calculation manual (ODCM) radiological effluent occurrences performance indicator, as defined in Inspection Procedure 71151. This inspection procedure is complete.

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Routine Review

a. Inspection Scope

Throughout the inspection period, the inspectors performed daily reviews of items entered into the licensee's corrective action program and periodically attended the licensee's condition report screening meetings. The inspectors verified that licensee personnel were identifying problems at an appropriate threshold and entering these problems into the corrective action program for resolution. The inspectors verified that the licensee developed and implemented corrective actions commensurate with the significance of the problems identified. The inspectors also reviewed the licensee's problem identification and resolution activities during the performance of the other inspection activities documented in this report.

b. Findings

No findings were identified.

.2 Semiannual Trend Review

a. Inspection Scope

The inspectors reviewed the licensee's corrective action program, performance indicators, system health reports, and other documentation to identify trends that might

indicate the existence of a more significant safety issue. The inspectors verified that the licensee was taking corrective actions to address identified adverse trends. The inspectors did not review any cross-cutting themes because none existed at the site.

The inspectors documented in NRC Inspection Report 05000298/2016002 an apparent trend associated with the organization's implementation of their process for planning, controlling, and executing work activities, such that nuclear safety is the overriding priority; this includes the identification and management of risk commensurate with the work and the need for coordination with different work groups or job activities [H.5 Work Management]. In response to the inspectors' observations, the licensee established an interdisciplinary team of individuals to review the trend in work management. The licensee entered this trend into the corrective action program as Condition Report CR-CNS-2016-03783 and initiated an apparent cause evaluation (ACE) to investigate this condition. The inspectors reviewed this ACE and the effectiveness of its corrective actions. Additionally, the inspectors identified multiple examples associated with the implementation of the work management process. These examples included:

- CR-CNS-2016-03562: The licensee missed limiting condition for operation (LCO) 3.3.1.1, Condition A, entry for intermediate range monitor (IRM)/average power range monitor (APRM) maintenance and calibrations. The licensee initiated an ACE to investigate this condition.

- CR-CNS-2016-06901: Residual heat removal (RHR) pump D tripped due to performing preventative maintenance on the primary containment isolation signal (PCIS) relay K31.

- CR-CNS-2016-07645: A loss of shutdown cooling occurred in Refueling Outage (RE) 29 during PCIS relay K27 preventative maintenance. The licensee initiated a root cause evaluation to investigate this condition.

- CR-CNS-2016-07527: Work Order (WO) 5065847 contained the incorrect torque value for the emergency diesel generator 2 cam cover bolting.

- CR-CNS-2016-07902: WO 5057310 contained inadequate post-maintenance testing (PMT) work instructions for G sump.

- CR-CNS-2016-08045: WO 5156429 PMT work instructions required revision for hydraulic control unit 10-15. The reason the PMT work instructions required revision was because they were to be performed during a clearance order release, but clearance order release did not indicate this.

- CR-CNS-2016-08334: WO 5065109 high pressure turbine replacement PMT was not completed.

These activities constituted completion of one semiannual trend review sample, as defined in Inspection Procedure 71152.

b. Observations and Assessments

1. In response to the missed LCO 3.3.1.1, Condition A, entry for WO 5060683, 5013046, and 5013047, the licensee initiated Condition Report CR-CNS-2016-03562 and an ACE to investigate this condition. The licensee's investigation determined two apparent causes (AC):

    - AC 1: Maintenance plans 800000028235 and 800000028236 incorrectly provided discussion on the potential impact on the APRM downscale companion trip. Plant impact statements were incorrect.

    - AC 2: The organization elected to change the schedule with intent to support the work load of the maintenance shops. Originally, these WOs (5060683, 5013046, and 5013047) were scheduled for different days. This change in the schedule did not have adequate oversight or review to ensure work should be performed at the same time.

    The licensee's corrective actions (CA) for the identified causes were:

    - CA 1: Additional oversight for schedule changes reinforced at the weekly shift manager phone call. Discussions where held with each crew related to how the event occurred and lessons learned. This action was completed June 26, 2016.

    - CA 2: The control room supervisor (CRS) presented findings to the crews and work control administrators. Review events with operations management prior to standing watch in the control room. This action was completed July 7, 2016.

    - CA 3: A revision to maintenance plans 800000028235 and 800000028236 to provide discussion on the potential impacts to the APRM downscale companion trip. This action is scheduled to be completed on January 12, 2017.

    - CA 4: Each shift manager and work control supervisor meets with their subordinates and discusses the findings from this evaluation. Reinforce the expectation for additional oversight for changes to schedule and requirement to inform the CRS that the companion APRM operability must be addressed when bypassing an IRM. This action was completed September 1, 2016.

    - CA 5: Operations management will perform a seminar with work week directors and maintenance superintendents to discuss findings from this evaluation. Reinforce the expectation for additional oversight for changes to the schedule. This action was completed September 22, 2016.

    The inspectors identified that the licensee's extent of condition review included a search of condition reports for the last 3 years (July 1, 2013, to July 24, 2016) for potential missed LCO entries. The licensee identified several condition reports related to LCO entries associated with environmental qualified terminal boxes and

seismic sensitive instrumentation. However, the licensee's extent of condition review did not consider AC 1 and AC 2 as part of the review. The licensee did not review a sampling of maintenance plans for potential inadequate plant impact statements that could lead to a missed LCO entry. Additionally, the licensee did not review sampling of scheduled WOs to determine if all entries into LCOs were identified.

The inspectors also questioned the effectives of CA 1, CA 2, and CA 5. Specifically, during Refueling Outage (RE) 29, two examples of schedule changes resulted in plant events. The examples were:

- CR-CNS-2016-06901: RHR pump D tripped due to performing preventative maintenance on primary containment isolation signal (PCIS) relay K31. This relay maintenance was originally recognized by the licensee to be performed during a RHR pump shutdown cooling maintenance window. However, it was scheduled during a time that did not preclude a RHR pump from running.

- CR-CNS-2016-07645: A loss of shutdown cooling occurred in RE 29 during PCIS relay K27 preventative maintenance. The licensee initiated a root cause evaluation to investigate this condition.

2. In response to the inspectors' observations documented in NRC Inspection Report 05000298/2016002, associated with H.5, the licensee established an interdisciplinary team of individuals to review this trend. The licensee entered this trend into the corrective action program as Condition Report CR-CNS-2016-03783 and initiated an ACE to investigate this condition. The licensee's investigation determined the AC was that behaviors associated with execution and overall sensitivity to the work management process do not consistently meet Cooper Nuclear Station's standards.

   The licensee's CAs for the identified cause were:

   - CA 1: Work with management to schedule briefings with the leadership team. This action was completed September 15, 2016.

   - CA 2: Develop briefing material to highlight awareness and increase sensitivity to the work management process. This action was completed on December 2, 2016.

   - CA 3: Conduct a discussion about the findings from this evaluation at a leadership and alignment meeting prior to RE 29. This action was completed September 30, 2016.

   - CA 4: Distribute a message using the explosives detectors (puffer message) to reinforce the need to adhere to station processes, procedures, and other written guidance. This action was completed on September 9, 2016.

   - The licensee also credited CAs associated with CR-CNS-2016-03851 to obtain top decile performance. These actions are scheduled to be completed in December 2016 and January 2017.

The inspectors identified that the licensee's extent of condition review was limited to the work management process and practices at Cooper Nuclear Station. The licensee did not consider a sampling method of planned WOs to determine if they meet the requirements of the licensee's work management process. In response to the inspectors' observations, the licensee entered this into their corrective action program for resolution as Condition Report CR-CNS-2016-08835. This condition report was closed to Condition Report CR-CNS-2016-03783, which includes a CA to re-perform the extent of condition evaluation. The licensee revised the extent of condition review to direct a sampling of WOs from the last 6 months of calendar year 2016 through calendar year 2017 and evaluate new CAs as required.

The inspectors determined that the work management, H.5, trend represented a continued adverse trend worthy of additional monitoring for long term corrective actions and sustained improvement.

c. Findings

No findings were identified.

.3 Annual Follow-up of Selected Issues

a. Inspection Scope

The inspectors selected one issue for an in-depth follow-up:

- On November 30, 2016, review of safety-related service water piping that was below design minimum wall thickness or had through wall leakage.

  The inspectors assessed the licensee's problem identification threshold, cause analyses, extent of condition reviews and compensatory actions. The inspectors verified that the licensee appropriately prioritized the planned corrective actions and that these actions were adequate to correct the condition.

These activities constituted completion of one annual follow-up sample, as defined in Inspection Procedure 71152.

b. Findings

No findings were identified.

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

.1 (Closed) Licensee Event Report (LER) 05000298/2016003-00, "Scaffold Construction Places Plant in a Condition Prohibited by Technical Specifications"

a. Inspection Scope

On September 14, 2016, during testing of the reactor recirculation motor generator (RRMG) ventilation air operated secondary containment isolation valves (SCIVs), valve HV-AOV-265 failed to close when the valve was stroked for timing. The valve was declared inoperable, and in order to comply with the Technical Specification (TS) 3.6.4.2 required action to isolate the ventilation path within 8 hours, operations

personnel commenced preparations for transitioning the plant to single loop operation. As part of these preparations, the licensee reduced power to approximately 50 percent to support removing the associated RRMG set from service.

The licensee's investigation identified that during the attempt to close the valve, the air supply line became pinched between the moving cylinder of the valve actuator and a scaffold that had been erected on June 29, 2016, to support work on a different valve. As such, air was restricted from exhausting from the air cylinder, pneumatically locking the piston and preventing the valve from closing. Further investigation revealed that the air line had been crimped and required replacement. After removing the scaffolding interference, the air line for valve HV-AOV-265 was replaced, and the valve stroke testing was re-performed satisfactorily. The valve was declared operable, and the plant was returned to 100 percent power.

The licensee initiated a root cause evaluation to determine the cause of the event. The licensee reported this failure under 10 CFR 50.73(a)(2)(i)(B) as a condition prohibited by TS because the scaffolding, installed in June 2016, was in a position to block the movement of the valve since that time. The inspectors reviewed the event, including station logs and TS requirements; walked down the affected components; and discussed the events with the licensee. The inspectors also reviewed the root cause evaluation, extent of condition and cause reviews, and the corrective actions associated with the event to ensure they were appropriate.

This licensee event report is closed.

b. Findings

Introduction. The inspectors reviewed a self-revealed, Green, non-cited violation of TS 3.6.4.2, "Secondary Containment Isolation Valves," for the licensee's failure to maintain SCIV, HV-AOV-265, operable as a result of erecting scaffolding that interfered with valve operation. Specifically, between June 29, 2016, and September 14, 2016, the licensee erected scaffolding in close proximity of ventilation SCIV, HV-AOV-265, such that, during valve stroking, the scaffolding would pinch the actuator air line and prevent the valve from closing, rendering the valve inoperable for approximately 10 weeks.

Description. On September 14, 2016, during testing of the RRMG ventilation air operated valves (AOVs), one of the valves, HV-AOV-265, failed to close as required by TS. The licensee declared the ventilation valve inoperable and entered Limiting Condition for Operation (LCO) 3.6.4.2, Condition A. In order to comply with TS upon discovery of the condition, the licensee needed to isolate the affected flow path with a motor-operated SCIV in series with the inoperable AOV. This action would have isolated ventilation flow to an RRMG set, and resulted in the need to take the associated recirculation pump out of service. As a result, the licensee began preparations to transition the plant to single loop operation, including reducing the plant to approximately 50 percent power, in order to take the recirculation pump and RRMG out of service.

When the plant reached 50 percent power, the licensee completed its investigation into the cause of the valve failure. Upon investigation, the licensee discovered that the air supply line to the AOV was pinched between the valve actuator and a scaffold that was erected to support work on a nearby component. The licensee took action to modify the scaffold and replace the pinched air line. The licensee subsequently tested the valve

and successfully stroked it closed within TS required times. The licensee declared the valve operable and terminated preparations for single loop operation. In addition, the licensee performed an extent of condition review and identified at least two other RRMG ventilation valves with air lines that were being impacted to a lesser extent by similar scaffolding. This resulted in slowed valve stroke times for these additional valves, and the licensee took action to correct these issues.

The licensee initiated a root cause evaluation to review the cause of the event. The licensee determined that the root cause of the event was that personnel involved in the planning, construction, and inspection of the scaffold built for a nearby component were not aware of the unique external movement path of the valve actuator. Specifically, the actuator for this ventilation valve and several others was of a design that required external movement of the actuator and its associated air line during valve stroking. As a result of this movement, scaffolding that was not originally touching the actuator pinched the air line when the actuator moved. The inspectors noted that the scaffolding was erected on June 29, 2016, and as a result, valve HV-AOV-265 would not have functioned and was inoperable during the time that the scaffold was in place.

The inspectors noted that a similar event occurred in 2001, which had the same cause. At that time, Procedure 7.0.7, "Scaffolding Construction and Control," was revised to include specific requirements for operations personnel to walk down and review the scaffold build plan and identify and document any operational concerns associated with the work. During the 2016 event, the inspectors observed that the process was followed; however, the operations walk-down failed to identify any operational concerns relating to the potential movement of the actuator. The inspectors noted that operations personnel performing the walk-down had stated during interviews that they did not recognize the external movement path of the actuator. The inspectors concluded that the operations personnel did not have sufficient training and plant knowledge to recognize the hazard, and in addition, Procedure 7.0.7 did not contain special scaffolding construction considerations for ventilation components equipped with actuators requiring external movement. As a result, the inspectors concluded that the licensee failed to ensure that personnel, equipment, procedures, and other resources were available and adequate to support nuclear safety.

The licensee initiated corrective actions to revise the procedure to include specific guidance for the planning, building, and inspection of scaffolds in the vicinity of valve HV-AOV-265 and other AOVs having actuators of a similar design. The licensee also took action to install signage to warn personnel of the external movement of these AOVs.

Analysis. The licensee's failure to implement Procedure 7.0.7, "Scaffolding Construction and Control," Revision 34, to ensure scaffolding did not adversely affect plant equipment, in violation of Technical Specification 3.6.4.2, was a performance deficiency. The performance deficiency was determined to be more than minor, and therefore a finding, because it was associated with the structure, system, and component and barrier performance attribute of the Barrier Integrity Cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (secondary containment) protect the public from radionuclide releases caused by accidents or events. Specifically, the improperly erected scaffolding prevented the operation of an SCIV, rendering it inoperable for approximately 10 weeks. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination

Process (SDP) for Findings At-Power," dated June 19, 2012, the inspectors determined that the finding had very low safety significance (Green) because it only represented a degradation of the radiological barrier function provided for the control room, reactor building, spent fuel pool building, or standby gas treatment (SBGT) system. The finding had a cross-cutting aspect in the area of human performance associated with resources. Specifically, the licensee failed to ensure that personnel, equipment, procedures, and other resources were available and adequate to support nuclear safety [H.1].

Enforcement. Technical Specification (TS) 3.6.4.2 requires, in part, that in Modes 1, 2, and 3, each secondary containment isolation valve (SCIV) shall be operable. TS 3.6.4.2, Condition A, requires, in part, that for one penetration flow path with one SCIV inoperable, the licensee shall isolate the affected penetration flow path within 8 hours. TS 3.6.4.2, Condition C, requires that if required actions and associated completion times of Condition A or B are not met, the licensee shall be in Mode 3 within 12 hours and be in Mode 4 within 36 hours. Contrary to the above, between June 29, 2016 and September 14, 2016, while in Mode 1 with one SCIV inoperable, the licensee failed to isolate the affected penetration flow path within 8 hours and failed to take action to be in Mode 3 within 12 hours and in Mode 4 within 36 hours. Specifically, the licensee erected scaffolding in close proximity of ventilation SCIV HV-AOV-265, such that, during valve stroking, the scaffolding would pinch the actuator air line and prevent the valve from closing, rendering the valve inoperable for approximately 10 weeks. Immediate corrective actions included removal of the scaffolding, replacement of the pinched air line, and restoration of the valve to operable status. Because this violation was of very low safety significance (Green) and was entered into the licensee's corrective action program as Condition Report CR-CNS-2016-05608, this violation is being treated as a non-cited violation in accordance with Section 2.3.2.a of the NRC Enforcement Policy. (NCV 05000298/2016004-03, "Failure to Maintain Service Water Pump Maintenance Procedure")

.2 (Closed) Licensee Event Report (LER) 05000298/2016004-00, "Closure of Multiple Main Steam Isolation Valves due to High Flow Signal"

a. Inspection Scope

On September 24, 2016, at 8:40 p.m., during reactor cooldown for Refueling Outage (RE) 29, Cooper Nuclear Station control room operators closed the inboard main steam isolation valves (MSIV) to minimize steam flow to control the reactor cooldown rate. Reactor pressure was controlled by the main steam line drains, and the condensate/feed system was available for reactor water level control.

Several hours later, the licensee decided to increase the cooldown rate, as allowed by plant procedures and technical specifications, and opted to open one MSIV. As part of the process to reopen an MSIV, the licensee equalized the pressure across the MSIVs to below the limit of 200 psid in accordance with Station Procedure 2.2.56, "Main Steam System," Revision 49. When a differential pressure of 190 psid was established, the licensee proceeded with opening an MSIV. Upon opening the A inboard MSIV, valve MS-AO-80A, a Group 1 primary containment isolation signal was immediately received due to a main steam line high steam flow signal. The isolation signal resulted in a closure of all MSIVs and main steam line (MSL) drain valves. The control room operators subsequently equalized pressure across the valves and successfully reopened

all MSIVs at 6:52 p.m. on September 25, 2016, restoring the MSL decay heat removal path, and reestablishing desired plant cooldown rates.

The cause of the event was insufficient procedural guidance regarding the limitations on opening the MSIVs. To correct this issue, the applicable procedure was revised to change the differential pressure limits for opening MSIVs from 200 psid to 80 psid.

Because this event was a valid actuation of the Group 1 primary containment isolation signal, the licensee reported this under 10 CFR 50.72 and 50.73. The inspectors reviewed the event, including station logs and TS requirements; walked down the affected components; and discussed the events with the licensee. The inspectors also reviewed the apparent cause evaluation, extent of condition reviews, and the corrective actions associated with the event to ensure they were appropriate.

The licensee event report is closed.

b. Findings

Introduction. The inspectors reviewed a self-revealed, Green, non-cited violation of Technical Specification 5.4.1.a for the licensee's failure to maintain Station Procedure 2.2.56, "Main Steam System," Revision 49, to prevent a main steam line high flow Group 1 primary containment isolation signal when opening an inboard main steam isolation valve. Specifically, the licensee failed to maintain Station Procedure 2.2.56 with adequate differential pressure limits for re-opening closed main steam isolation valves during plant shutdown, which caused the unexpected closure of all the open main steam isolation and drain valves during the cooldown process.

Description. On September 24, 2016, at 8:00 p.m., the station scrammed the reactor from 20 percent power in preparation for commencing Refueling Outage (RE) 29. At 8:40 p.m., operations personnel closed all inboard main steam isolation valves (MSIVs) to control the reactor cooldown rate in accordance with Station Procedure 2.1.4, "Normal Shutdown," Revision 152.

Several hours later, the licensee decided to increase the cooldown rate, as allowed by plant procedures and technical specifications, and opted to open one MSIV. As part of the process to reopen an MSIV, the licensee equalized the pressure across the MSIVs to below the limit of 200 psid in accordance with Station Procedure 2.2.56. When a differential pressure of 190 psid was established, the licensee proceeded with opening an MSIV. Upon opening the A inboard MSIV, valve MS-AO-80A, a Group 1 primary containment isolation signal was immediately received due to a main steam line high steam flow signal. The isolation signal resulted in a loss of the main steam line decay heat removal path, which caused reactor coolant system pressure and temperature to increase by approximately 13 psig and 3 degrees Fahrenheit, respectively, during the event. The immediate corrective actions were to reset the Group 1 isolation and open the main steam line drains to recommence plant cooldown and restore the main steam line decay heat removal path. The control room operators subsequently equalized pressure across the valves and successfully reopened all MSIVs at 6:52 p.m. on September 25, and reestablished desired plant cooldown rates. The licensee entered this deficiency into the corrective action program as Condition Report CR-CNS-2016-05835 and initiated an apparent cause evaluation (ACE) to investigate the condition.

The ACE concluded the apparent cause of this event was that insufficient procedure guidance existed regarding the limitations for opening MSIVs. Specifically, Station Procedure 2.2.56 included the following precaution, “Do not attempt to open MSIVs with a differential pressure across the valve > 200 psid.” This precaution was based on vendor specifications and the Updated Safety Analysis Report associated with protection of the valve’s pneumatic cylinder from equipment damage. However, prior to 1988, station procedures required an MSIV differential pressure of 50 psid before opening inboard MSIVs, but no documented basis for this limit was provided. In 1988, the station changed the MSIV differential pressure limit from 50 psid to 200 psid. The licensee determined that the 200 psid limit was inadequate because having such a high differential pressure across the valve when opening it made the main steam system vulnerable to an unanticipated Group 1 isolation signal. As a final corrective action, the licensee established the MSIV differential pressure limitation of 80 psid in Station Procedure 2.2.56, Revision 50, to prevent a main steam line high flow Group 1 isolation signal from occurring when reopening MSIVs during normal plant shutdown operations.

The inspectors noted that prior to 2016, the licensee closed the MSIVs seven times to control the reactor cooldown rate, with the last occurrence in 2009. In those instances, due to having ideal plant conditions, the licensee was able to reopen the MSIVs each time without challenging the 200 psid limit and without a main steam line high flow Group 1 isolation signal occurring. The inspectors noted that this procedural section to reopen a closed MSIV during plant cooldown was rarely challenged and did not present many opportunities to identify the inadequate procedural limits during recent years. As a result, the inspectors determined that this issue was not indicative of current performance. The inspectors also determined that this issue was self-revealed because it was identified as a result of an unplanned MSL A steam line high flow Group 1 primary containment isolation signal when opening valve MS-AOV-80A.

Analysis. The licensee’s failure to maintain Station Procedure 2.2.56 to prevent a main steam line high flow Group 1 isolation signal when opening an inboard MSIV, in violation of Technical Specification 5.4.1.a, was a performance deficiency. The performance deficiency was determined to be more than minor, and therefore a finding, because it was associated with the procedural quality attribute of the Initiating Events Cornerstone, and affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown. Specifically, the Group 1 isolation signal closed the MSL drain valves, which resulted in a loss of the MSL decay heat removal path and caused reactor coolant system pressure and temperature to increase. The inspectors determined Inspection Manual Chapter 0609, Appendix G, Attachment 1, “Shutdown Operations Significance Determination Process Phase 1 Initial Screening and Characterization of Findings,” dated May 9, 2014, was not applicable because plant temperature and pressure were not within the normal residual heat removal/decay heat removal conditions. Using Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated June 19, 2012, the inspectors determined that the finding screened as having very low safety significance (Green) because it did not cause a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of a trip to a stable shutdown condition. A cross-cutting aspect was not assigned to this finding because the performance deficiency occurred in 1988 when the licensee changed the procedural limits for differential pressure across the MSIVs when reopening them, and therefore, was not indicative of current licensee performance.

Enforcement. Technical Specification 5.4.1.a requires, in part, that written procedures be established, implemented, and maintained covering the applicable procedures recommended in Appendix A to Regulatory Guide 1.33, "Quality Assurance Program Requirements," Revision 2, February 1978. Regulatory Guide 1.33, Appendix A, Section 4.I, requires procedures for startup, operation, and shutdown of the main steam system. Contrary to the above, from 1988 until 2016, the licensee failed to maintain procedures for startup, operation, and shutdown of the main steam system. Specifically, the licensee failed to maintain Station Procedure 2.2.56, "Main Steam System," Revision 49, with adequate differential pressure limits to prevent a main steam line high flow Group 1 primary containment isolation when opening an inboard MSIV. The isolation signal resulted in a loss of the main steam line decay heat removal path, which caused reactor coolant system pressure and temperature to increase by approximately 13 psig and 3 degrees Fahrenheit, respectively, during the event. The immediate corrective actions were to reset the Group 1 isolation and open the main steam line drains to recommence plant cooldown. This violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the Enforcement Policy, because it was of very low safety significance (Green) and was entered into the licensee's corrective action program as Condition Report CR-CNS-2016-05835.
(NCV 05000298/2016004-04, "Failure to Maintain Main Steam System Operating Procedure")

.3 (Closed) Licensee Event Report (LER) 05000298/2016005-00, "Implementation of Enforcement Guidance Memorandum 11-003, Revision 3, Causes Conditions Prohibited by Technical Specifications"

a. Inspection Scope

During Refueling Outage 29, Cooper Nuclear Station performed operations with a potential for draining the reactor vessel (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 reactor pressure vessel water level below the top of fuel, without crediting the use of a mitigating measure to terminate the uncovering of fuel. Secondary containment is required by Technical Specification (TS) 3.6.4.1 to be operable during OPDRV activities. The required action for this specification is to suspend the OPDRV operations. Therefore, entering the OPDRV activity 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 Enforcement Guidance Memorandum (EGM) 11-003, Revision 3, on January 15, 2016, 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 the 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 reactor pressure vessel water level to identify the onset of a loss of inventory event;

(b) maintaining the capability to isolate the potential leakage paths; (c) prohibiting Mode 4 (cold shutdown) OPDRV activities; and (d) prohibiting movement of irradiated fuel with the spent fuel pool storage gates removed in Mode 5; and (4) ensure that the licensee follow all other Mode 5 TS requirements for OPDRV activities.

The inspectors reviewed this licensee event report for potential performance deficiencies and violations of regulatory requirements. The inspectors reviewed the station's implementation of the EGM 11-003, Revision 3, during the OPDRV activities. Specific observations included:

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

2. The inspectors noted that the reactor vessel level was maintained at least greater than 21 feet above the top of the reactor pressure vessel flange as required by TS 3.9.6. The inspectors also verified that at least one safety-related pump was 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 reactor pressure vessel flange was greater than 24 hours.

3. The inspectors verified that the OPDRV activities were not conducted in Mode 4 and that the licensee did not move irradiated fuel during the OPDRV activities. The inspectors verified that two independent means of measuring reactor pressure vessel water level were available for identifying the onset of loss of inventory events.

TS 3.6.4.1 requires, in part, that secondary containment shall be operable during OPDRV activities. TS 3.6.4.1, Condition C, requires the licensee to initiate actions to suspend OPDRV activities immediately when secondary containment is inoperable. Contrary to the above, from October 1, 2016, through October 25, 2016, Cooper Nuclear Station failed to suspend OPDRV activities when secondary containment was inoperable. Specifically, the station conducted the following four OPDRV activities without an operable secondary containment:

- Reactor recirculation pumps A and B (RR-P-A and RR-P-B) maintenance without jet pump plugs installed while performing Station Procedure 6.1SGT.401, "SGT A Fan Capacity Test, SGT B Cooling Flow Test and Check Valve IST (Div 1)," Revision 19.

- Pumps RR-P-A and RR-P-B maintenance and control rod drive (CRD) withdrawal/bypass operations in support of hydraulic control unit (HCU) 42-31 maintenance during repairs of main steam line (MSL) B outboard main steam isolation valve (MSIV) MS-AOV-86B.

- Pumps RR-P-A and RR-P-B maintenance without the jet pump plugs installed while draining the reactor core isolation cooling system and flushing MSIVs.

- Pump RR-P-A maintenance, CRD-V-113 and CDR-V-105 freeze seal, and CRD venting. These OPDRVs were in progress while secondary containment was inoperable for repair of valves MS-AOV-86B and 86A, reactor building personnel airlock seal repair, shift of reactor building ventilation, service water valve 531 draining, and residual heat removal valve 57/67 draining.

These conditions were reported by the licensee as conditions prohibited by TS. The licensee entered this issue into its corrective action program as Condition Report CR-CNS-2016-06202.

Since this violation occurred during the discretion period described in EGM 11-003, Revision 3, 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-16-226).

In accordance with EGM 11-003, Revision 3, each licensee that receives discretion must submit a license amendment required 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.

This licensee event report is closed.

b. Findings

No findings were identified.

.4 Event Follow-up: Fire in the Heater Bay Area

a. Inspection Scope

At 10:18 a.m. on October 4, 2016, a fire was reported in the turbine building heater bay area. The fire was reported by a worker performing fire watch duties for nearby work activities. The licensee discovered that the fire was caused by a rag in close proximity to tungsten inert gas (TIG) welding activities. The welder involved received a small burn to his arm, which resulted in first aid treatment being required. The fire was extinguished within 3 minutes by the fire watch for the activity, using a fire extinguisher. The licensee entered their Emergency Procedure 5.1Incident, and the fire brigade was dispatched to respond to the fire. The residents responded to the control room to evaluate the licensee's actions in response to the event. The inspectors reviewed the licensee's emergency classification matrix to determine if any emergency action levels (EALs) were met. The inspectors determined that fire was not located in an area that would have prompted entry into the EALs, and in addition, the fire was put out within 15 minutes.

The licensee's subsequent human performance investigation revealed that the area was walked down prior to the welding activities in order to look for transient combustibles, like loose rags. However, the rag that caught fire was determined to have been beneath the welding blanket during the walk-down, and the walk-down was not comprehensive enough to identify the fire hazard. The inspectors observed that station procedures directed that a 5-foot radius around the work be cleared of combustibles prior to the initiation of welding activities. The inspectors also noted that as a follow-up action, the

licensee's safety personnel walked down other ongoing hot work to look for transient combustibles. In addition, the licensee took action to pass on learnings from the event to the rest of the site. During the event, the inspectors assessed licensee response to the fire, including command and control of the fire brigade that was dispatched to ensure the fire had been extinguished. The inspectors also walked down the location where the fire occurred and other areas where hot work was in progress to evaluate the effectiveness of the licensee's immediate actions. This issue was entered into the licensee's corrective action program as CR-CNS-2016-06427.

b. Findings

No findings were identified.

These activities constituted completion of four event follow-up samples, as defined in Inspection Procedure 71153.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On October 7, 2016, the inspectors presented the radiation safety inspection results to Mr. O. Limpias, former Vice President-Nuclear and Chief Nuclear Officer, and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

On November 21, 2016, the inspector presented the results of the in-office and onsite inspection of the changes to the licensee's emergency plan implementing procedures to Mr. R. Penfield, Director, Nuclear Safety Assurance, and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

On January 11, 2016, the inspectors presented the inspection results to Mr. K. Higginbotham, Vice President-Nuclear and Chief Nuclear Officer, and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

T. Barker, Manager, Engineering Program and Components
J. Bebb, Health Physicist, Radiation Protection
D. Buman, Director, Nuclear Safety Assurance
B. Chapin, Manager, Maintenance
T. Chard, Manager, Quality Assurance
L. Dewhirst, Manager, Corrective Action and Assessment
K. Dia, Director, Engineering
J. Flaherty, Senior Licensing Engineer
T. Forland, Engineer, Licensing
G. Gardner, Engineering Design Manager
D. Goodman, Manager, Operations
K. Higginbotham, Vice President-Nuclear and Chief Nuclear Officer
D. Kimball, Director, Nuclear Oversight
J. Olberding, Licensing Specialist, Licensing
R. Penfield, Director, Nuclear Safety Assurance
J. Reimers, Manager, System Engineering
J. Shaw, Manager, Licensing
J. Stough, Manager, Emergency Preparedness
C. Sunderman, Manager, Radiation Protection

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000298/2016004-01	NCV	Failure to Maintain Reactor Vessel Assembly Procedure to Ensure Adequate Moisture Separator Shielding (Section 1R13)
05000298/2016004-02	NCV	Failure of an Analysis to Demonstrate that Changes Did Not Reduce the Effectiveness of the Emergency Plan (Section 1EP4)
05000298/2016004-03	NCV	Failure to Maintain Service Water Pump Maintenance Procedure (Section 4OA3)
05000298/2016004-04	NCV	Failure to Maintain Main Steam System Operating Procedure (Section 4OA3)

Closed

05000298/2016003-00	LER	Scaffold Construction Places Plant in a Condition Prohibited by Technical Specifications (Section 4OA3)
05000298/2016004-00	LER	Closure of Multiple Main Steam Isolation Valves due to High Flow Signal (Section 4OA3)
05000298/2016005-00	LER	Implementation of Enforcement Guidance Memorandum 11-003, Revision 3, Causes Conditions Prohibited by Technical Specifications (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

Procedures

Number	Title	Revision
2.2.32	Fuel Pool Cooling and Demineralizer System	96
2.2.56	Main Steam System	49
2.2.67	Reactor Core Isolation Cooling System	73
2.2.69.2	RHR System Shutdown Operations	94
2.2.69.3	RHR Suppression Pool Cooling and Containment Spray	46
2.4FPC	Fuel Pool Cooling Trouble	34
6.1MISC.503	31 Day Venting of ECCS and RCIC Injection/Spray Subsystem Piping (DIV 1)	7
14.0.10	Instrument System Valve Configuration Management	7

Work Orders

5057932	5057960	5059817

Section 1R05: Fire Protection

Miscellaneous Documents

Number	Title	Revision
	Fire Impairment, FP16-BLDG-DOOR-H111 BAT RM	1
	Fire Impairment, FP-BLDG-DOOR-H200 5071129	1
11-085	NEDC, Fire Safety Analysis for Fire CB-A1 Report R1906-008-CBA1	3

Procedures

Number	Title	Revision
0-Barrier	Barrier Control Process	21
0-Barrier-Control	Control Building	6
0.23	CNS Fire Protection Plan	74
0.50.5	Outage Shutdown Safety	35
0.7.1	Control of Combustibles	40
2.1.6	Primary Containment Access Preparation and Closeout Activities	15

Miscellaneous Documents

Number	Title	Revision
2.2.38.2	Portable Heating System	17

Condition Reports (CRs)

CR-CNS-2009-01895 CR-CNS-2016-07509 CR-CNS-2016-08025

Work Orders

5071129

Section 1R11: Licensed Operator Requalification Program and Licensed Operator Performance

Miscellaneous Documents

Number	Title	Revision
SKL0515291	Performance Mode #2	0

Procedures

Number	Title	Revision
0-EN-TQ-210	Conduct of Simulator Training	9C0
2.1.1	Startup Procedure	185
2.1.9	Low Power Operation for Maintenance Activities (Hot Standby)	67

Condition Reports (CRs)

CR-CNS-2016-08046 CR-CNS-2016-08058 CR-CNS-2016-08063

Section 1R12: Maintenance Effectiveness

Miscellaneous Documents

Number	Title	Revision
CGI 2003-010	Engineering Change	2
CGI 2003-011	Engineering Change	2
CGI 2003-012	Engineering Change, Dedication of 3M Scotch Super 33+ Electrical Tape	2
CGI-10615536	Engineering Change, CGI for Flexitallic Spiral Wound Gaskets	1
PC-CONT2A	Maintenance Rule Function PC-CONT2A Performance Criteria Basis	4

Miscellaneous Documents

Number	Title	Revision
PC-CONT2B	Maintenance Rule Function PC-CONT2B Performance Criteria Basis	4

Procedures

Number	Title	Revision
0.40.8	Circuit Evaluations	4
3.40	Primary Containment Leakage Rate Testing Program	12
6.PC.205	Instrument Line Excess Flow Check Valve Test	14
6.PC.511	High Pressure Coolant Injection (HPCI) Local Leak Rate Tests	14
6.PC.513	Main Steam Local Leak Rate Tests	25
7.3.16	Low Voltage Relay Removal and Installation	22
7.3.26.9	Essential 3M Tape Installation	8

Condition Reports (CRs)

CR-CNS-2014-06420	CR-CNS-2014-07694	CR-CNS-2015-06635	CR-CNS-2016-06185
CR-CNS-2016-06227	CR-CNS-2016-06463	CR-CNS-2016-06470	CR-CNS-2016-06901
CR-CNS-2016-07592			

Work Orders

4949494	4951317

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Miscellaneous Documents

Number	Title	Revision
	Shutdown Safety Contingency Plan RE29-027, RHR Loop A SDC Flush	0
CNS RP-18B	Dosimeter Histograms for Workers Receiving Dose Rate Alarms During Moisture Separator Lift	October 29, 2016
RE29-029	Lowering RPV Level Band OPDRV	0
09-102	NEDC, CNS Internal Flooding Analysis	1
2006	Burns and Roe, CNS Flow Diagram Circulating, Screen Wash, and Service Water Systems, Sheet 1	90
2006	Burns and Roe, CNS Flow Diagram Circulating, Screen Wash, and Service Water Systems, Sheet 2	48

Miscellaneous Documents

Number	Title	Revision
2006	Burns and Roe, CNS Flow Diagram Circulating, Screen Wash, and Service Water Systems, Sheet 3	56
2016-05	ALARA, Radiological Job Plan – RE-29 Refuel Floor Work	0
2016-545	Radiation Work Permit, Rx Vessel Disassembly and Reassembly	0
921D136	PCIS Outboard Relay Cabinet (9-42)	4
	Electronic Dosimetry Histograms for Selected Personnel	10/26/2016
	Refuel Floor Area Radiation Monitor Computer Histories	10/26/2016
	Selected Licensee Personnel Statements	10/26/2016

Procedures

Number	Title	Revision/Date
0-CNS-WM-104	On-Line Schedule Risk Assessment	3
0-PWG-01	Procedure Writer's Guide	20
0.50.5	Outage Shutdown Safety	35
0.50.5, Att. 6	Outage System Status Report – Div 1, 4160 V Bus Out of Service	October 7, 2016
0.50.5, Att. 6	Outage System Status Report – Div 1, 4160 V Bus Out of Service	October 9, 2016
2.0.1.2	Operations Procedure Policy	45
2.2.3.2	Screenwash System	100
2.2.7.1	Service Water System	120
2.2.9	Core Spray System	78
2.2.18	4160V Auxiliary Power Distribution System	192
2.2.66	Reactor Water Cleanup	109
2.2.69.1	RHR LPCI Mode	30
2.2.69.2	RHR System Shutdown Operations	94
2.3_B-3	Alarm Procedure – SW Pump B & D Discharge Header Low Pressure	37
2.3_S-1	Alarm Procedure – Panel S – Annunciator S-1	24
2.4SDC	Shutdown Cooling Abnormal Procedure	15
5.1.RAD	Building Radiation Trouble	18
6.MISC.502	ASME Class 1 System Leakage Test	51

Miscellaneous Documents

Number	Title	Revision
6.2SW.302	SW Pressure Instrument Calibration and Isolation Logic Functional Test (DIV 2)	11
7.3.1.12	GE IAV-54E (27) Relay Testing and Maintenance	7
7.3.10.2	Main Generator Potential Transformer Removal and Installation	3
7.3.32	Station Temporary Grounding	11
7.4Reassembly	Reactor Vessel Reassembly	13
15.2SW.601	SW Bay Emergency Sparger Functional Test (DIV 2)	3
9.EN-RP-100	Radiation Worker Expectations	10
9.EN-RP-123	Radiological Controls for Highly Radioactive Objects	02
9.EN-RP-141	Job Coverage	17
9.RADOP.2	Radiation Safety Standards and Limits	16
9.EN-RP-110-04	Radiation Protection Risk Assessment Process	10
9.EN-RP-110-05	ALARA Planning and Controls	04
4.8	Area Radiation Monitoring System	14
0-EN-HU-106	Procedure and Work Instruction Use and Adherence	3C0

Condition Reports (CRs)

CR-CNS-2008-05767	CR-CNS-2016-04884	CR-CNS-2016-06334	CR-CNS-2016-06691
CR-CNS-2016-07552	CR-CNS-2016-07645	CR-CNS-2016-07654	CR-CNS-2016-07656
CR-CNS-2016-07780	CR-CNS-2016-07933		

Work Orders

5064335	5064989	5094025	5155502	5155805
5155827	5156880			

Section 1R15: Operability Determinations and Functionality Assessments

Miscellaneous Documents

Number	Title	Revision
2851-3	18” SW-1 Class IVP – Reactor Building Isometric Drawing	N10
2852-8	SW-2 Service Water Class IVP – Reactor Building Isometric Drawing	N16

Miscellaneous Documents

Number	Title	Revision
SW-E-9-2851-3	Ultrasonic Thickness Measurement Report – RHR SW Data	December 13, 2016
SW-E-11-2852-8	Thickness Measurement Report – REC SW Data	August 29, 2016

Procedures

Number	Title	Revision
0-EN-HU-106	Procedure and Work Instruction Use and Adherence	3C0
0.26	Surveillance Program	70
2.2.67	Reactor Core Isolation Cooling System	73
6.RCIC.309	RCIC ($\leq$165 psig) Beginning of Cycle Test	28
6.1DG.302	Undervoltage Logic Functional, Load Shedding, and Sequential Loading Test (DIV 1)	85

Condition Reports (CRs)

CR-CNS-2016-05123	CR-CNS-2016-06582	CR-CNS-2016-07192	CR-CNS-2016-07200
CR-CNS-2016-07426	CR-CNS-2016-07603	CR-CNS-2016-07991	CR-CNS-2016-07992
CR-CNS-2016-08079	CR-CNS-2016-08094	CR-CNS-2016-08096	CR-CNS-2016-08369
CR-CNS-2016-08373	CR-CNS-2016-08580	CR-CNS-2016-08631	CR-CNS-2016-08766
CR-CNS-2016-08783	CR-CNS-2016-08790		

Work Orders

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Section 1R18: Plant Modifications

Miscellaneous Documents

Number	Title	Revision
6033800	Change Evaluation Document, Startup Station Service Transformer Replacement	0

Section 1R19: Post-Maintenance Testing

Miscellaneous Documents

Number	Title	Revision
6039901	Engineering Change, Stop Drill for Steam Dryer Skirt Crack Mitigation Near 180 Degrees	0

Miscellaneous Documents

Number	Title	Revision
10919459	Technical Evaluation Part Equivalency – AO-PCV-32	1

Procedures

Number	Title	Revision
0.4	Procedure Change Process	65
2.2.67	Reactor Core Isolation Cooling System	73
6.MISC.502	ASME Class 1 System Leakage Test	50, 51
6.MS.301	Main Steam Isolation Valve Limit Switch Channel Calibration	11, 12
6.MS.302	MSIV Accumulator Functional Test	5
6.PC.513	Main Steam Local Leak Rate Tests	26
6.PCIS.301	PCIS Group 2, 3, and 6 Isolation Logic System Functional Test and Reactor Building Vent Monitor Functional Test	24
6.RCIC.102	RCIC IST and 92 Day Test	33
6.RCIC.309	RCIC (≤165 psig) Beginning of Cycle Test	28
6.SW.102	Service Water System Post-LOCA Flow Verification	49
6.1SW.401	Diesel Generator Service Water Check Valve and Sump Test (IST)(DIV 1)	2

Condition Reports (CRs)

CR-CNS-2016-06357	CR-CNS-2016-06364	CR-CNS-2016-06361	CR-CNS-2016-06463
CR-CNS-2016-06470	CR-CNS-2016-06901	CR-CNS-2016-07138	CR-CNS-2016-07279
CR-CNS-2016-07442	CR-CNS-2016-07493	CR-CNS-2016-07592	CR-CNS-2016-07634
CR-CNS-2016-07636	CR-CNS-2016-07826	CR-CNS-2016-07827	CR-CNS-2016-07851
CR-CNS-2016-07852	CR-CNS-2016-07869	CR-CNS-2016-07890	CR-CNS-2016-07895
CR-CNS-2016-07896	CR-CNS-2016-07897	CR-CNS-2016-07898	CR-CNS-2016-07899
CR-CNS-2016-07900	CR-CNS-2016-07901	CR-CNS-2016-07903	CR-CNS-2016-07904
CR-CNS-2016-07907	CR-CNS-2016-07943	CR-CNS-2016-07957	CR-CNS-2016-08039
CR-CNS-2016-08112	CR-CNS-2016-08113	CR-CNS-2016-08114	CR-CNS-2016-08115
CR-CNS-2016-08122	CR-CNS-2016-08156	CR-CNS-2016-08159	CR-CNS-2016-08167
CR-CNS-2016-08209			

Work Orders

4825169	5057932	5057960	5059866	5059880

Work Orders

5059881	5064335	5064361	5064986	5064990
5065080	5149082	5152163	5152222	5152225
5152860	5153228	5154164		

Section 1R20: Refueling and Other Outage Activities

Miscellaneous Documents

Number	Title
HV-268MV	Contingency Plan RE29-036
HVRX-1-RE29	Clearance Order

Procedures

Number	Title	Revision/Date
	RE29 Risk Profile Pre-Outage Review of Rev. 1 Schedule	August 25, 2016
0-EN-HU-103	Human Performance Error Reviews	6C0
0.12	Working Hour Limitations and Personnel Fatigue Management	31
0.50.5	Outage Shutdown Safety	35
0.9	Tagout	89
2.1.9	Low Power Operation for Maintenance Activities (Hot Standby Condition)	67
2.2.56	Main Steam System	49

Condition Reports (CRs)

CR-CNS-2016-00131	CR-CNS-2016-00144	CR-CNS-2016-01081	CR-CNS-2016-01291
CR-CNS-2016-01480	CR-CNS-2016-01713	CR-CNS-2016-03566	CR-CNS-2016-05957
CR-CNS-2016-06783	CR-CNS-2016-07040	CR-CNS-2016-07045	CR-CNS-2016-07203
CR-CNS-2016-07254	CR-CNS-2016-07260	CR-CNS-2016-07254	CR-CNS-2016-07502
CR-CNS-2016-07764	CR-CNS-2016-08104	CR-CNS-2016-08128	CR-CNS-2016-08133
CR-CNS-2016-08384	CR-CNS-2016-08388	CR-CNS-2016-08402	CR-CNS-2016-08409

Work Orders

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Section 1R22: Surveillance Testing

Miscellaneous Documents

Title	Date
IPTE Briefing Sheet for 6.1DG.302	October 18, 2016

Procedures

Number	Title	Revision
EN-OP-116	Infrequently Performed Tests or Evolutions	12
2.1.22	Recovering From a Group Isolation	60
6.PCIS.301	PCIS Group 2, 3, and 6 Test and Reactor Building Vent Monitor Functional Test	24
6.SC.501	Secondary Containment Leak Test	28
6.1DG.302	Undervoltage Logic Functional, Load Shedding, and Sequential Loading Test (DIV 1)	85
6.1PRM.304	Reactor Building Ventilation Radiation Monitor Channel Calibration and Functional Test/Source Check (DIV 1)	18
6.1PRM.704	Reactor Building Ventilation Radiation Monitor Channel Functional Test (DIV 1)	8
6.2PRM.704	Reactor Building Ventilation Radiation Monitor Channel Functional Test	7

Condition Reports (CRs)

CR-CNS-2016-05361	CR-CNS-2016-07171	CR-CNS-2016-07172	CR-CNS-2016-07192
CR-CNS-2016-07200	CR-CNS-2016-07945	CR-CNS-2016-07947	

Work Orders

4996819	5022611	5022898

Section 1EP4: Emergency Action Level and Emergency Plan Changes

Miscellaneous Documents

Title	Date
50.54(q) Evaluation 2013-034, Procedure 5.7.1, Emergency Classification, Revision 46	
50.54(q) Evaluation 2016-011, Procedure 5.7.1, Emergency Classification, Revision 54	January 27, 2016

Section 2RS1: Radiological Hazard Assessment and Exposure Controls

Air Sampling Surveys

Location	Title	Date
RB 901’ Steam Tunnel	MS-AOV-A086A Breach	October 4, 2016
RX – RR “B” Drywell	Job Coverage / RR “B” Seal Removed	October 4, 2016
RX 1001’ Spent Fuel Pool	Stowed Refuel Bridge Mast	October 5, 2016
RX Drywell Under Vessel	Under Vessel Work	October 2, 2016
TG 932’ Stop Valve #1	Breach of Stop Valve #1	October 4, 2016

Audits and Self-Assessments

Number	Title	Date
16-05	QA Audit - Radiological Controls	July 12, 2016

Miscellaneous Documents

Title	Revision/Date
CNS Fuel Pool Inventory	
FME Plan for RE29 Refuel Floor	July 19, 2016
Hot Spot Master Index	June 8, 2016
LHRA Entry Door / Gate Check	June 2, 2016
Onsite Radioactive Materials Storage Areas	January 27, 2016
RE29 Torus Desludging & Inspection FME / Housekeeping Project Plan	0
Source Inventory List	August 29, 2016
Source Leak Tests	December 2015
Source Leak Tests	June 2016

Procedures

Number	Title	Revision
9.ENN-RP-102	Radiological Control	3

Miscellaneous Documents

	Title	Revision/Date
9.EN-RP-101	Access Control for Radiologically Controlled Areas	18
9.EN-RP-108	Radiation Protection Posting	12
9.EN-RP-109	Hot Spot Program	4
9.EN-RP-121	Radioactive Material Control	1
9.EN-RP-141	Job Coverage	17
9.EN-RP-151	Radiological Diving	2
9.EN-RP-311	Electronic Alarming Dosimeters	4
9.RADOP.1	Radiation Protection At CNS	14
9.RADOP.5	Airborne Radioactivity Sampling	28
9.RADOP.10	Radioactive Sources Control and Accountability	22
9.RADOP.14	Off-Site Radioactive Material Storage	5
9.RADOP.21	Radiological Control of Systems with Potential for Changing Radiological Conditions	0

Radiation Surveys, CNS-

Number	Title	Date
1605-0030	Reactor Bldg. 931' – B RHR HX	May 25, 2016
1605-0031	Reactor Bldg. 931' – Quarterly Survey	May 25, 2016
1606-0026	Reactor Bldg. - Refuel Floor	June 20, 2016
1608-0025	Reactor Bldg. 931' – Quarterly Survey	August 18, 2016
1608-0032	Reactor Bldg. 931' – B RHR HX	August 18, 2016
1608-0033	Radwaste Bldg. 903' - Filter Demin Valve Room	August 25, 2016
1609-0047	Radwaste Bldg. 903' - Filter Demin Valve Room	September 24, 2016
1610-0112	Turbine Bldg. 932' – Breach of Stop Valve #1	October 4, 2016
1610-0143	Reactor Bldg. 1001' - Refuel Floor	October 5, 2016

Miscellaneous Documents

Title	Revision/Date

Radiation Work Permits

Number	Title	Revision
2016-105	Radiography Activities	0
2016-439	Management, Engineering, NRC, QA, Safety, D/W Coordinators, Visitors, etc., OUTAGE Misc. Walk-downs	0
2016-501	All Valve Work & Support in Rx Bldg LHRA's and the RW 877' FD/WC Tk LHRA Rm	0
2016-506	Under Vessel CRDMs Changeout Also Includes Keyway Worker & RP Under Vessel Coverage	0
2016-509	Torus Desludge Diving – for DIVERS only	0
2016-511	ISI&FAC in Drywell & Steam Tunnel (includes Walk-downs, Oversight, Insulation Removal/Install, Surface Preps, Nozzle Door Operations)	0
2016-539	Management, Engineering, NRC, Safety, QA, etc., routine activities in HRAs & HCAs (NOT for Steam Tunnel entry)	0
2016-540	Management, Engineering, NRC, Safety, D/W Coordinators, QA, etc., Walk-downs in DRYWELL & Steam Tunnel	0
2016-548	Refuel Floor Support & Oversight	0

Condition Reports (CRs)

CR-CNS-2015-03299	CR-CNS-2015-03706	CR-CNS-2015-03916	CR-CNS-2015-05114
CR-CNS-2015-05574	CR-CNS-2015-06511	CR-CNS-2015-06957	CR-CNS-2015-07167
CR-CNS-2016-00497	CR-CNS-2016-02810	CR-CNS-2016-03032	CR-CNS-2016-03085
CR-CNS-2016-04906	CR-CNS-2016-06448	CR-CNS-2016-06475	

Section 2RS2: Occupational ALARA Planning and Controls (71124.02)

ALARA Planning and In-Progress Reviews

Number	Title	Revision
2016-01	CRDM Replacements	0
2016-04	LLRTs	0
2016-12	RE29 Outage ISI and FAC Examinations and Insulation	0, 1, 2, 3
2016-15	Inspect/Repair RF-CV-15CV	0, 1

Audits and Self-Assessments

Number	Title	Date
LO-2016-0007	Formal Benchmark, Collective Radiation Exposure Minimization	
QAD2016-0024	Quality Assurance Audit Report, Radiological Controls	August 10, 2016

Miscellaneous Documents

Title
2015 Cooper Nuclear Station ALARA Program
Cooper Nuclear Station, Five Year Cumulative Radiation Exposure Reduction Plan, 2016-2020
Plant Configuration Log for Friday, February 15, 2015
Plant Configuration Log for Thursday, May 7, 2015
Plant Configuration Log for Wednesday, October 5, and Thursday, October 6

Procedures

Number	Title	Revision
9.ALARA.4	Radiation Work Permits	22
9.EN-RP-100	Radiation Worker Expectations	10
9.EN-RP-108	Radiation Protection Posting	12
9.EN-RP-110	ALARA Program	9
9.EN-RP-141	Job Coverage	17
9.RADOP.2	Radiation Safety Standards and Limits	16
9.RW.9	Filling Containers with Waste/Radioactive Material	17
9.RESP.1	Respiratory Protection Program	16
9.EN-RP-110-04	Radiation Protection Risk Assessment Process	10
9.EN-RP-110-05	ALARA Planning and Controls	4
9.EN-RP-110-06	Outage Dose Estimating and Tracking	0
9.EN-RP-141-01	Job Coverage using Remote Monitoring Technology	2
2.3_9-3-1	Alarm Procedure, Panel 9-3, Annunciator 9-3-1	36
4.8	Area Radiation Monitoring System	13

Radiation Work Permits

Number	Title	Revision
2016-416	LLRT Reactor Building and Outside Support	0
2016-439	Management, Engineering, NRC, QA, Safety, D/W Coordinators, Visitors etc., Outage Miscellaneous Walk-downs	0
2016-501	All Valve Work and support in Reactor Building LHRA's and the Radwaste 877 FT FD/WC Tk LHRA Room	0
2016-506	Under-vessel CRDMs Change-outs (also includes Keyway Worker and RP Under-vessel Coverage)	0
2016-507	CRDM De-torque/Torque, Label, Stabe/Demobe, PIPS, QC, etc., CRDM Transport Unload/Load Boxes. Dry Run	0
2016-511	ISI and FAC in Drywell and Steam Tunnel (includes Walk-downs, Oversight, Insulation Removal/Install, Surface Preps, Nozzle Door Operations)	0, 1
2016-516	LLRT in Reactor Building SWP Areas, LHRA and HCA, Not in Drywell or Steam Tunnel Areas	0
2016-540	Management, Engineering, NRC, Safety, D/W Coordinators, QA, etc., Walk-downs in Drywell and Steam Tunnel	1

Condition Reports (CRs)

CR-CNS-2016-01718	CR-CNS-2016-02913	CR-CNS-2016-03446	CR-CNS-2016-03573
CR-CNS-2016-03694	CR-CNS-2016-04165	CR-CNS-2016-06495	CR-CNS-2016-06496
CR-CNS-2016-06500			

Section 4OA1: Performance Indicator Verification

Miscellaneous Documents

Number	Title	Revision/Date
	CNS Mitigating System Performance Index (MSPI) Basis Document	7
	CNS MSPI Deviation Reports	October 1, 2015 – September 31, 2016
	CNS Station Logs	October 1, 2015 – September 31, 2016

Miscellaneous Documents

Number	Title	Revision/Date
	MSPI System Documentation and Data Review Forms	October 1, 2015 – September 31, 2016
99-02	NEI, Regulatory Assessment Performance Indicator Guideline	7

Procedures

Number	Title	Revision
0-EN-LI-114	Performance Indicator Process	21

Section 4OA2: Problem Identification and Resolution

Procedures

Number	Title	Revision
0.40.8	Circuit Evaluations	4
7.2.53.12	Cooper Bessemer Bolting and Torque Program	13
7.3.16	Low Voltage Relay Removal and Installation	22

Condition Reports (CRs)

CR-CNS-2016-03783	CR-CNS-2016-04485	CR-CNS-2016-04984	CR-CNS-2016-05123
CR-CNS-2016-05167	CR-CNS-2016-05219	CR-CNS-2016-05558	CR-CNS-2016-05628
CR-CNS-2016-05963	CR-CNS-2016-06901	CR-CNS-2016-07313	CR-CNS-2016-07359
CR-CNS-2016-07527	CR-CNS-2016-07649	CR-CNS-2016-08835	

Work Orders

5056956	5065847	5120798	5121083	5121085
5121125	5147447	5148640	5149082	5150017
5150817				

Section 4OA3: Follow-up of Events and Notices of Enforcement Discretion

Miscellaneous Documents

Number	Title	Revision
PRE-RE29-LOG#138	Scaffold Request and Evaluation Form – HV-MOV-264MV	June 29, 2016

Miscellaneous Documents

Number	Title	Revision

Procedures

Number	Title	Revision
0-EN-HU-106	Procedure and Work Instruction Use and Adherence	3C0
2.1.4	Normal Shutdown	152
2.1.9	Low Power Operation for Maintenance Activities (Hot Standby Condition)	67
2.1.22	Recovering From a Group Isolation	60
2.2.33	High Pressure Coolant Injection System	78
2.2.56	Main Steam System	49, 50
2.2.67	Reactor Core Isolation Cooling System	73
5.1Incident	Site Emergency Incident	35
5.7.1	Emergency Classification	56
6.PC.513	Main Steam Local Leak Rate Tests	26
6.RCS.601	Technical Specification Monitoring of RCS Heatup/Cooldown Rate	22
7.0.7	Scaffolding Construction and Control	34

Condition Reports (CRs)

CR-CNS-2015-00147	CR-CNS-2015-01115	CR-CNS-2015-02751	CR-CNS-2015-03348
CR-CNS-2015-04320	CR-CNS-2015-06202	CR-CNS-2016-05607	CR-CNS-2016-05611
CR-CNS-2016-05614	CR-CNS-2016-05835	CR-CNS-2016-06185	CR-CNS-2016-06310
CR-CNS-2016-06427	CR-CNS-2016-06431	CR-CNS-2016-06808	CR-CNS-2016-07634
CR-CNS-2016-07636			

Work Orders

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The following items are requested for the
Occupational Radiation Safety Inspection
at Cooper Nuclear Station
October 3 – 7, 2016
Integrated Report 2016004

Inspection areas are listed in the attachments below.

Please provide the requested information on or before **September 15, 2016**.

Please submit this information using the same lettering system as below. For example, all contacts and phone numbers for Inspection Procedure 71124.0**1** should be in a file/folder titled "**1**- A," applicable organization charts in file/folder "**1**- B," etc.

If information is placed on *ims.certrec.com*, please ensure the inspection exit date entered is at least 30 days later than the onsite inspection dates, so the inspectors will have access to the information while writing the report.

In addition to the corrective action document lists provided for each inspection procedure listed below, please provide updated lists of corrective action documents at the entrance meeting. The dates for these lists should range from the end dates of the original lists to the day of the entrance meeting.

If more than one inspection procedure is to be conducted and the information requests appear to be redundant, there is no need to provide duplicate copies. Enter a note explaining in which file the information can be found.

If you have any questions or comments, please contact John O'Donnell at (817) 200-1441 or john.odonnell@nrc.gov.

PAPERWORK REDUCTION ACT STATEMENT

This letter does not contain new or amended information collection requirements subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). Existing information collection requirements were approved by the Office of Management and Budget, control number 3150-0011.

1. Radiological Hazard Assessment and Exposure Controls (71124.01) and Performance Indicator Verification (71151)
Date of Last Inspection: May 18, 2015

A. List of contacts and telephone numbers for the Radiation Protection Organization Staff and Technicians

B. Applicable organization charts

C. Audits, self-assessments, and LERs written since date of last inspection, related to this inspection area

D. Procedure indexes for the radiation protection procedures

E. Please provide specific procedures related to the following areas noted below. Additional Specific Procedures may be requested by number after the inspector reviews the procedure indexes.

1. Radiation Protection Program Description
2. Radiation Protection Conduct of Operations
3. Personnel Dosimetry Program
4. Posting of Radiological Areas
5. High Radiation Area Controls
6. RCA Access Controls and Radiation Worker Instructions
7. Conduct of Radiological Surveys
8. Radioactive Source Inventory and Control
9. Declared Pregnant Worker Program

F. List of corrective action documents (including corporate and sub-tiered systems) since date of last inspection

a. Initiated by the radiation protection organization
b. Assigned to the radiation protection organization

NOTE: The lists should indicate the <u>significance level</u> of each issue and the <u>search criteria</u> used. Please provide in document formats which are "searchable" so that the inspector can perform word searches.

If not covered above, a summary of corrective action documents since date of last inspection involving unmonitored releases, unplanned releases, or releases in which any dose limit or administrative dose limit was exceeded (for Public Radiation Safety Performance Indicator verification in accordance with IP 71151)

G. List of radiologically significant work activities scheduled to be conducted during the inspection period (If the inspection is scheduled during an outage, please also include a list of work activities greater than 1 rem, scheduled during the outage with the dose estimate for the work activity.)

H. List of active radiation work permits

I. Radioactive source inventory list

   a. All radioactive sources that are required to be leak tested

   b. All radioactive sources that meet the 10 CFR Part 20, Appendix E, Category 2 and above threshold. Please indicate the radioisotope, initial and current activity (w/assay date), and storage location for each applicable source.

J. The last two leak test results for the radioactive sources inventoried and required to be leak tested. If applicable, specifically provide a list of all radioactive source(s) that have failed its leak test within the last two years

K. A current listing of any non-fuel items stored within your pools, and if available, their appropriate dose rates (Contact / @ 30cm)

L. Computer printout of radiological controlled area entries greater than 100 millirem since the previous inspection to the current inspection entrance date. The printout should include the date of entry, some form of worker identification, the radiation work permit used by the worker, dose accrued by the worker, and the electronic dosimeter dose alarm set-point used during the entry (for Occupational Radiation Safety Performance Indicator verification in accordance with IP 71151).

2. Occupational ALARA Planning and Controls (71124.02)
Date of Last Inspection: March 21, 2016

A. List of contacts and telephone numbers for ALARA program personnel

B. Applicable organization charts

C. Copies of audits, self-assessments, and LERs, written since date of last inspection, focusing on ALARA

D. Procedure index for ALARA Program

E. Please provide specific procedures related to the following areas noted below. Additional Specific Procedures may be requested by number after the inspector reviews the procedure indexes.

1. ALARA Program
2. ALARA Committee
3. Radiation Work Permit Preparation

F. A summary list of corrective action documents (including corporate and sub-tiered systems) written since date of last inspection, related to the ALARA program. In addition to ALARA, the summary should also address Radiation Work Permit violations, Electronic Dosimeter Alarms, and RWP Dose Estimates

NOTE: The lists should indicate the significance level of each issue and the search criteria used. Please provide in document formats which are "searchable" so that the inspector can perform word searches.

G. List of work activities greater than 1 rem, since date of last inspection, Include original dose estimate and actual dose.

H. Site dose totals and 3-year rolling averages for the past 3 years (based on dose of record)

I. Outline of source term reduction strategy

J. If available, provide a copy of the ALARA outage report for the most recently completed outages for each unit

K. Please provide your most recent Annual ALARA Report.