UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
612 EAST LAMAR BLVD, SUITE 400
ARLINGTON, TEXAS 76011-4125

August 11, 2011

David J. Bannister, Vice President
and Chief Nuclear Officer
Omaha Public Power District
Fort Calhoun Station FC-2-4
P. O. Box 550
Fort Calhoun, NE 68023-0550

Subject: FORT CALHOUN - NRC INTEGRATED INSPECTION REPORT NUMBER 05000285/2011003

Dear Mr. Bannister:

On June 30, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Fort Calhoun Station. The enclosed integrated inspection report documents the inspection findings, which were discussed on July 14, 2011, with Mr. T. Nellenbach, Plant Manager and other members of your staff.

The inspections examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents three NRC-identified findings (Green) and one self-revealing finding of very low safety significance (Green). All four of these findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as noncited violations, consistent with Section 2.3.2 of the NRC Enforcement Policy. If you contest the violation or the significance of the noncited violation, 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, D.C. 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 612 E. Lamar Blvd, Suite 400, Arlington, Texas, 76011-4125; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Fort Calhoun facility. In addition, if you disagree with the crosscutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV, and the NRC Resident Inspector at Fort Calhoun Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, and its enclosure, will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy or proprietary information so that it can be made available to the Public without redaction.

Sincerely,

/RA/

James Drake, Chief (Temporary)
Project Branch E
Division of Reactor Projects

Docket: 50-285
License: DPR-40

Enclosure:
NRC Inspection Report 05000285/2011003
w/Attachment: Supplemental Information

cc w/Enclosure:

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SUNSI Rev Compl	☒ Yes ☐ No	ADAMS	☒ Yes ☐ No	Reviewer Initials	JFM
Publicly Avail	☒ Yes ☐ No	Sensitive	☐ Yes ☒No	Sens. Type Initials	JFM

SRI:DRP/	RI:DRP/	SPE:DRP/	C:DRS/EB1	C:DRS/EB2
JCKirkland	JFWingebach	RVAzua	TRFarnholtz	NFO'Keefe
/E-JDrake for/	***/E-RA/***	***/JMelfi for/***	***/RA/***	***/SGraves for/***
8/10/2011	8/10/2011	8/1/2011	8/2/2011	8/3/2011
C:DRS/OB	C:DRS/PSB1	C:DRS/PSB2	C:DRS/TSB	actingC:DRP/
MSHaire	MCHay	GEWerner	DAPowers	JFDrake
/JLarson for/	***/RA/***	***/LCarson for/***	***/RA/***	***/RA/***
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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	05000285
License:	DPR-40
Report:	05000285/2011003
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	9610 Power Lane Blair, NE 68008
Dates:	April 1 through June 30, 2011
Inspectors:	J. Kirkland, Senior Resident Inspector J. Wingebach, Resident Inspector A. Fairbanks, Resident Inspector P. Elkman, Senior Emergency Preparedness Inspector L. Ricketson, Senior Health Physicist C. Alldredge, Health Physicist J. Melfi, Project Engineer I. Anchondo, Reactor Inspector M. Young, Reactor Inspector
Approved By:	James Drake, Chief, (Temporary) Project Branch E Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000285/2011003; 04/01/2011 – 06/30/2011; Fort Calhoun Station, Integrated Resident and Regional Report; Radiological Hazard Assessment and Exposure Controls, Surveillance Testing, Emergency Action Level and Emergency Plan Changes, Fire Protection, Operability Evaluation

The report covered a 3-month period of inspections by resident inspectors and announced baseline inspections by region-based inspectors. Three NRC-identified findings (Green) and one self-revealing finding of very low safety significance (Green) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." The crosscutting aspect is determined using Inspection Manual Chapter 0310, "Components within the Crosscutting Areas." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix R, Section III.O for the failure to ensure an adequate seismic design of the reactor coolant pumps oil collection system. The licensee used 2-inch copper pipe with brazed joints in the lube oil collection system. The seismic analysis of the system assumed the use of ASME Section IX during the installation of the system, but no codes or standards were used by the licensee for the brazed joints.

  The inspectors determined that the failure to design and install an adequate oil collection system which included provisions for the drain lines to the oil collection tank was a performance deficiency. This finding had a credible impact on safety because the inadequate installation and design of the oil collection systems presented a degradation of a fire confinement component, which had a fire prevention function of not allowing an oil leak. The inspectors determined the finding was more than minor because it impacted the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences, and the related attribute of protection against external factors, such as a fire. The inspectors reviewed Inspection Manual Chapter 0609, Appendix F, and determined the finding was of very low safety significance, because of the low degradation rating of the fire confinement category related to the as found condition of the oil collection piping, the extremely low frequency of reactor coolant pump oil leaks, minor actual reactor coolant pump oil leaks during the past operating cycle, and other area fire protection defense-in-depth features such as automatic fire detection, manual suppression capability, and safe shutdown capability from the main control room. This finding involved a legacy issue associated with a modification for original installation; therefore, there were no assigned cross-cutting aspects (Section 1R05).

- Green. A self-revealing noncited violation of Fort Calhoun Technical Specification 5.8.1, "Procedures," occurred due to the failure of the licensee to ensure that adequate procedures were available for maintenance which was conducted on the reactor protective systems power supplies. Specifically, there was no procedural guidance to require replacement of power supplies, or an engineering justification for continued operation, once power supplies exceeded their vendor recommended life, and/or showed signs of failure and degradation.

  The inspectors determined that the licensee's failure to provide procedural guidance to evaluate and/or replace age-degraded components was a performance deficiency. This was a result of the licensee's failure to properly implement a required procedure, and was within the licensee's ability to foresee and correct and could have been prevented. This performance deficiency was more than minor because it could be reasonably viewed as a precursor to a significant event, it could lead to a loss of the reactor protective system. The inspectors evaluated this finding using Inspection Manual Chapter 0609, Attachment 4, and determined that this finding was associated with the Mitigating Systems Cornerstone, specifically the primary degraded reactivity control contributor. Because this finding occurred while the unit was operating at full power, the inspectors used Inspection Manual Chapter 0609 to determine its significance. The inspectors determined that the finding represented a qualification deficiency confirmed not to result in a loss of functionality because none of the failures to date prevented a reactor protective systems channel from tripping. Therefore, in accordance with the Phase 1 screening, the finding was of very low risk significance.

  This finding had a crosscutting aspect in the area of problem identification and resolution associated with the component of operating experience because the licensee failed to adequately evaluate and communicate relevant internal and external operator experience [P.2(a)](Section 4OA2).

- Green. The inspectors identified a noncited violation of Technical Specification 5.8.1.a for failure to follow scaffold specification and construction Procedures SO-M-35 and PED-CSS-12. This led to the licensee declaring a number of emergency core cooling components inoperable and entering technical specification 2.0.1.

  The inspectors determined that not following a procedure required by Technical Specification 5.8.1.a was a performance deficiency. The finding was more than minor because if left uncorrected it would have the potential to lead to a more significant safety concern. The licensee routinely failed to perform seismic evaluations of scaffolds erected near safety-related equipment not constructed in accordance with Procedures PED-CSS-12 or SO-M-35 for preconfigured seismic scaffolding. The finding was associated with the Mitigation Systems Cornerstone while the reactor was operating; therefore, Inspection Manual Chapter 0609, Attachment 4 screening checklist was used. The finding was determined to have very low safety significance because it did not involve the total loss of any safety function, and did not contribute to external event initiated core damage accident sequences. The inspectors determined the primary cause of the finding was lack of the licensee's oversight of the scaffolding program. The finding had a crosscutting aspect in the area of human performance, specifically, work practices, in that, the licensee failed to ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety was supported [H.4(c)](Section 1R15).

Cornerstone: Occupational Radiation Safety

- Green. Inspectors identified a noncited violation of Technical Specification 5.8.1a for the failure to follow procedural requirements to plan and carry out decontamination work in the spent fuel pool transfer canal. On January 24, 2011, decontamination work was performed in the spent fuel pool transfer canal, using Radiation Work Permit 11-3317. While planning and controlling the work, the licensee failed to follow multiple procedure steps. Specifically, the licensee did not prepare an ALARA planning worksheet as the initial step of generating the radiation work permit, did not document justification for changing the electronic dosimeter set points which were eventually determined to be inappropriate, and did not perform an ALARA briefing before the entries were made into the spent fuel pool transfer canal, which was posted as a restricted locked high radiation area. The inspectors also determined that there were aspects of the procedure that contained vague expectations, which contributed to decisions being made without using the procedure.

  The failure to follow a procedure was a performance deficiency. The finding was more than minor because it negatively impacted the Occupational Radiation Safety Cornerstone's attribute of program and process, in that, by not following the procedure; radiological safety attributes built into the radiation work permit program were circumvented. Using Inspection Manual Chapter 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," the inspectors determined that the violation was of very low safety significance because: (1) it was not associated with ALARA planning or work controls, (2) there was no overexposure, (3) there was no substantial potential for an overexposure, and (4) the ability to assess dose was not compromised. This deficiency had a crosscutting aspect in the area of human performance related to work practices. Specifically, the licensee did not communicate human error prevention techniques, such as, holding pre-job briefs, self- and peer- checking, and proper documentation of activities [H.4.a](Section 2RS02).

B. Licensee-Identified Violations

None

REPORT DETAILS

Summary of Plant Status

The unit began the assessment period at 100 percent power. On April 9, 2011, the unit shut down for a refueling outage. The unit remained shut down for the remainder of the assessment period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R01 Adverse Weather Protection (71111.01)

.1 Readiness for Impending Adverse Weather Conditions

a. Inspection Scope

Since thunderstorms with potential tornados and high winds were forecast in the vicinity of the facility for June 19, 2011, the inspectors reviewed the plant personnel's overall preparations/protection for the expected weather conditions. On June 19, 2011, the inspectors walked down the off-site electrical distribution system because their safety-related functions could be affected, or required, as a result of high winds or tornado-generated missiles or the loss of offsite power. The inspectors evaluated the plant staff's preparations against the site's procedures and determined that the staff's actions were adequate. During the inspection, the inspectors focused on plant-specific design features and the licensee's procedures used to respond to specified adverse weather conditions. The inspectors also toured the plant grounds to look for any loose debris that could become missiles during a tornado. The inspectors' evaluated operator staffing and accessibility of controls and indications for those systems required to control the plant. Additionally, the inspectors reviewed the Updated Safety Analysis Report and performance requirements for the systems selected for inspection, and verified that operator actions were appropriate as specified by plant-specific procedures. The inspectors also reviewed a sample of corrective action program items to verify that the licensee identified adverse weather issues at an appropriate threshold and dispositioned them through the corrective action program in accordance with station corrective action procedures. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one (1) impending adverse weather sample as defined in Inspection Procedure 71111.01-05.

b. Findings

No findings were identified.

.2 Readiness to Cope with External Flooding

a. Inspection Scope

The inspectors evaluated the design, material condition, and procedures for coping with the design basis probable maximum flood. The evaluation included a review to check for deviations from the descriptions provided in the Updated Safety Analysis Report for features intended to mitigate the potential for flooding from external factors. As part of this evaluation, the inspectors checked for obstructions that could prevent draining, checked that the roofs did not contain obvious loose items that could clog drains in the event of heavy precipitation, and determined that barriers required to mitigate the flood were in place and operable. Additionally, the inspectors performed an inspection of the protected area to identify any modification to the site that would inhibit site drainage during a probable maximum precipitation event or allow water ingress past a barrier. The inspectors also reviewed the abnormal operating procedure for mitigating the design basis flood to ensure it could be implemented as written. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one (1) external flooding sample as defined in Inspection Procedure 71111.01-05.

b. Findings

No findings were identified.

1R04 Equipment Alignments (71111.04)

.1 Partial Walkdown

a. Inspection Scope

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

- April 17, 2011, Portions of the Shutdown Cooling System prior to draining to mid loop

- April 26, 2011, Portions of Raw Water and Component Cooling Water systems with one raw water header out of service

The inspectors selected these systems based on their risk significance relative to the reactor safety cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could affect the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, Updated Safety Analysis Report, technical specification requirements, administrative technical specifications, outstanding work orders, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also inspected accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program with the appropriate significance characterization. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of two (2) partial system walkdown samples as defined in Inspection Procedure 71111.04-05.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Fire Inspection Tours

a. Inspection Scope

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

- May 17, 2011, Fire Area 20.1, Personnel Air Lock Area (Room 58) and Corridor Auxiliary Building Main Floor (Room 26)
- May 18, 2011, Fire Area 23, Pipe Penetration Area (Room 59)
- May 18, 2011, Fire Area 24, Sampling Area (Room 60)
- May 19, 2011, Fire Area 30, Containment (Room 1)
- May 19, 2011, Fire Area 31, Intake Structure (Intake)

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

These activities constitute completion of five (5) quarterly fire-protection inspection samples as defined in Inspection Procedure 71111.05-05.

b. Findings

Failure to Adequately Design and the Oil Collection system for the Reactor Coolant Pump Motor Lube Oil

Introduction. The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix R, Section III.O which involved a failure to ensure an adequate seismic design of the reactor coolant pumps oil collection system.

Description. On May 19, 2011, during tours of containment, the inspectors questioned the design of the installed reactor coolant pump lube oil collection piping. The lube oil collection system consisted of stainless steel sheet metal pans around each reactor coolant pump with a drain hole in each pan, a rubber flexible coupling connecting it to drain pipes with pipe clamps, and routed via drain pipes to oil collection tanks. The drainpipes in question are 2-inch brazed copper pipes, installed in approximately 1980. There are two drain tanks, with two reactor coolant pumps draining to a tank.

The reactor coolant pump lube oil collection system is necessary to confine any oil discharged due to leakage or failure of the lubrication system and prevent it from becoming a fire hazard by draining it to a safe location. A lube oil fire in containment would increase containment air temperature and could affect the operability of safety-related equipment in containment. Accessibility into containment after a fire is limited. Due to these concerns, the NRC required a reactor coolant pump oil collection system, as noted in 10 CFR Part 50, Appendix R Section III.O, "Oil Collection System For Reactor Coolant Pump" for all plants. This rule states in part that there should be reasonable assurance that the oil collection system could withstand a safe shutdown earthquake. This rule became effective in December 1980.

The licensee installed the modification via design package MR-FC-78-057. The initial package was sent to the NRC for review, and the NRC accepted that the licensee was to install a reactor coolant pump lube oil collection system. The design was approved in NRC's Safety Evaluation Report dated November 17, 1980, which reviewed the June 6, 1979 letter to the Commission describing the licensee's system. The design package did not specify codes for the system to meet. The inspectors were informed no codes were required, and the system is not quality-related.

The inspector's review of the June 6, 1979, letter revealed that the use of copper drain lines was not discussed. ASME B31.1-1967, "Power Piping," and NFPA 30-1973, "Flammable and Combustible Liquids Code," prohibit the use of copper pipe with brazed joints for flammable or combustible liquids.

10 CFR Part 50, Appendix R, Section III.O requires that "The oil collection system shall be so designed, engineered, and installed that failure will not lead to fire during normal or design basis accident conditions and that there is reasonable assurance that the system will withstand the Safe Shutdown Earthquake." The licensee did an analysis to show that the installed oil collection system would survive an earthquake, via Calculation FC06645. Section 5.8 of Calculation FC06645 discusses the qualification of the braces joints, and makes several assumptions to qualify the strength of the brazed joints to ensure that the joints are stronger that the base material. These assumptions included statements that brazed joints were completed by qualified personnel under the requirements of ASME section IX, "Qualification Standards for Welding and Brazing Procedures, Welders, Brazers, and Welding and Brazing Operators." The inspectors were informed that ASME section IX was not followed. The inspectors were also informed that the brazed joints were not qualified to any standard. The inspectors also reviewed design change package MRFC 78-057. Quality control checks for this piping were initially specified, but were not done and the closeout review did not believe the check to be necessary. Since no standards were enforced for the brazed joints, and the design calculation assumed ASME Code Section IX standards were met for the brazed joints, the inspectors concluded there is not reasonable expectation that the system was seismically qualified.

Analysis. The inspectors determined that the failure to design and install an adequate oil collection system which included provisions for the drain lines to the oil collection tank was a performance deficiency. This finding had a credible impact on safety because the inadequate installation and design of the oil collection systems presented a degradation of a fire confinement component which has a fire prevention function to survive a safe shutdown earthquake. The inspectors determined the finding was more than minor because it impacted the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences, and the related attribute of protection against external factors, such as a fire. The inspectors reviewed Inspection Manual Chapter 0609, Appendix F, and determined the finding was of very low safety significance (Green), because of the low degradation rating of the fire confinement category related to the as found condition of the oil collection piping, the extremely low frequency of reactor coolant pump oil leaks, minor actual reactor coolant pump oil leaks during the past operating cycle, and other area fire protection defense-in-depth features such as automatic fire detection, manual suppression capability, and safe shutdown capability from the main control room. This finding involved a legacy issue associated with a modification for original installation; therefore, there were no assigned cross cutting aspects.

Enforcement. 10 CFR 50, Appendix R, Section III.O requires that “The oil collection system shall be so designed, engineered, and installed that failure will not lead to fire during normal or design basis accident conditions and that there is reasonable assurance that the system will withstand the Safe Shutdown Earthquake.” The drain lines for the oil collection system includes the use of 2-inch copper pipe with brazed joints. The seismic analysis of the oil collection system assumed that the brazed joints were qualified by the use of ASME Section IX (1974), and would be stronger than the base metal. However, ASME Section IX was not used during the construction of the system, and no testing of the joint strength was documented. Contrary to the above, on May 19, 2011, the licensee failed to adequately design and install an adequate oil collection system. Because the finding was of very low safety significance and had been entered into the licensee’s corrective action program as CR 2011-5992 this violation was treated as a noncited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2011003-01, “Failure to Adequately Design a Reactant Coolant Pump Lube Oil Collection System.”

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors reviewed the Updated Safety Analysis Report, the flooding analysis, and plant procedures to assess susceptibilities involving internal flooding; reviewed the corrective action program to determine if licensee personnel identified and corrected flooding problems; inspected underground bunkers/manholes to verify the adequacy of sump pumps, level alarm circuits, cable splices subject to submergence, and drainage for bunkers/manholes; and verified that operator actions for coping with flooding can reasonably achieve the desired outcomes. The inspectors also inspected the areas listed below to verify the adequacy of equipment seals located below the flood line, floor and wall penetration seals, watertight door seals, common drain lines and sumps, sump pumps, level alarms, and control circuits, and temporary or removable flood barriers. Specific documents reviewed during this inspection are listed in the attachment.

- May 17, 2011, Charging Pump Room Area

These activities constitute completion of one (1) flood protection measures inspection sample as defined in Inspection Procedure 71111.06-05.

b. Findings

No findings were identified.

1R08 In-service Inspection Activities (71111.08)

.1 Inspection Activities Other Than Steam Generator Tube Inspection, Pressurized Water Reactor Vessel Upper Head Penetration Inspections, and Boric Acid Corrosion Control (71111.08-02.01)

a. Inspection Scope.

The inspectors observed five (5) nondestructive examination activities and reviewed one (1) nondestructive examination activity that included three (3) types of examinations. The licensee did not identify any relevant indications accepted for continued service during the nondestructive examinations.

The inspectors directly observed the following nondestructive examinations:

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Emergency Core Cooling	Rod and Clevis (21-PR-1/12-LPSI-2003)	Visual (VT-3)
Shutdown Cooling	Pipe to Elbow Weld 12 (12-SDC-2003)	Ultrasonic
Shutdown Cooling	Elbow to Elbow Weld 13 (12-SDC-2003)	Ultrasonic
Safety Injection	Pipe to Valve HCV-230 Weld 09 (2-CH-28)	Liquid Penetrant
Safety Injection	Weld 06 (2-CH-28)	Liquid Penetrant

The inspectors reviewed records for the following nondestructive examinations:

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Shutdown Cooling	Elbow to Pipe Weld 14 (12-SDC-2003)	Ultrasonic

During the review and observation of each examination, the inspectors verified that activities were performed in accordance with the ASME Code requirements and applicable procedures. The inspectors also verified that qualifications of all nondestructive examination technicians performing the inspections were current.

These actions constitute completion of the requirements for Section 02.01.

b. Findings.

No findings were identified.

.2 Vessel Upper Head Penetration Inspection Activities (71111.08-02.02)

a. Inspection Scope.

The inspectors reviewed the results of the licensee's bare metal visual inspection of the reactor vessel upper head penetrations and verified that there was no evidence of boric acid challenging the structural integrity of the reactor head components and attachments. The inspectors also verified that the required inspection coverage was achieved and limitations were properly recorded. The inspectors verified that personnel performing the inspection were certified examiners to their respective nondestructive examination method. Specific documents reviewed during this inspection are listed in the attachment.

These actions constitute completion of the requirements for Section 02.02.

b. Findings.

No findings were identified.

.3 Boric Acid Corrosion Control Inspection Activities (71111.08-02.03)

a. Inspection Scope.

The inspectors evaluated the implementation of the licensee's boric acid corrosion control program for monitoring degradation of those systems that could be adversely affected by boric acid corrosion. The inspectors reviewed the documentation associated with the licensee's boric acid corrosion control walkdown as specified in Procedure PBD-10, "Boric Acid Corrosion Control Program," Revision 14, and SE-EQT-MX-002, "Carbon Steel and Low Alloy Steel Fasteners In-service Testing Inspections," Revision 12. The inspectors also reviewed the visual records of the components and equipment. The inspectors verified that the visual inspections emphasized locations where boric acid leaks could cause degradation of safety-significant components. The inspectors also verified that the engineering evaluations for those components where boric acid was identified gave assurance that the ASME Code wall thickness limits were properly maintained. The inspectors confirmed that the corrective actions performed for evidence of boric acid leaks were consistent with requirements of the ASME Code. Specific documents reviewed during this inspection are listed in the attachment.

These actions constitute completion of the requirements for Section 02.03.

b. Findings.

No findings were identified.

.4 Steam Generator Tube Inspection Activities (71111.08-02.04)

a. Inspection Scope.

The licensee did not perform tube inspection activities on the steam generators. The next inspection is scheduled for the 2012 fall outage.

These actions constitute completion of the requirements of Section 02.04.

b. Findings.

No findings were identified.

.5 Identification and Resolution of Problems (71111.08-02.05)

a. Inspection scope.

The inspectors reviewed 23 condition reports which dealt with in-service inspection activities and found the corrective actions for in-service inspection issues were appropriate. The specific condition reports reviewed are listed in the documents reviewed section. From this review, the inspectors concluded that the licensee had an appropriate threshold for entering in-service inspection issues into the corrective action program, and has procedures that direct a root cause evaluation when necessary. The licensee also has an effective program for implementing industry-operating experience related to in-service inspection activities. Specific documents reviewed during this inspection are listed in the attachment.

These actions constitute completion of the requirements of Section 02.05.

These activities constitute completion of one (1) sample as defined by 71111.08-05.

b. Findings.

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

On June 14, 2011, the inspectors observed a crew of licensed operators in the plant's simulator to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- Licensed operator performance
- Crew's clarity and formality of communications

- Crew's ability to take timely actions in the conservative direction
- Crew's prioritization, interpretation, and verification of annunciator alarms
- Crew's correct use and implementation of abnormal and emergency procedures
- Control board manipulations
- Oversight and direction from supervisors
- Crew's ability to identify and implement appropriate technical specification actions and emergency plan actions and notifications

The inspectors compared the crew's performance in these areas to preestablished operator action expectations and successful critical task completion requirements. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one (1) quarterly licensed-operator requalification program sample as defined in Inspection Procedure 71111.11.

b. <u>Findings</u>

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

a. <u>Inspection Scope</u>

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

- June 16, 2011, Maintenance Rule scoping of the Reactor Protective System Power Supplies
- June 16, 2011, a(4) status of 480 volt bus feeder breaker 1B3A

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

- Implementing appropriate work practices
- Identifying and addressing common cause failures
- Scoping of systems in accordance with 10 CFR 50.65 (b)

- Characterizing system reliability issues for performance
- Charging unavailability for performance
- Trending key parameters for condition monitoring
- Ensuring proper classification in accordance with 10 CFR 50.65 (a)(1) or -(a)(2)
- Verifying appropriate performance criteria for structures, systems, and components classified as having an adequate demonstration of performance through preventive maintenance, as described in 10 CFR 50.65(a)(2), or as requiring the establishment of appropriate and adequate goals and corrective actions for systems classified as not having adequate performance, as described in 10 CFR 50.65 (a)(1)

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the corrective action program with the appropriate significance characterization. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of two (2) quarterly maintenance effectiveness samples as defined in Inspection Procedure 71111.12-05.

b. Findings

No findings were identified.

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

a. Inspection Scope

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

- April 17, 2011, Risk management actions associated with drain down to mid-loop
- April 21, 2011, Raw water flooding concern while in lowered inventory
- May 20, 2011, Component Cooling Water maintenance that resulted in cavitation, no risk assessment
- May 31, 2011, Risk management actions associated with onset of Missouri River flooding

The inspectors selected these activities based on potential risk significance relative to the reactor safety cornerstones. As applicable for each activity, the inspectors verified that licensee personnel performed risk assessments as required by 10 CFR 50.65 (a)(4) and that the assessments were accurate and complete. When licensee personnel performed emergent work, the inspectors verified that the licensee personnel promptly assessed and managed plant risk. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed the technical specification requirements and inspected portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of four (4) maintenance risk assessments and emergent work control inspection samples as defined in Inspection Procedure 71111.13-05.

b. <u>Findings</u>

No findings were identified.

1R15 Operability Evaluations (71111.15)

a. <u>Inspection Scope</u>

The inspectors reviewed the following issues:

- April 1, 2011, Reactor Coolant Pump casing gasket functionality
- April 4, 2011, Operability of Emergency Core Cooling System pumps due to scaffolding interference
- April 12, 2011, Operability of Power Operated Relief Valve PCV-102-1 after auto-closure while the switch was in the open position
- May 20, 2011, Component Cooling Water Pump AC-3B operability when depressurizing system
- June 26, 2011, Operability of reactor coolant system leak detection systems

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

These activities constitute completion of five (5) operability evaluations inspection samples as defined in Inspection Procedure 71111.15-04

b. Findings

Introduction. The inspectors identified a Green noncited violation of Technical Specification 5.8.1.a for failure to follow scaffold specification and construction Procedures SO-M-35 and PED-CSS-12.

Description. On March 30, 2011, during a walkdown of Room 81 and Room 19, the inspectors identified scaffolding erected in close proximity of safety related equipment that did not appear to be in accordance with plant procedures. Specifically, Procedure PED-CSS-12, "Standard Specification for Scaffold Construction" and Procedure SO-M-35, "Scaffolding Installation Control," required engineering evaluations to verify seismic qualifications for scaffolds which were not erected using previously evaluated and seismically qualified designs when near safety related equipment. The previously evaluated and seismically qualified designs are freestanding scaffolds that require specific standoff distances and do not require tie offs. None of the scaffolds the inspectors observed that were built around safety related equipment had been constructed using the previously evaluated and seismically qualified designs in Procedure PED-CSS-12. No evaluations for seismic adequacy had been performed. The licensee created Condition Report 2011-2399 and modified or removed the scaffolding questioned by the inspectors. The licensee did not perform an extent of condition inspection of other scaffolds.

On April 1, 2011, the shift manager created Condition Report 2011-2480 requesting clarification and guidance regarding scaffold installation near safety related equipment. On April 4, 2011 the inspectors expressed a concern to the shift manager regarding scaffolding erected in Rooms 21 and 22 in close proximity to emergency core cooling equipment. Condition Report 2011-2522 was created and the scaffolds were inspected by a structural engineer because they had not been previously evaluated. The scaffolds were determined to not be seismically qualified. The plant entered Technical Specification 2.0.1 and immediately removed the inadequate scaffolding.

The licensee performed an extent of condition inspection of all scaffolding in safety-related areas; as no engineering evaluations had been performed for scaffolds that were not erected in accordance with the Procedure PED-CSS-12 seismically qualified designs.

The licensee's scaffolding program was implemented by a contractor employed by the licensee. Specifically, one person in the contractor organization was responsible for the administrative portion of the program. When asked by the inspectors, few on-shift personnel had even a basic understanding of the scaffolding requirements contained in Procedures PED-CSS-12 and SO-M-35.

Analysis. The inspectors determined that not following a procedure required by Technical Specification 5.8.1.a, was a performance deficiency. The finding was more than minor because if left uncorrected it would have the potential to lead to a more significant safety concern. The licensee routinely failed to perform seismic evaluations of scaffolds erected near safety-related equipment when not constructed in accordance with Procedures PED-CSS-12 or SO-M-35 for preconfigured seismic scaffolding. The finding was associated with the Mitigation Systems Cornerstone while the reactor was operating; therefore Inspection Manual Chapter 0609, Attachment 4 screening checklist was used. The finding was determined to have very low safety significance because it did not involve the total loss of any safety function, or contribute to external event initiated core damage accident sequences. The inspectors determined the primary cause of the finding was lack of the licensee's oversight of the scaffolding program. The finding had a crosscutting aspect in the area of human performance, specifically, work practices in that the licensee failed to ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety is supported [H.4(c)].

Enforcement. Technical Specification 5.8.1.a requires that written procedures and administrative policies shall be established, implemented and maintained covering applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978. Appendix A, Section 9, paragraph A of Regulatory Guide 1.33 requires, in part, "Maintenance that can affect the performance of safety-related equipment should be preplanned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances." The licensee failed to follow procedural requirements to evaluate scaffolding installed in close proximity to safety related equipment for seismic considerations. On April 1, 2011, when notified of the concern, the licensee generated Condition Report 2011-2522 and removed the scaffolding that evening. Because the violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation was treated as a noncited violation, consistent with the Enforcement Policy: NCV 05000285/2011003-02, "Failure to Follow Scaffolding Procedure."

1R18 Plant Modifications (71111.18)

.1 Temporary Modifications

a. Inspection Scope

To verify that the safety functions of important safety systems were not degraded, the inspectors reviewed the temporary modification to modify the configuration of the Polar Crane (HE-1) main hoist motor driver resistors. Specifically, to bypass failed resistor grids and restore main hoist operation to a normal fast speed and a slightly increased slow speed.

The inspectors reviewed the temporary modification and the associated safety-evaluation screening against the system design bases documentation, including the Updated Safety Analysis Report and the technical specifications, and verified that the modification did not adversely affect the system operability/availability. The inspectors also verified that the installation and restoration were consistent with the modification documents and that configuration control was adequate. Additionally, the inspectors verified that the temporary modification was identified on control room drawings, appropriate tags were placed on the affected equipment, and licensee personnel evaluated the combined effects on mitigating systems and the integrity of radiological barriers.

These activities constitute completion of one (1) sample for temporary plant modifications as defined in Inspection Procedure 71111.18-05.

b. Findings

No findings were identified.

.2 Permanent Modifications

The inspectors reviewed key parameters associated with energy needs, materials, replacement components, timing, heat removal, control signals, equipment protection from hazards, operations, flow paths, pressure boundary, ventilation boundary, structural, process medium properties, licensing basis, and failure modes for the permanent modification identified as voltage regulator replacement for Emergency Diesel Generator 2.

The inspectors verified that modification preparation, staging, and implementation did not impair emergency/abnormal operating procedure actions, key safety functions, or operator response to loss of key safety functions; post-modification testing will maintain the plant in a safe configuration during testing by verifying that unintended system interactions will not occur; systems, structures and components' performance characteristics still meet the design basis; the modification design assumptions were appropriate; the modification test acceptance criteria will be met; and licensee personnel identified and implemented appropriate corrective actions associated with permanent plant modifications. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one (1) sample for permanent plant modifications as defined in Inspection Procedure 71111.18-05.

b. Findings

No findings were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the following postmaintenance activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- May 18, 2011, Postmaintenance test of Low Pressure Safety Injection Pump, SI-1B, suction valve actuator rebuild
- May 19, 2011, Postmaintenance test following replacement of positioner relay on Containment Cooling Coil Component Cooling Water Outlet Valve, HCV-402C
- May 25, 2011, Postmaintenance test following replacement of Electro-Pneumatic Transducer on Containment Spray Header Isolation Valve, HCV-344
- May 31, 2011, Postmaintenance test of Emergency Diesel Generator 1 following voltage regulator replacement

The inspectors selected these activities based upon the structure, system, or component's ability to affect risk. The inspectors evaluated these activities for the following (as applicable):

- The effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed
- Acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate

The inspectors evaluated the activities against the technical specifications, the Updated Safety Analysis Report, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with postmaintenance tests to determine whether the licensee was identifying problems and entering them in the corrective action program and that the problems were being corrected commensurate with their importance to safety. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of four (4) postmaintenance testing inspection samples as defined in Inspection Procedure 71111.19-05.

b. Findings

No findings were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the outage safety plan and contingency plans for the refueling outage which began on April 9, 2011, to confirm that licensee personnel had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense in depth. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below.

- Configuration management, including maintenance of defense in depth, is commensurate with the outage safety plan for key safety functions and compliance with the applicable technical specifications when taking equipment out of service.
- Clearance activities, including confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing.
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error.
- Status and configuration of electrical systems to ensure that technical specifications and outage safety-plan requirements were met, and controls over switchyard activities.
- Monitoring of decay heat removal processes, systems, and components.
- Verification that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system.
- Reactor water inventory controls, including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss.
- Controls over activities that could affect reactivity.
- Maintenance of secondary containment as required by the technical specifications.
- Refueling activities, including fuel handling and sipping to detect fuel assembly leakage.
- Licensee identification and resolution of problems related to refueling outage activities.

Specific documents reviewed during this inspection are listed in the attachment.

These activities counts towards completion of the refueling outage and other outage inspection sample as defined in Inspection Procedure 71111.20-05, which will be closed in a later inspection report.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the Updated Safety Analysis Report, procedure requirements, and technical specifications to ensure that the surveillance activities listed below demonstrated that the systems, structures, and/or components tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the significant surveillance test attributes were adequate to address the following:

- Preconditioning
- Evaluation of testing impact on the plant
- Acceptance criteria
- Test equipment
- Procedures
- Jumper/lifted lead controls
- Test data
- Testing frequency and method demonstrated technical specification operability
- Test equipment removal
- Restoration of plant systems
- Fulfillment of ASME Code requirements
- Updating of performance indicator data
- Engineering evaluations, root causes, and bases for returning tested systems, structures, and components not meeting the test acceptance criteria were correct

- Reference setting data
- Annunciators and alarms setpoints

The inspectors also verified that licensee personnel identified and implemented any needed corrective actions associated with the surveillance testing.

- April 7, 2011, Review of QC-ST-ECCS-0001, Quarterly ECCS Gas Accumulation Detection, which was performed on February 24, 2011
- April 8, 2011, Component Cooling Water Pump, AC-3C, Inservice Test
- April 9, 2011, QC-ST-ECCS-0002, Refueling ECCS Gas Accumulation Detection
- April 19, 2011, Containment Isolation Valve HCV-383-3 Local Leak Rate Test, IC-ST-AE-3833 (Containment Isolation Valve)
- May 21, 2011, Safety Injection System Remote Position Indicator Verification Surveillance Test

Activities were performed during the review of the refueling and quarterly emergency core cooling systems gas accumulation detection surveillances that were associated with Temporary Instruction 2515/177, “Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems.”

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of five (5) surveillance testing inspection samples as defined in Inspection Procedure 71111.22-05.

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

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

a. Inspection Scope

The inspectors performed an in-office onsite review of the Fort Calhoun Radiological Emergency Response Plan, Section A, “Assignment of Organizational Responsibility,” Revision 13, and Appendix A, “Letters of Agreement,” Revision 21. These revisions added the Fremont Fire Department to perform radiological monitoring and decontamination at the Fremont Middle School Reception Center.

These revisions were compared to their previous revisions, 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, and to the standards in 10 CFR 50.47 (b) to determine if the revisions adequately implemented the requirements of 10 CFR 50.54 (q). This review was not documented in a safety evaluation report and did not constitute approval of licensee-generated changes; therefore, these revisions are subject to future inspection.

These activities constitute completion of one (1) sample as defined in Inspection Procedure 71114.04-05.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Occupational and Public Radiation Safety

2RS01 Radiological Hazard Assessment and Exposure Controls (71124.01)

a. Inspection Scope

This area was inspected to: (1) review and assess licensee's performance in assessing the radiological hazards in the workplace associated with licensed activities and the implementation of appropriate radiation monitoring and exposure control measures for both individual and collective exposures, (2) verify the licensee is properly identifying and reporting Occupational Radiation Safety Cornerstone performance indicators, and (3) identify those performance deficiencies that were reportable as a performance indicators and which may have represented a substantial potential for overexposure of the worker.

The inspectors used the requirements in 10 CFR Part 20, the technical specifications, and the licensee's procedures required by technical specifications as criteria for determining compliance. During the inspection, the inspectors interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspectors performed walkdowns of various portions of the plant, performed independent radiation dose rate measurements, and reviewed the following items:

- Performance indicator events and associated documentation reported by the licensee in the Occupational Radiation Safety Cornerstone

- The hazard assessment program, including a review of the licensee's evaluations of changes in plant operations and radiological surveys to detect dose rates, airborne radioactivity, and surface contamination levels

- Instructions and notices to workers, including labeling or marking containers of radioactive material, radiation work permits, actions for electronic dosimeter alarms, and changes to radiological conditions

- Programs and processes for control of sealed sources and release of potentially contaminated material from the radiologically controlled area, including survey performance, instrument sensitivity, release criteria, procedural guidance, and sealed source accountability

- Radiological hazards control and work coverage, including the adequacy of surveys, radiation protection job coverage, and contamination controls; the use of electronic dosimeters in high noise areas; dosimetry placement; airborne radioactivity monitoring; controls for highly activated or contaminated materials (non-fuel) stored within spent fuel and other storage pools; and posting and physical controls for high radiation areas and very high radiation areas

- Radiation worker and radiation protection technician performance with respect to radiation protection work requirements

- Audits, self-assessments, and corrective action documents related to radiological hazard assessment and exposure controls since the last inspection

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of the one (1) required sample as defined in Inspection Procedure 71124.01-05.

b. Findings

No findings were identified.

2RS02 Occupational ALARA Planning and Controls (71124.02)

a. Inspection Scope

This area was inspected to assess performance with respect to maintaining occupational individual and collective radiation exposures as low as is reasonably achievable (ALARA). The inspectors used the requirements in 10 CFR Part 20, the technical specifications, and the licensee's procedures required by technical specifications as criteria for determining compliance. During the inspection, the inspectors interviewed licensee personnel and reviewed the following items:

- Site-specific ALARA procedures and collective exposure history, including the current 3-year rolling average, site-specific trends in collective exposures, and source-term measurements.

- ALARA work activity evaluations/post job reviews, exposure estimates, and exposure mitigation requirements.

- The methodology for estimating work activity exposures, the intended dose outcome, the accuracy of dose rate and man-hour estimates, and intended versus actual work activity doses and the reasons for any inconsistencies.

- Records detailing the historical trends and current status of tracked plant source terms and contingency plans for expected changes in the source term due to changes in plant fuel performance issues or changes in plant primary chemistry.

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

- Audits, self-assessments, and corrective action documents related to ALARA planning and controls since the last inspection.

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of the one (1) required sample as defined in Inspection Procedure 71124.02-05.

b. Findings

Introduction. Inspectors identified a Green noncited violation of Technical Specification 5.8.1a for the failure to follow procedure requirements to plan and carry out decontamination work in the spent fuel pool transfer canal.

Description. On January 24, 2011, decontamination work was performed in the spent fuel pool transfer canal. The radiological aspects of the work were covered using Radiation Work Permit 11-3317. Two entries were made into the transfer canal to perform the decontamination. The first entry was to pressure wash the transfer canal and collect debris, and the second entry was to remove the collected debris. After the two entries were completed, the licensee determined that 390 millirem was accrued for Task 1 of Radiation Work Permit 11-3317, out of an originally estimated 115 millirem. The Radiation Protection Manager stopped work on the radiation work permit until an evaluation could be performed. Condition Report 2011-0543 was written and an apparent cause evaluation was completed, and determined that management oversight was the apparent cause and that procedural compliance was a contributing cause. Specific procedure compliance violations that the licensee identified included: not receiving ALARA committee approval prior to work, not logging a stay time into the radiation work permit log, not using the appropriate pre-job brief form or ensuring that all necessary personnel were present at the meeting, and setting electronic alarming dosimeters inappropriately.

The inspectors reviewed the apparent cause evaluation, radiation work permit, and work order and determined that the licensee had not identified all of the performance deficiencies. There were an additional three steps in Procedure RP-301 “ALARA Planning/RWP Development and Control” which were not followed in this evolution. Specifically, the licensee did not prepare an ALARA Planning worksheet as the initial step of generating the radiation work permit, did not document justification for changing the electronic dosimeter set points, which were eventually determined to be inappropriate, and did not perform an ALARA briefing before the entries were made into the spent fuel pool transfer canal, which was posted as a restricted locked high radiation area. The inspectors also determined that there were additional aspects of the procedure that contained vague expectations, which contributed to decisions being made without using the procedure.

Originally, Task 1 was estimated to allow a dose of 115 millirem and the decontamination was to be completed using an underwater robot. The task accumulated 86 millirem; therefore, 100 millirem was added to the task estimate before the two personnel entries, leaving an available 129 millirem in the budget. While reviewing the dose history, the inspectors determined that the licensee added an insufficient amount to the dose estimate before the evolution. Prior to the work, it was estimated that the first entry would use approximately 100 millirem, and the second entry would use approximately 150 millirem. The inspectors questioned why work was planned that would surpass the dose estimate for the task and were told that because the overall radiation work permit included enough estimated dose, an evaluation prior to the work was not performed. To verify that this action followed the licensee's procedural guidance, the inspectors reviewed Procedure RP-301. It states, in part, that a work in progress review should be initiated "when either dose or [radiation work permit] RWP-hour estimates are adjusted or as needed to document relevant job information (i.e., scope changes, delays, problems encountered, job progress updates, etc.)" However, a work in progress review was not completed for this evolution until after the entries had been completed and the dose budget had been exceeded. However, there was no procedural requirement stating that this needed to be filled out prior to work, although the licensee stated that it was an expectation. The inspectors noted the licensee's cause-evaluation addressed procedural compliance, but did not address the quality or clarity of procedural guidance.

The inspectors determined that the additional NRC identified deviations from procedure requirements and expectations contributed to the task going over budget.

Analysis. The failure to follow a procedure was a performance deficiency. The finding was more than minor because it negatively impacted the Occupational Radiation Safety Cornerstone's attribute of program and process, in that by not following the procedure, radiological safety attributes built into the radiation work permit program were circumvented. The absence of the procedure attributes did not ensure that there was adequate protection of worker health from exposure to radiation. Using Inspection Manual Chapter 0609, Appendix C "Occupational Radiation Safety Significance Determination Process," the finding was determined to be of very low safety significance because: (1) it was not associated with ALARA planning or work controls, (2) it was not an overexposure, (3) there was not a substantial potential for overexposure, and (4) the ability to assess dose was not compromised. In addition, this finding had a crosscutting aspect in the area of human performance related to work practices. Specifically, the licensee did not communicate human error prevention techniques, such as holding pre-job briefs, self- and peer-checking, and proper documentation of activities [H.4.a].

Enforcement. Technical Specification 5.8.1.a required that written procedures be established, implemented and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978. Appendix A. Section 7.e covers exposure controls, including access control to radiation areas including a radiation work permit system. Fort Calhoun Procedure RP-301 "ALARA Planning/RWP Development and Control," contained specific instructions for creating and controlling a radiation work permit. Specifically, Step 7.2.3 stated in part "Complete Form FC-1306 (ALARA Planning Worksheet) as the initial step in generating an RWP [radiation work permit]." Additionally, Step 7.5.1 stated in part, "Upon request to change EAD [electronic alarming dosimeter] dose set points, document the change and justification in the applicable RWP logbook." Step 7.5.2 stated, in part, "Upon request to change EAD dose rate set points, document the change and justification in the applicable RWP logbook." Further, Step 7.6.1 stated in part "ALARA briefings are required for work where the possibility exists that a person could receive a dose of greater than or equal to 150 millirem for a single entry." Contrary to the above, specific instructions were bypassed when creating and controlling Radiation Work Permit 11-3317. Specifically, when creating and controlling this radiation work permit in order to perform decontamination of the spent fuel pool transfer canal, a form FC-1306 was not created as the initial step of creating the radiation work permit, justification for electronic alarming dosimeter set points were not documented in the radiation work permit logbook, and an ALARA briefing was not performed when the possibility for greater than or equal to 150 millirem existed. Since this violation was of very low safety significance and was documented in the licensee's corrective action program as Condition Report 2011-3413, it was being treated as a noncited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2011003-03, "Failure to Follow Radiation Work Permit Procedure."

2RS03 In-plant Airborne Radioactivity Control and Mitigation (71124.03)

a. Inspection Scope

This area was inspected to verify in-plant airborne concentrations are being controlled consistent with ALARA principles and the uses of respiratory protection devices on-site do not pose an undue risk to the wearer. The inspectors used the requirements in 10 CFR Part 20, the technical specifications, and the licensee's procedures required by technical specifications as criteria for determining compliance. During the inspection, the inspectors interviewed licensee personnel, performed walkdowns of various portions of the plant, and reviewed the following items:

- The licensee's use, when applicable, of ventilation systems as part of its engineering controls

- The licensee's respiratory protection program for use, storage, maintenance, and quality assurance of NIOSH certified equipment, qualification and training of personnel, and user performance

- The licensee’s capability for refilling and transporting self-contained breathing apparatus air bottles to and from the control room and operations support center during emergency conditions, status of self-contained breathing apparatus staged and ready for use in the plant and associated surveillance records, and personnel qualification and training

- Audits, self-assessments, and corrective action documents related to in-plant airborne radioactivity control and mitigation since the last inspection

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of the one (1) sample as defined in Inspection Procedure 71124.03-05.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

Cornerstones: Initiating Events, Public Radiation Safety, and Occupational Radiation Safety

.1 Data Submission Issue

a. Inspection Scope

The inspectors performed a review of the performance indicator data submitted by the licensee for the first Quarter 2011 performance indicators for any obvious inconsistencies prior to its public release in accordance with Inspection Manual Chapter 0608, “Performance Indicator Program.”

This review was performed as part of the inspectors’ normal plant status activities and, as such, did not constitute a separate inspection sample.

b. Findings

No findings were identified.

.2 Unplanned Scrams per 7000 Critical Hours (IE01)

a. Inspection Scope

The inspectors sampled licensee submittals for the unplanned scrams per the 7000 critical hour’s performance indicator for the period from the second quarter 2010 through the first quarter 2011. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports, and NRC integrated inspection reports for the period of April 1, 2010 through March 31, 2011 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of one (1) unplanned scrams per 7000 critical hours sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.3 Unplanned Scrams with Complications (IE02)

a. Inspection Scope

The inspectors sampled licensee submittals for the unplanned scrams with complications performance indicator for the period from the second quarter 2010 through the first quarter 2011. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports, and NRC integrated inspection reports for the period of April 1, 2010 through March 31, 2011 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of one (1) unplanned scrams with complications sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.4 Unplanned Power Changes per 7000 Critical Hours (IE03)

a. Inspection Scope

The inspectors sampled licensee submittals for the unplanned power changes per 7000 critical hour's performance indicator for the period from the second quarter 2010 through the first quarter 2011. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6. The inspectors reviewed the licensee's operator narrative logs, issue reports, maintenance rule records, event reports, and NRC integrated inspection reports for the period of April 1, 2010 through March 31, 2011 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of one (1) unplanned transients per 7000 critical hours sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.5 Occupational Exposure Control Effectiveness (OR01)

a. Inspection Scope

The inspectors reviewed performance indicator data for the second quarter 2010 through the first quarter 2011. The objective of the inspection was to determine the accuracy and completeness of the performance indicator data reported during these periods. The inspectors used the definitions and clarifying notes contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, as criteria for determining whether the licensee was in compliance.

The inspectors reviewed corrective action program records associated with high radiation area (greater than 1 rem/hr) and very high radiation area non-conformances. The inspectors reviewed radiological, controlled area exit transactions greater than 100 millirem. The inspectors also conducted walkdowns of high radiation areas (greater than 1 rem/hr) and very high radiation area entrances to determine the adequacy of the controls of these areas.

These activities constitute completion of one (1) sample of the occupational exposure control effectiveness as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.6 Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual Radiological Effluent Occurrences (PR01)

a. Inspection Scope

The inspectors reviewed performance indicator data for the second quarter 2010 through the first quarter 2011. The objective of the inspection was to determine the accuracy and completeness of the performance indicator data reported during these periods. The inspectors used the definitions and clarifying notes contained in NEI Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 6, as criteria for determining whether the licensee was in compliance.

The inspectors reviewed the licensee’s corrective action program records and selected individual annual or special reports to identify potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose.

These activities constitute completion of one (1) sample of the radiological effluent technical specifications/offsite dose calculation manual radiological effluent occurrences as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

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

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

As part of the various baseline inspection procedures discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee’s corrective action program at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. The inspectors reviewed attributes that included the complete and accurate identification of the problem; the timely correction, commensurate with the safety significance; the evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent of condition reviews, and previous occurrences reviews; and the classification, prioritization, focus, and timeliness of corrective actions. Minor issues entered into the licensee’s corrective action program because of the inspectors’ observations are included in the attached list of documents reviewed.

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

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. The inspectors accomplished this through review of the station's daily corrective action documents.

The inspectors performed these daily reviews as part of their daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. Findings

No findings were identified.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's corrective action program and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors focused their review on repetitive equipment issues, but also considered the results of daily corrective action item screening discussed in Section 4OA2.2, above, licensee trending efforts, and licensee human performance results. The inspectors nominally considered the 6-month period of January 1, 2011 through June 30, 2011 although some examples expanded beyond those dates where the scope of the trend warranted.

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

These activities constitute completion of one (1) sample of the single semi-annual trend inspection as defined in Inspection Procedure 71152-05.

b. Findings

Introduction. A self-revealing Green noncited violation of Fort Calhoun Technical Specification 5.8.1, “Procedures,” occurred due to the failure of the licensee to ensure that adequate procedures were available for maintenance that was conducted on the reactor protective systems power supplies. Specifically, there was no procedural guidance to require replacement of power supplies, or an engineering justification for continued operation, once power supplies exceeded their vendor recommended life, and/or showed signs of failure and/or degradation.

Description. On February 24, 2010, an 18-volt Dewar power supply for the B Channel Reactor Protection System Axial Power Distribution calculator failed. An apparent cause analysis determined that the power supply failed due to the failure of an internal resistor. During the extent of condition review of the power supply failure, the apparent cause analysis identified that 15 other Dewar power supplies were installed in the reactor protective systems, and all of them were of the same age as the power supply that failed. The apparent cause analysis noted that the power supplies were all scheduled to be replaced, but, there was not sufficient justification for immediate replacement.

On February 20, 2011, a 15-volt Foxboro power supply for the reactor protective systems Channel A reactor coolant system temperature indication failed. An apparent cause analysis determined that the power supply failed due to a failed fuse which resulted in a short in the -15 volt rectifier. The failed fuse and shorted rectifier were determined to be age related. During the extent of condition review of the power supply failure, the apparent cause analysis identified that 20 other Foxboro power supplies were installed in the plant. Nine of the power supplies did not have preventive maintenance tasks to refurbish/replace them on a periodic basis and two of the power supplies had preventive maintenance that had been suspended. The apparent cause analysis further determined there were eleven reactor coolant system power supplies that had no preventive maintenance to refurbish/replace them. All 11 of the power supplies had been installed since issuance of the initial operating license in 1973. Though the 31 power supplies noted in the extent of condition review were all past the vendor recommended lifetime, the licensee failed to provide justification for continued operation.

On February 21, 2011, a 5-volt Lambda power supply for the D reactor coolant system channel variable over power trip failed. As was the case with the previously mentioned failures, this power supply was determined to have failed due to age related degradation. The extent of condition review identified that there were 115 power supplies in the reactor protective system. All but two of the listed power supplies were older than the vendor recommended lifetime, and the licensee had not performed any justification for continued operation.

Following each of the three power supply failures, the licensee evaluated internal and external operating experience. Following the first and second failures, the licensee added internal operating experience that showed a trend of increasing power supply failures due to age-related degradation.

In September 2009, the licensee had made changes to the reactor protective system maintenance rule documents, indicating that components within the reactor program system functional group, including power supplies, were not allowed to run to failure. The licensee scheduled replacement of some, but not all, of the reactor protective systems power supplies in their Equipment Reliability and Optimization Project though the Equipment Reliability and Optimization Project prioritized power supply replacements, it did not include an engineering justification for continued operation past vendor recommended lifetimes. As individual power supplies began failing in 2010, there was no procedural guidance to perform evaluations on power supplies that had exceeded their vendor recommended lifetime.

Analysis. The inspectors determined that the licensee's failure to provide procedural guidance to evaluate and/or replace age-degraded components was a performance deficiency. This was a result of the licensee's failure to properly implement a required procedure, and was within the licensee's ability to foresee and correct and could have been prevented. This performance deficiency was more than minor because it could be reasonably viewed as a precursor to a significant event, i.e., could lead to a loss of the reactor protective system. The inspectors evaluated this finding using Inspection Manual Chapter 0609, Attachment 4, and determined that this finding was associated with the Mitigating Systems Cornerstone, specifically the primary degraded reactivity control contributor. Because this finding occurred while the unit was operating at full power, the inspectors used Inspection Manual Chapter 0609 to determine its significance. The inspectors determined that the finding represented a qualification deficiency confirmed not to result in a loss of functionality because none of the failures to date prevented a reactor protective systems channel from tripping. Therefore, in accordance with the Phase 1 screening, the finding was of very low risk significance (Green).

This finding had a crosscutting aspect in the area of problem identification and resolution associated with the component of operating experience because the licensee failed to adequately evaluate and communicate relevant internal and external operating experience [P.2(a)].

Enforcement. Fort Calhoun Station Technical Specification 5.8.1 requires, in part, that the licensee establish and implement written procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Specifically, Regulatory Guide 1.33 requires procedures for performing maintenance and inspection or replacement of parts that have a specified lifetime. Contrary to the above, the licensee failed to provide procedural guidance to evaluate or replace power supplies known to be significantly past their vendor recommended lifetime. Because the violation was of very low safety significance and was entered into the licensee's corrective action program as Condition Report 2011-1655, this violation was being treated as a noncited violation consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2011003-04, “Failure to Provide Procedural Guidance to Replace or Evaluate Age Degraded Components.”

4OA3 Event Follow-up (71153)

.1 (Open) Licensee Event Report 05000285/2011-001-00: Inadequate Flooding Protection Due To Ineffective Oversight

During identification and evaluation of flood barriers, unsealed through wall conduit penetrations in the outside wall of the intake structure were identified that are below the licensing basis flood elevation.

A summary of the root causes included: a weak procedure revision process; insufficient oversight of work activities associated with external flood matters; ineffective identification, evaluation and resolution of performance deficiencies related to external flooding; and "safe as is" mindsets relative to external flooding events.

The penetrations were temporarily sealed and a configuration change was developed and implemented whereby permanent seals were installed. Comprehensive corrective actions to address the root and contributing causes were being addressed through the corrective action program.

.2 (Open and Closed) Licensee Event Report 05000285/2010-005-01: Inoperability of the Emergency Diesel Generator Fuel Oil Transfer System

On November 29, 2010, during the performance of a work order, voltage at reactor protective system connection T-74 was found 39 millivolt (mV) higher than connection T-17 (reactor protective system ground). The allowed limit is 4 millivolt. Connection T-74 is the signal common lead for steam generator pressure channels 902 and 905 inputs to trip unit 6 (low steam generator pressure) and trip unit 7 (asymmetric steam generator transient). Further investigation determined that the affected channels should have been declared inoperable. With a channel of the reactor protective system inoperable, the appropriate section of technical specifications should have been entered. The technical specification limiting condition for operation action times were not met.

This licensee event report was reviewed by the inspectors. A Green noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI was issued regarding this condition (05000285/2011002-02). This licensee event report was closed.

.3 (Closed) Licensee Event Report 05000285/2011-003-00: Inadequate Flooding Protection due to Inadequate Oversight

During identification and evaluation of flood barriers, unsealed through wall penetrations in the outside wall of the intake, auxiliary and chemistry and radiation protection buildings were identified that were below the licensing basis flood elevation.

The penetrations were temporarily sealed and a configuration change was developed and implemented whereby permanent seals were installed. Comprehensive corrective actions to address the root and contributing causes were being addressed through the corrective action program.

This licensee event report was closed. The inspectors will review the condition described in revision 1 of this licensee event report.

.4 (Open) Licensee Event Report 05000285/2011-003-01: Inadequate Flooding Protection due to Inadequate Oversight

During identification and evaluation of flood barriers, unsealed through wall penetrations in the outside wall of the intake, auxiliary and chemistry and radiation protection buildings were identified that were below the licensing basis flood elevation.

A summary of the root causes included: a weak procedure revision process; insufficient oversight of work activities associated with external flood matters; ineffective identification, evaluation and resolution of performance deficiencies related to external flooding; and “safe as is” mindsets relative to external flooding events.

The penetrations were temporarily sealed and a configuration change was developed and implemented whereby permanent seals were installed. Comprehensive corrective actions to address the root and contributing causes are being addressed through the corrective action program.

.5 (Open and Closed) Licensee Event Report 05000285/2011-004-00: Isolation of Both Trains of Safety Related Auxiliary Feedwater

On February 5, 2011, during plant startup activities, operations personnel initiated a transition from auxiliary feedwater to main feedwater while in Mode 2 (hot standby condition). During the transition, auxiliary feedwater was being supplied by a safety-related motor-driven auxiliary feedwater pump (FW-6) through the auxiliary feedwater nozzles (HCV-1107A/B and HCV 1108A/B). With the main feedwater aligned and feeding both steam generators, the control room operator was directed to shut down FW-6 and return the system to its normal alignment. During this activity the control room operator placed both inboard isolation valves, as directed by procedure, HCV-1107A and HCV-1108A, into their closed position. This action defeated automatic initiation via an auxiliary feedwater actuation signal to open the valves, rendering both trains of auxiliary feedwater inoperable.

The condition was recognized and the control switches were placed in "Auto" restoring both trains to operable.

This licensee event report was reviewed by the inspectors. A Green noncited violation of Fort Calhoun Technical Specification 5.8.1 was issued regarding this condition (05000285/2011002-01). This licensee event report was closed.

.6 (Open) Licensee Event Report 05000285/2011-005-00: Failure to Correctly Enter Technical Specifications Limiting Condition for Operation for the Reactor Protective System

On June 14, 2010, the reactor protective system M2 contactor (similar to the reactor protective system breakers) failed to open during periodic surveillance testing. Operations declared the reactor protective system M2 contactor inoperable and entered Technical Specification Limiting Condition for Operation 2.15(1) because the reactor protective system M2 contactor did not have a specifically defined limiting condition for operation. Subsequent reviews determined that the station continued to operate in a condition not allowed by technical specification on June 14, 2010, and June 15, 2010, for a period of approximately 20.5 hours. Technical Specification 2.0.1 should have been invoked. (Section 2.0.1, similar to Standard Technical Specification 3.0.3.)

The Licensee determined the root cause for this error was the failure to implement an interim technical specification strategy when funding for standard improved technical specification was deferred.

The operations staff had been directed to enter Technical Specification 2.0.1 for any failures of these contactors. Fort Calhoun Station will conduct a formal review of other components which do not have specific technical specification limiting condition for operation action statements and station actions that could be non-conservative with regard to entering Technical Specification 2.0.1. The review will identify those items that need administrative controls and place them in the appropriate station procedures.

.7 <u>(Open) Licensee Event Report 05000285/2011-006-00: Inoperability of Both Trains of Containment Coolers due to a Mispositioned Valve</u>

On March 22, 2011, during the performance of a test on containment cooler valves, a technician discovered that NGHCV-400A-A3, “CCW Inlet Valve HCV-400A Nitrogen Supply Isolation Valve,” was in the closed position. This was not the correct position. He informed the control room of the condition. At the time of discovery, containment cooler VA-3B was inoperable and unable to support the performance of a surveillance test. Operations declared VA-3A inoperable as the backup nitrogen supply to HCV-400A for containment cooler VA-3A cooling coil was unavailable. Operations entered Technical Specification 2.0.1 (similar to Standard Technical Specification 3.0.3) since both VA-3A and VA-3B were simultaneously inoperable. An equipment operator was dispatched to open NG-HCV-400A-A3. After NG-HCV-400A-A3 was opened, VA-3A was declared operable. Technical Specification 2.0.1 was then exited.

The licensee determined the root cause of this event was the station’s leadership oversight effort has not been effective in the areas of use of the station’s corrective action program, human performance tools, and safe work practices in reducing the potential for mispositioning events.

The immediate corrective action of opening the affected valve restored VA-3A to an operable condition. Additional corrective actions to address the root and generic implications of this event will be addressed by the station’s corrective action process.

.8 <u>(Open) Licensee Event Report 05000285/2011-007-00: Violation of Technical Specifications due to Reactor Coolant System Boundary Leakage</u>

On April 12, 2011, during the performance of an inspection of reactor coolant pump, RC-3C, and its studs, a small boric acid leak was discovered. The leak was on a ¾-inch nominal diameter stainless steel pipe welded to the pump upstream of isolation valve RC-270 (RC-3C, reactor coolant pump casing gasket leak detection pressure indication alarm PIA-3195 root valve). The pipe connects the area between the inner and outer gaskets on the reactor coolant pump casing to a pressure indicator alarm. The pipe had a through-wall crack. On May 30, 2011, it was determined that the failure of the pipe violated technical specifications for zero pressure boundary leakage.

The licensee determined the root cause of the crack was transgranular stress corrosion cracking caused by a post-manufacturing bend.

The affected pipe was replaced. Equivalent indicator piping for the other three reactor coolant pumps was inspected and one of those lines was also replaced due to an unacceptable bend in the pipe. The line was evaluated for transgranular stress corrosion cracking and no indications were found. Associated instrument lines for the reactor coolant pumps were evaluated and no susceptibility to transgranular stress corrosion cracking was identified.

4OA5 Other Activities

.1 <u>(Open) NRC TI 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal and Containment Spray Systems (NRC Generic Letter 2008-01)"</u>

As documented in Section 1R22, the inspectors confirmed the acceptability of the described licensee actions. Surveillance Testing Associated with Temporary Instruction 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems"

a. <u>Inspection Scope</u>. The inspectors reviewed procedures used for conducting surveillances and determination of void volumes to ensure that the void criteria was satisfied and would be reasonably assured to satisfactory until the next scheduled void surveillance (TI 2515/177, Section 04.03.a). The inspectors also reviewed procedures used for filling and venting the following conditions, which may have introduced voids into the subject systems to verify that the procedures acceptably addressed testing for such voids and provided acceptable processes for their reduction or elimination (TI 2515/177, Section 04.03.b). Specifically, inspectors verified that:

Gas intrusion prevention, refill, venting, monitoring, trending, evaluation, and void correction activities were acceptably controlled by approved operating procedures (TI 2515/177, Section 04.03.c.1).

- Procedures ensured that systems did not contain voids that may jeopardize operability (TI 2515/177, Section 04.03.c.2).

- Procedures established that void criteria were satisfied and would be reasonably ensured to be satisfied until the next scheduled void surveillance (TI 2515/177, Section 04.03.c.3).

- The licensee entered changes into the corrective action program as needed to ensure acceptable response to issues. In addition, the inspectors confirmed that a clear schedule for completion was included for corrective action program entries that had not been completed (TI 2515/177, Section 04.03.c.5).

- Procedures included independent verification that critical steps were completed (TI 2515/177, Section 04.03.c.6).

The inspectors verified the following with respect to surveillance and void detection:

- Specified surveillance frequencies were consistent with technical specification surveillance requirements (TI 2515/177, Section 04.03.d.1).

- Surveillance frequencies were stated or, when conducted more often than required by technical specifications, the process for their determination was described (TI 2515/177, Section 04.03.d.2).

- Surveillance methods were acceptably established to achieve the needed accuracy (TI 2515/177, Section 04.03.d.3).

- Surveillance procedures included up-to-date acceptance criteria (TI 2515/177, Section 04.03.d.4).

- Procedures included effective follow-up actions when acceptance criteria are exceeded or when trending indicates that criteria may be approached before the next scheduled surveillance (TI 2515/177, Section 04.03.d.5).

- Venting procedures and practices utilized criteria such as adequate venting durations and observing a steady stream of water (TI 2515/177, Section 04.03.d.7).

- An effective sequencing of void removal steps was followed to ensure that gas does not move into previously filled system volumes (TI 2515/177, Section 04.03.d.8).

- Venting results were trended periodically to confirm that the systems are sufficiently full of water and that the venting frequencies are adequate. The inspectors also verified that records on the quantity of gas at each location are maintained and trended as a means of preemptively identifying degrading gas accumulations (TI 2515/177, Section 04.03.d.10).

- Surveillances were conducted at any location where a void may form, including high points, dead legs, and locations under closed valves in vertical pipes (TI 2515/177, Section 04.03.d.11).

- The licensee ensured that systems were not pre-conditioned by other procedures that may cause a system to be filled, such as by testing, prior to the void surveillance (TI 2515/177, Section 04.03.d.12).

The inspectors verified the following with respect to void control:

- Void removal methods were acceptably addressed by approved procedures (TI 2515/177, Section 04.03.f.1).

Documents reviewed are listed in the attachment to this report.

This inspection effort counts towards the completion of TI 2515/177 which will be closed in a later inspection report.

b. Findings

No findings were identified.

2 (Closed) NRC Temporary Instruction 2515/183, "Follow-up to the Fukushima Daiichi Nuclear Station Fuel Damage Event"

a. Inspection Scope

The inspectors assessed the activities and actions taken by the licensee to assess its readiness to respond to an event similar to the Fukushima Daiichi nuclear plant fuel damage event. This included (1) an assessment of the licensee's capability to mitigate conditions that may result from beyond design basis events, with a particular emphasis on strategies related to the spent fuel pool, as required by NRC Security Order Section B.5.b issued February 25, 2002, as committed to in severe accident management guidelines, and as required by 10 CFR 50.54(hh); (2) an assessment of the licensee's capability to mitigate station blackout conditions, as required by 10 CFR 50.63 and station design bases; (3) an assessment of the licensee's capability to mitigate internal and external flooding events, as required by station design bases; and (4) an assessment of the thoroughness of the walkdowns and inspections of important equipment needed to mitigate fire and flood events, which were performed by the licensee to identify any potential loss of function of this equipment during seismic events possible for the site.

b. Findings

Inspection Report 05000285/2011010 (ML11133A339) documented detailed results of this inspection activity. Following issuance of the report, the inspectors conducted detailed follow-up on selected issues. No findings were identified during this follow-up inspection.

.3 <u>(Closed) NRC Temporary Instruction 2515/184, "Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs)"</u>

The inspectors reviewed the licensee's severe accident management guidelines (SAMGs), implemented as a voluntary industry initiative in the 1990's, to determine (1) whether the SAMGs were available and updated, (2) whether the licensee had procedures and processes in place to control and update its SAMGs, (3) the nature and extent of the licensee's training of personnel on the use of SAMGs, and (4) licensee personnel's familiarity with SAMG implementation.

The results of this review were provided to the NRC task force chartered by the Executive Director for Operations to conduct a near-term evaluation of the need for agency actions following the Fukushima Daiichi fuel damage event in Japan. Plant-specific results for Fort Calhoun Station were provided as Enclosure 1 to a memorandum to the Chief, Reactor Inspection Branch, Division of Inspection and Regional Support, dated May 27, 2011 (ML111470264).

4OA6 Meetings

<u>Exit Meeting Summary</u>

On April 22, 2011, the inspectors presented the results of the radiation safety inspections to Mr. J. Reinhart, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On April 27, 2011, the inspectors discussed the results of in-office inspection of changes to the licensee's emergency plan with Mr. A. Berck, Supervisor, Emergency Planning, and other members of the licensee's staff. The licensee acknowledged the issues presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified

On April 27, 2011, the inspectors presented the inspection results of the in-service inspection activities to Mr. J. Reinhart, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On August 9, 2011, the inspectors presented the inspection results of the inspection to Mr. T. Nellenbach, Plant Manager, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

S. Andersen, Supervisor Engineering Programs
M. Anderson, Supervisor, RAD Waste
S. Baughn, Manager Nuclear Licensing
A. Berck, Supervisor, Emergency Planning
P. Downey, In-service Inspection Program Engineer
D. Gautreau, Reactor Operator
J. Goodell, Division Manager
J. Grewe, Welding Engineer
D. Guinn, Supervisor Regulatory Compliance
J. Herman, Division Manager Nuclear Engineering
R. Hodgson, Manager, Radiation Protection
T. Jamieson, Supervisor, RAD Operations
B. Lisowyi, Project Manager
A. Lollis, Supervisor, ALARA
C. Longua, Shift Manager
E. Matzke, Senior Nuclear Licensing Engineer, Regulatory Compliance
T. Nellenbach, Plant Manager
J. Reinhart, Site Vice President
B. Schawe, Control Room Supervisor
J. Shryock, Reactor Operator
J. Willett, Reactor Engineer

NRC Personnel

J. Dixon, Senior Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

2515/177	TI	Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal and Containment Spray Systems (NRC Generic Letter 2008-01 (Section 4OA5)
05000285/2011-001-00	LER	Inadequate Flooding Protection Due To Ineffective Oversight (Section 4OA3)

Opened

05000285/2011-003-01	LER	Regarding Inadequate Flooding Protection due to Inadequate Oversight (Section 4OA3)
05000285/2011-005-00	LER	Failure to Correctly Enter Technical Specifications Limiting Condition for Operation for the Reactor Protective System (Section 4OA3)
05000285/2011-006-00	LER	Inoperability of Both Trains of Containment Coolers due to a Mispositioned Valve (Section 4OA3)
05000285/2011-007-00	LER	Violation of Technical Specifications due to Reactor Coolant System Boundary Leakage (Section 4OA3)

Opened and Closed

05000285/2010-005-01	LER	Inoperability of the Emergency Diesel Generator Fuel Oil Transfer System (Section 4OA3)
05000285/2011-003-00	LER	Regarding Inadequate Flooding Protection due to Inadequate Oversight (Section 4OA3)
05000285/2011-004-00	LER	Isolation of Both Trains of Safety Related Auxiliary Feedwater (Section 4OA3)
05000285/2011003-01	NCV	Failure to Adequately Design a Reactant Coolant Pump Lube Oil Collection System (Section 1R05)
05000285/2011003-02	NCV	Failure to Follow Scaffolding Procedure (Section 1R15)
05000285/2011003-03	NCV	Failure to Follow Radiation Work Permit Procedure (Section 2RS02)
05000285/2011003-04	NCV	Failure to Provide Procedural Guidance to Replace or Evaluate Age Degraded Components (Section 4OA2)

Closed

2515/183	TI	Follow-up to the Fukushima Daiichi Nuclear Station Fuel Damage Event (Section 4OA5)
2515/184	TI	Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs) (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

PROCEDURES

NUMBER	TITLE	REVISION
AOP-01	Acts of Nature	27
EPIP-TSC-2	Catastrophic Flooding Preparations	12
PE-RR-AE-1000	Floodgate Inspection and Repair	9
PE-RR-AE-1001	Flood Barrier and Sandbag Staging and Installation	8

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION / DATE
	Fort Calhoun Technical Specification 2.16, River Level	0
	Safety Evaluation Report of the Omaha Public Power District Fort Calhoun Station Unit No. 1, Supplement 1	April 23, 1973
USAR 2.7	Hydrology	11
USAR 9.8	Raw Water System	28

Section 1R04: Equipment Alignment

PROCEDURES

NUMBER	TITLE	REVISION
OI-RC-1A	Operating Instruction - RCS Instrumentation	20
OI-SC-1	Shutdown Cooling Initiation	52
OI-SC-5	Shutdown Cooling Purification	29

CONDITION REPORTS

2011-2890

DRAWINGS

NUMBER	TITLE	REVISION
11405-M-10	Auxiliary Coolant Component Cooling System	30

DRAWINGS

NUMBER	TITLE	REVISION
11405-M-100	Raw Water Flow Diagram	99
D-4768	Primary Plant Simplified Flow Path Diagram	6
E-23866-210-130	Safety Injection and Containment Spray System Flow Diagram	111

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION
USAR 4.0	Reactor Coolant System	4
USAR 7.4	Regulating Systems	6
USAR 9.3	Shutdown Cooling	12
USAR 14.3	Safety Analysis Boron Dilution Incident	18

Section 1R05: Fire Protection

CONDITION REPORTS

2011-4584 2011-5992

WORK ORDERS

164969

DRAWINGS

NUMBER	TITLE	REVISION
1928-M-015	Assembly - Oil Collection System for RC Pump & Motor 3A	3
1928-M-019	Oil Collection System View 1-1	2
1928-M-025	Exploded Isometric of Oil Collection System for Motors 3A & 3C	3
1928-SK-010	Oil Collection Drain Arrangement	1
D-4094, Sheet 1	Fire Detection System Ground Floor Plan	8
D-4094, Sheet 2	Fire Detection System Basement Floor Plan 995’-6”	5
D-4094, Sheet 3	Fire Detection System Aux. Bldg. & Containment Elevation 1025’-0”	6

DRAWINGS

NUMBER	TITLE	REVISION
D-4094, Sheet 4	Fire Detection System Operating Floor Plan Elev. 1036’-0”	13
D-4094, Sheet 7	Fire Detection System Turbine Floor Plan Elev. 1036’-0”	3
D-4094, Sheet 8	Fire Protection System in the Technical Support Center & Intake Structure	6
D-4147, Sheet 1	Containment & Auxiliary Building Elevation 1036’ Portable Fire Extinguisher Locations	8
D-4147, Sheet 2	Ground Floor Plan Elevation 1023’-0” Portable Fire Extinguisher Locations	6
D-4147, Sheet 3	Ground Floor Plan Elevation 1007’-0” Portable Fire Extinguisher Locations	11
D-4147, Sheet 4	Ground Floor Plan Elevation 989’-0” Portable Fire Extinguisher Locations	5
D-4770, Sheet 4	RC-3A, C, D Oil Collection Plan Views and Sections at Pan Types 1, 2 & 3	1

PROCEDURES

NUMBER	TITLE	REVISION
SO-G-103	Standing Order, Fire Protection Operability Criteria and Surveillance Requirements	25
SO-G-102	Standing Order, Fire Protection Program Plan	9
SO-G-91	Standing Order, Control and Transportation of Combustible Materials	26
SO-G-58	Standing Order, Control of Fire Protection System Impairments	37
SO-G-28	Standing Order, Station Fire Plan	79

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION
EA-FC-97-001	Fire hazards Analysis Manual	15
FC05814	UFHA Combustible Loading Calculation	11

USAR 9.11	Updated Safety Analysis Report, Fire Protection Systems	21

Section 1R06: Flood Protection Measures

PROCEDURES

NUMBER	TITLE	REVISION
AOP-10	Loss of Circulating Water	2
AOP-11	Loss of Component Cooling Water	15
AOP-18	Loss of Raw Water	7

DRAWINGS

NUMBER	TITLE	REVISION
E-23866-210-120, Sht. Cov.	Composite Flow Diagram Chemical & Volume Control System P & ID	48

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	DATE
FSAR, Appendix M	Postulated High Energy Line Rupture Outside the Containment	June, 1973
	Individual Plant Examination Submittal	December, 1993

Section 1R08: In-service Inspection Activities

PROCEDURES

NUMBER	TITLE	REVISION
FCSG-15-31	Fall Protection Requirements	8
FCSG-24	Corrective Action Program Guidelines	31
FCSG-8	Procedure Format and Content	26
NOD-QP-37	Performance Indicator Program	24
OPPD-PT-98-1	Liquid Penetrant Examination – Solvent Removable, Visible Dye Technique	4
OPPD-UT-98-2	Manual Ultrasonic Examination of Austenitic Piping Welds	3

PROCEDURES

NUMBER	TITLE	REVISION
OPPD-VT-98-3	Visual Examination for Mechanical and Structural Condition of Components and Their Supports	2
PBD-10	Boric Acid Corrosion Control Program	14
QCP-281	Calibration of Ultrasonic Testing Units	4
QCP-310	Liquid Penetrant Examination (Solvent Removable)	18
QCP-400	Visual Inspection	4
SE-EQT-MX-002	Carbon Steel and Low Alloy Steel Fasteners Inservice Testing Inspections	12
SE-ST-CONT-0001	Containment General Structural Inspection	14
SO-G-7	Procedure Use and Adherence	67
SPP-10	Filler Metal Control	16
SPP-10	Filler Metal Control	16
WDI-PJF-1305131-FSR-001	CEDM Seal Housing Eddy Current Examination	1
WDI-STD-1003	CRDM and/or CEDM Seal Housing Eddy Current Examination	0

MISCELLANEOUS

NUMBER	TITLE	DATE
	Boric Acid Program Performance Indicator	April 2010 – March 2011
	Boric Acid Corrosion Control Program Health Report 4th Quarter 2010	
08-QUA-059	Quality Department Surveillance Report – Station Engineering	August 21, 2008
2009-1228	Boric Acid Corrosion Control Self-Assessment	March 2, 2011
QCIR 20090333	Boric Acid Inspection	November 15, 2009
QCIR 20090381	Boric Acid on Studs	November 15, 2009
QCIR 20110095	Perform Bare Metal Visual Inspection	April 22, 2011
QCIR 20110125	Inspect for Source of B.A. Leakage	April 4, 2011

CONDITION REPORTS

20091908	20092111	20092279	20095434	20095806	20101726
20104789	20105273	20105453	20105454	20111595	20113065
20113321	20113372	20113400	20113423	20113437	20113440
20113442	20113443	20113445	20113453	20113458	20113592
20113599	20113625	20113764	20113764		

WORK ORDERS

00318770 00318775

Section 1R12: Maintenance Effectiveness

PROCEDURES

NUMBER	TITLE	REVISION
SO-O-21	Conduct of Operations	85

MISCELLANEOUS DOCUMENTS

TITLE	DATE
Simulator package	June 14, 2011
Open Simulator Discrepancy Reports (All)	
Current Simulator Differences List	

Simulator Modification Procedures
Verification and Validation Procedures
Current operator license list from Fort Calhoun Station

Section 1R13: Maintenance Risk Assessment and Emergent Work Controls

CONDITION REPORTS

2011-4771

PROCEDURES

NUMBER	TITLE	REVISION / DATE
ANSI N18.7	Administrative Controls for Nuclear Power Plants	1972
SO-M-100	Standing Order, Conduct of Maintenance	53
SO-M-101	Standing Order, Maintenance Work Control	86
SO-O-21	Shutdown Operations Protection Plan	39
OI-RC-2A	RCS Fill and Drain Operations	72

MISCELLANEOUS DOCUMENTS

TITLE	DATE
Shift Outage Manager's Report	April 21, 2011
Shift Outage Manager's Report	May 20, 2011

Section 1R15: Operability Evaluations

CONDITION REPORTS

1999-1833	2011-1622	2011-2399	2011-2400	2011-2425
2011-2480	2011-2522	2011-2866	2011-4711	2011-4894
2011-5826				

PROCEDURES

NUMBER	TITLE	REVISION
ARP-CB-1,2,3/A6	Annunciator Response Procedure	42
PED-CSS-12	Standard Specification for Scaffold Construction	6
FC1145	Scaffold Request Form	16
SO-M-35	Scaffolding Installation Control	19

DRAWINGS

NUMBER	TITLE	REVISION
11405-M-10	Auxiliary Coolant Component Cooling System	30

11405-M-253	Steam Generator Feedwater and Blowdown	48

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	DATE
FC06700	NPSH for single operating CCW pump	April 16, 1998
FC05464	Sensitivity of Containment Sump Level and Dew Point Temperature	May 14, 1990
NOD-QP-31.2	Functionality Evaluation Form	April 13, 2011
	Root Cause Analysis Report (2011-2400)	May 19, 2011
NOD-QP-31.1	Operability Evaluation Form (2011-2425)	April 5, 2011

Section 1R18: Plant Modifications

DRAWINGS

NUMBER	TITLE	REVISION
MC0274, Sh 1	Motor Control Resistor	1
R74678, Sh 2	Schematic Diagram 4-Motion Static Steepless	0
B-4471	HE-1 Resistor Seismic Supports	0

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	DATE
EC 52306	Polar Crane (HE-1) Main Hoist Resistor Temp Mod	April 14, 2011

Section 1R19: Postmaintenance Testing

WORK ORDERS

316915	370511	396095	412235

PROCEDURES

NUMBER	TITLE	REVISION
IC-RR-IX-0300	Mounting Instructions for Conoflow E/P Transducer and Regulator	4
IC-PM-IA-0401	HCV-344 & HCV-345 Backup Nitrogen Supply Functional Test	2
IC-CP-01-0344	Calibration of Containment Spray Header Isolation Valve, Loop H-344	7

Section 1R20: Refueling and Other Outage Activities

CONDITION REPORTS

199700464 200303706

PROCEDURES

NUMBER	TITLE	REVISION
OI-RC-2A	RCS Fill and Drain Operations	72
AOP-19	Loss of Shutdown Cooling	17
AOP-22	Reactor Coolant Leak	32
AOP-32	Loss of 4160 Volt or 480 Volt Bus Power	16
OP-3A	Plant Shutdown	82

Section 1R22: Surveillance Testing

PROCEDURES

NUMBER	TITLE	REVISION
IC-ST-AE-3833	Type C Local Leak Rate Test of Penetration M-HCV-383-3	22
OI-CC-1	Component Cooling System Normal Operation	68
OI-CC-1	Component Cooling System Normal Operation	68
OI-CS-1	Containment Spray – Normal Operation	38
OI-CS-1	Containment Spray – Normal Operation	38
OI-SC-1	Shutdown Cooling Initiation	52
OI-SC-1	Shutdown Cooling Initiation	52
OI-SI-1	Safety Injection – Normal Operation	129
OI-SI-1	Safety Injection – Normal Operation	129
OP-ST-CCW-3022	AC-3C Component Cooling Water Pump Inservice Test	17
OP-ST-VX-3019	Safety Injection System Remote Position Indicator Verification Surveillance Test	16

PROCEDURES

NUMBER	TITLE	REVISION
PBD-32	Managing Gas Accumulation in Safety Systems	3
PBD-32	Managing Gas Accumulation in Safety Systems	3
QC-ST-ECCS-0001	Quarterly ECCS Gas Accumulation Detection	9
QC-ST-ECCS-0001	Quarterly ECCS Gas Accumulation Detection	9
QC-ST-ECCS-0002	Refueling ECCS Gas Accumulation Detection	3
QC-ST-ECCS-0002	Refueling ECCS Gas Accumulation Detection	3

CONDITION REPORTS

2010-1450

MISCELLANEOUS

TITLE	DATE
Gas Voiding ST Trend Data	February 24. 2011

WORK ORDERS

394483

Section 2RS01: Radiological Hazard Assessment and Exposure Controls

PROCEDURES

NUMBER	TITLE	REVISION
RP-202	Radiological Surveys	38
RP-204	Radiological Area Controls	58
RP-670	Declared Pregnancy/ Anticipated Pregnancy Procedure	0
RP-AD-200	Radiation Protection Surveillance Program	34
RP-AD-600	Dosimetry Program	21
RPP	Radiation Protection Program	25
RP-ST-RM-0002	Surveillance Test Radioactive Material Sources Surveillance	8
SO-G-101	Standing Order Radiation Worker Practices	34

AUDITS, SELF-ASSESSMENTS, AND SURVEILLANCES

NUMBER	TITLE	DATE
58	Radiation Protection	June 2, 2010

CONDITION REPORTS

2010-2843	2010-3131	2010-3133	2010-3260	2010-3733
2010-5418	2010-6846	2011-0340	2011-0520	2011-0543
2011-0545	2011-0573	2011-0596	2011-0818	2011-1065
2011-1230	2011-1626	2011-2029	2011-2215	2011-3327
2011-3335	2011-3337	2011-3346	2011-3347	2011-3355
2011-3356	2011-3367	2011-3369		

Section 2RS02: Occupational ALARA Planning and Controls

PROCEDURES

NUMBER	TITLE	REVISION
FCSG-32	Work Week Management	23
RP-202	Radiological Surveys	38
RP-204	Radiological Area Controls	58
RP-301	ALARA Planning/RWP Development and Control	41
RP-670	Declared Pregnancy/ Anticipated Pregnancy Procedure	0
RP-AD-600	Dosimetry Program	21
RP-AS-300	ALARA Program	23

AUDITS, SELF-ASSESSMENTS, AND SURVEILLANCES

NUMBER	TITLE	DATE
58	Radiation Protection	June 2, 2010

CONDITION REPORTS

2010-2843	2010-3260	2010-4261	2010-4264	2010-4851
2010-5143	2010-6846	2011-0340	2011-0520	2011-0521
2011-0524	2011-0543	2011-0545	2011-0546	2011-0573
2011-0596	2011-0818	2011-1084	2011-1748	2011-1789
2011-2067	2011-3156	2011-3327	2011-3335	2011-3337
2011-3338	2011-3346	2011-3347	2011-3355	2011-3356
2011-6465				

SURVEYS

NUMBER	TITLE	DATE
10-0454	Containment Elevation 1045: Troubleshoot FE-142	May 28, 2010
10-0456	Containment Elevation 1045: Remove old FI-42 cable & Measure	May 31, 2010
10-0461	Containment Elevation 1045: Replace Flow Indicator Soft cable for FI-42	June 2, 2010
10-0466	Charging Pump CH-1A/1B/1C Repack CH-1B	June 3, 2010
10-0643	Charging Pump CH-1A/1B/1C Repack and Baffle Seal	August 5, 2010
10-0681	Containment Elevation 1013 Sample SITs/ vent LPSI Header RCS Leakage Inspection	August 26, 2010
10-0682	Containment Elevation 994: Investigate Leak Rate	August 26, 2010
11-0052	Spent Fuel Pool Canal Fuel Transfer Machine Post Drain Down Transfer Canal	January 20, 2011
11-0070	Spent Fuel Pool Decon Transfer Canal	January 24, 2011
11-0091	Spent Fuel Pool RHRA Bucket Debris Removal	February 2, 2011

Section 2RS03: In-plant Airborne Radioactivity Control and Mitigation

PROCEDRES

NUMBER	TITLE	REVISION
RP-502	Use of Respiratory Protection Equipment	18
RP-505	Issue and Control of Respiratory Protection Equipment	15
RP-509	Radiation Protection Respirator Fit Testing	21
RP-513	Radiation Protection Baron II SCBA Fill System	13
RP-AD-500	Respiratory Protection Program	19

CONDITION REPORTS

2010-3113	2010-3173	2010-3817	2010-3858	2010-4188
2010-4261	2010-4264	2010-4269	2010-4403	2010-4763
2010-4766	2010-4851	2010-4934	2010-5143	2010-5562
2010-5608	2010-5715	2010-6864	2010-6864	2010-6864
2011-0182	2011-0524	2011-1084	2011-1748	2011-1789
2011-1844	2011-2013			

Section 4OA1: Performance Indicator Verification

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION / DATE
	Various Operator Logs	April 1, 2010 to March 31, 2011
NEI 99-02	Regulatory Assessment Indicator Guideline	6

Section 4OA2: Identification and Resolution of Problems

CONDITION REPORTS

2010-0502	2010-0977	2010-5390	2011-1278	2011-1304
2011-1655	2009-2537	2009-1770	2011-1671	2011-2866
200502139	199901833	2011-5826		

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	DATE
FC-OPS-084-99	Memorandum	October 22, 1999
LIC-09-0052	Licensee Event Report 2009-002	August 3, 2009
LIC-93-0074	OPPD to NRC letter regarding Application for Amendment of Operating License	February 12, 1993
M85848	NRC to OPPD letter regarding Amendment No. 165 to technical specification	August 25, 1994
MC0194	Amendment No. 226 issuance	May 7, 2004

Section 4OA5: Other Activities

CONDITION REPORTS

2011-0609	2011-2072	2011-2078	2011-2110	2011-2132
2011-2150	2011-2164	2011-2165	2011-2305	2011-2324
2011-2331	2011-2336	2011-2338	2011-2348	2011-2352
2011-2355	2011-2380	2011-2386	2011-2448	2011-2451
2011-2470	2011-2471	2011-2520	2011-2531	2011-2532
2011-2562				

SAMG-ABBREV	Abbreviation	0
SAMG-ASSINST	Assessment of Instrumentation and Equipment for Severe Accident Management	0
SAMG-BD-B	Core Badly Damaged and Containment Bypassed (BD/B)	5
SAMG-BD-CC	Core Badly Damaged and Containment Closed and Cooled (BD/CC)	5
SAMG-BD-CH	Core Badly Damaged and Containment Challenged (BD/CH)	5
SAMG-BD-I	Core Badly Damaged and Containment Impaired (BD/I)	5
SAMG-CALCAID	Calculation Aids	3
SAMG-EX-B	Ex-Vessel and containment Bypassed (EX/B)	5
SAMG-EX-CC	Ex-Vessel and Containment Closed and Cooled (EX/CC)	5
SAMG-EX-CH	Ex-Vessel and Containment Challenged (EX/CH)	5
SAMG-EX-I	Ex-Vessel and containment Impaired (EX/I)	5
SAMG-GLOSSARY	Glossary	0
SAMG-INTRO	Introduction	0
SAMG-PHASE 1	Initial Diagnosis	1
SAMG-PHASE 2	Verification of Diagnosis	0
SAMG-RESTOR	Restoration	3
SAMG-RESTORATT	Restoration Attachments	2
OCAG-1	Operational Contingency Action Guideline	13
EOP-07	Station Blackout	14
EOP-20	Functional Recovery Procedure	24
EOP/AOP	EOP/AOP Attachments	29

Attachments

USAR-2.7	Hydrology	11
PE-RR-AE-1001	Flood Barrier and Sandbag Staging and Installation	8
AOP-01	Acts of Nature	25
AOP-06	Fire Emergency	24