UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 EAST LAMAR BLVD
ARLINGTON, TEXAS 76011-4511

April 24, 2013

Mr. Louis Cortopassi, Site Vice President
Omaha Public Power District
Fort Calhoun Station FC-2-4
P.O. Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: FORT CALHOUN STATION INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) INSPECTION REPORT 05000285/2013007 AND 07200054/2013001

Dear Mr. Cortopassi:

A routine site inspection was completed of your dry cask storage activities associated with your Independent Spent Fuel Storage Installation (ISFSI) on January 14 - 17, 2013. A briefing of the status of findings for the site visit was provided January 17, 2013 to your staff. Following the site visit, an in-house review was conducted of your auxiliary building crane up-rate project which increased the crane's rated capacity from 75 tons to 106 tons. A telephonic exit was conducted on April 17, 2013 at the conclusion of the in-office review. The focus of this inspection was to verify ongoing compliance with the Transnuclear Certificate of Compliance No. 1004 and the associated Technical Specifications, the Transnuclear Standardized Nuclear Horizontal Modular Storage (NUHOMS) Updated Final Safety Analysis Report (UFSAR), and the regulations in 10 CFR Part 20 and Part 72. Particular emphasis was placed on the modifications made to increase the rated capacity of your crane and to maintain it as single failure proof in accordance with U. S. Nuclear Regulatory Commission (NRC) guidance.

The inspection reviewed changes made to your ISFSI program since the last NRC inspection in August 2009 in the areas of radiation safety, cask thermal monitoring, quality assurance, corrective action program, safety evaluations, upgrades to the auxiliary building crane, and the effects on your ISFSI of the 2011 flooding that occurred at your site. As a result of the inspection, the NRC has determined that one Severity Level IV violation of NRC requirements occurred. This violation related to the requirement in 10 CFR 72.30(b) to submit an ISFSI Decommissioning Funding Plan to the NRC for review and approval by December 17, 2012. Contrary to this requirement, Fort Calhoun had not submitted their ISFSI Decommissioning Funding Plan by the required date. This violation is being treated as a Non-Cited Violation (NCV), consistent with Section 2.3.2 of the Enforcement Policy. This NCV is described in the subject inspection report.

If you contest this NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional

Administrator, Region IV DMNS Director, Office of Enforcement, and the NRC Resident Inspectors at Fort Calhoun.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if you choose to provide one, will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), 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.

Should you have any questions concerning this inspection, please contact the undersigned at 817-200-1191 or Mr. Lee Brookhart at 817-200-1549.

Sincerely,

/RA/

D. Blair Spitzberg, Ph.D., Chief
Repository & Spent Fuel Safety Branch
Division of Nuclear Materials Safety

Dockets: 50-285, 72-54
Licenses: DPR-40

Enclosure w/attachments:
Inspection Report 05000285/2013007;
07200054/2013001

Attachments:
1. Supplemental Information
2. Loaded Casks at Fort Calhoun Nuclear Station

cc w/attachments: Listserv®

Administrator, Region IV DMNS Director, Office of Enforcement, and the NRC Resident Inspectors at Fort Calhoun.

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Letter to Louis Cortopassi from D. Blair Spitzberg, dated April 24, 2013

SUBJECT: FORT CALHOUN STATION INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) INSPECTION REPORT 05000285/2013007 AND 07200054/2013001

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	05000285, 07200054
Licenses:	DPR-40
Report Nos.:	05000285/2013007 and 07200054/2013001
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Nuclear Station, Independent Spent Fuel Storage Installation (ISFSI)
Location:	P.O. Box 310 Fort Calhoun, NE 68023
Dates:	January 14-17, 2013
Inspectors	Vincent Everett, Senior Inspector Lee Brookhart, Inspector
Accompanying Personnel:	Eric Simpson, RIV RSFS, Inspector-in-Training
Approved By:	D. Blair Spitzberg, Ph.D., Chief Repository and Spent Fuel Safety Branch Division of Nuclear Materials Safety

Enclosure

EXECUTIVE SUMMARY

Fort Calhoun Nuclear Station
NRC Inspection Report 05000285/2013007 and 07200054/2013001

The U. S. Nuclear Regulatory Commission (NRC) conducted a routine inspection of activities related to the safe handling and storage of spent nuclear fuel at the Fort Calhoun Independent Spent Fuel Storage Installation (ISFSI). The inspection included a review of selected topical areas to evaluate compliance with the applicable NRC regulations and the provisions of your general license in accordance with the Transnuclear Standardized Nuclear Horizontal Modular Storage (NUHOMS) system. Ten concrete horizontal storage modules (HSMs) were currently loaded and stored on the Fort Calhoun ISFSI pad. The HSMs were being maintained in good condition. Radiological conditions around the ISFSI were low. A review of the environmental monitoring program demonstrated that radiological exposures to offsite locations were not occurring from the storage of the spent fuel at the ISFSI. Personnel exposures during cask loading for the 2009 loading campaign were low and were comparable to the doses typically seen during loading campaigns at the other Region IV sites which have shown low personnel doses. Temperature monitoring of the HSMs was being performed in accordance with technical specifications with temperature readings below technical specification limits. The quality assurance program and corrective action program were being effectively implemented to capture and correct issues related to the dry cask storage program.

Two significant areas were reviewed during this inspection. The first involved the impact of flooding on the storage of the spent fuel at the ISFSI. Though the 2011 flood did not reach the elevation of the ISFSI pad, a review was performed to determine the potential for a more significant flood and its possible effects, including one that would result from failure of upstream dams. The second area involved the up-rating of the auxiliary building crane from 75 tons to 106 tons. This inspection report documents the modifications to your crane trolley to increase the load capacity. The review of the seismic stability of your auxiliary building support structures to hold the weight of a loaded OS197H transfer cask was incorporated into the NRC Region IV inspection of your seismic modeling for several of your building structures being performed by the Inspection Manual Chapter (IMC) 350 inspection team. The conclusions reached concerning the auxiliary building crane's support structures are included in NRC Inspection Report 50-285/2013-008.

Operation of an ISFSI at Operating Plants (60855.1)

- The Fort Calhoun Quality Department had included ISFSI related activities in their audit and surveillance program. Quality assurance audits and surveillances performed in 2011 reviewed activities and documentation associated with the ISFSI program. No significant conditions adverse to quality were found. (Section 1.2.a)

- Radiological conditions at the ISFSI were evaluated which included conducting a survey of the area around the ISFSI and the HSMs. Radiation levels recorded on the dosimeters around the ISFSI pad showed low radiation levels, as expected for an ISFSI with ten casks. Offsite monitoring data from the 2009, 2010, and 2011 annual environmental reports documented that there were no offsite radiological impacts attributable to ISFSI operations. (Section 1.2.b)

- Documents and records related to the 2006 and 2009 ISFSI cask loading campaigns were reviewed. Information included personnel electronic alarming dosimetry (EAD) records, canister survey records, and estimated neutron doses to workers during cask loading activities. Worker doses to load a cask have continued to improve, with the last campaign averaging 0.084 person-rem/cask. (Section 1.2.c)

- Technical Specification 1.3.2 temperature monitoring requirements for the HSMs were performed daily as required. (Section 1.2.d)

- Required records were maintained that described the specific fuel parameters for the spent fuel stored in each of the licensee's loaded casks. (Section 1.2.e)

- Selected condition reports were reviewed for the period 2009 through 2012. A wide range of issues had been identified and resolved. Resolution of the issues was appropriate for the safety significance of the issue. No adverse trends were identified during the review. (Section 1.2.f)

- The ISFSI pad base mat was built at an elevation of 1009 feet 10 inches. This is higher than the Army Corps of Engineer's calculated maximum probable flood height without an upstream dam failure of 1009 feet 4 inches. During 2011, flood waters reached 1006 feet 11 inches. Though the ISFSI pad was not flooded, it was surrounded by water. After the 2011 event, Fort Calhoun began the process to re-evaluate the maximum possible flood level at the site, including a possible upstream dam failure, and the affect on the stored fuel and HSMs. (Section 1.2.g)

- The licensee upgraded the existing 75 ton single failure proof auxiliary building HE-2 crane to a rated load capacity of 106 tons to support the 2009 cask loading campaign. An NRC in-office review of the crane upgrade documentation was performed after the onsite inspection was completed. NRC inspectors reviewed documentation related to the replaced components, the basis for the crane's designation as single-failure proof, the 125 percent and 100 percent crane load tests, the 200 percent hook load test, the 300 percent lift yoke load test, the newly replaced wire ropes' breaking strengths, the crane's revised operating procedures, and the calculated maximum weight of the loaded cask that will be lifted by the crane. No issues related to the trolley were identified during the review. Concurrent with this inspection, the NRC's IMC 350 inspection team was in the process of reviewing the seismic methodology that had been used for various structures at Fort Calhoun. This review included the adequacy of the auxiliary building crane support structures and is documented in Inspection Report 50-285/2013-008. (Section 1.2.h through m)

- On an annual basis, the licensee performed a visual inspection of the ISFSI pad and accessible HSMs' surfaces for signs of degradation or structural cracking. No new degradation or cracking has been observed at the ISFSI pad or on the HSM surfaces since 2009 when a number of small cracks were documented. All documented defects were evaluated as superficial surface issues which would not affect the function of the HSMs or ISFSI pad. (Section 1.2.n)

- Each holder of a license under 10 CFR Part 72 must submit a decommissioning funding plan for NRC review and approval in accordance with 10 CFR 72.30(b). Per Federal Register Notice 35573 dated June 17, 2011, this new rule took effect on December 17, 2012. Contrary to this requirement, Fort Calhoun had not submitted their ISFSI decommissioning funding plan by December 17, 2012. The NRC has determined that this is a Severity Level IV Violation of 10 CFR 72.30(b). Since the licensee entered the issue into their corrective action program, and because the issue was not a repetitive violation or willful, this violation is being treated as a Non-Cited Violation (NCV), consistent with Section 2.3.2 of the NRC Enforcement Policy. Subsequent to this inspection, the licensee submitted the ISFSI Decommissioning Funding Report on March 6, 2013. (Section 1.2.o)

Review of 10 CFR 72.212(b) Evaluations (60856.1)

- The licensee was maintaining the 10 CFR 72.212 Evaluation Report current as required. No changes to the 10 CFR 72.212 report had been made since the last NRC inspection in 2009. (Section 2)

Review of 10 CFR 72.48 Evaluations (60857)

- All required safety screenings and safety evaluations had been performed in accordance with procedures and 10 CFR 72.48 requirements. All screenings reviewed were determined to be adequately evaluated. No safety evaluations were performed since the last inspection in 2009. (Section 3)

Report Details

Summary of Facility Status

Ten HSMs consisting of the HSM-H design and containing 32PT dry shielded canisters (DSCs) were currently in storage at the Fort Calhoun ISFSI. Each 32PT canister holds 32 pressurized water reactor spent fuel assemblies. The HSMs were located within the plant's Part 50 protected area. Two loading campaigns had been performed at Fort Calhoun. The first loading campaign in 2006 loaded four canisters. The second loading campaign in 2009 loaded six canisters. A tour of the ISFSI pad and adjacent area found the HSMs to be in good physical condition. The HSMs were being monitored in compliance with Technical Specification 1.3.2 "Thermal Performance." Dosimeters along the ISFSI fence were providing radiological dose data within the expected levels for an ISFSI with ten HSMs in storage. The current ISFSI pad can hold 40 HSM storage modules in four 2 x 5 arrays. Currently only one 2 x 5 HSM array had been completed and loaded with canisters. In 2009, Fort Calhoun upgraded their 75 ton auxiliary building Ederer X-SAM crane to 106 tons. This allowed the licensee to utilize the 100 ton OS197H transfer cask during the 2009 loading campaign. Previously, Fort Calhoun had utilized the 75 ton lightweight OS197L transfer cask.

The first loading campaign in 2006 was completed using Certificate of Compliance No. 1004, Amendment 8 and the Transnuclear Updated Final Safety Analysis Report (UFSAR), Revision 9. The second loading campaign in 2009 was completed using Certificate of Compliance No. 1004, Amendment 9, and UFSAR, Revision 10.

The reactor at Fort Calhoun is a Combustion Engineering (CE) pressurized water reactor with a core containing 133 fuel assemblies. Refueling outages are performed every 18 months. The spent fuel pool contained 1083 spaces, of which 158 were open and available. The reactor was currently defueled with 868 spent fuel assemblies in storage in the spent fuel pool. The spent fuel pool also contained 36 new fuel assemblies and 21 unusable spaces that contained debris cans, cooling discharge/suction piping, and dummy bundles.

1 Operations of an Independent Spent Fuel Storage Installation (ISFSI) at Operating Plants (60855.1)

1.1 Inspection Scope

An inspection of the status of the loaded casks at the Fort Calhoun ISFSI was completed to verify compliance with requirements of the Transnuclear Certificate of Compliance No. 1004 and the Transnuclear UFSAR. The inspection reviewed a broad range of topics including audits and surveillances conducted by the licensee, condition reports related to the ISFSI and the auxiliary building HE-2 crane, environmental radiological data collected around the ISFSI for the past several years, compliance with Technical Specification 1.3.2 for temperature monitoring of the casks, current issues associated with the ISFSI design basis flood level, and the up-rating of the auxiliary building HE-2 crane from 75 tons to 106 tons. A tour of the ISFSI area was completed and radiological dose rates were measured around the perimeter of the ISFSI pad and near the HSMs.

1.2 Observations and Findings

a. Quality Assurance Audits and Surveillances

The Fort Calhoun Quality Department had included ISFSI related activities in their audit and surveillance program. A review was conducted of the ISFSI quality assurance (QA) related surveillances and audits performed since the last inspection in August 2009. A QA audit performed by the licensee during October 2011 reviewed the activities and documentation associated with the ISFSI program. This included a review of records for all the spent fuel loaded during the 2006 and 2009 loading campaigns. Quality Assurance Audit Report 11-QUA-084, issued on November 15, 2011, found that the records adequately documented the fuel bundle information for all spent fuel assemblies in each of the loaded casks. The QA audit did not find any significant conditions adverse to quality within the ISFSI program.

In addition to the QA audit in 2011, the following three QA surveillance reports related to nuclear fuel handling were reviewed: Quality Department Surveillance Report 09-QUA-076, dated December 17, 2009, Quality Department Surveillance Report 10-QUA-089, dated December 28, 2010, and Quality Department Surveillance Report 11-QUA-048, dated June 24, 2011. Of these three reports, Surveillance 09-QUA-076 resulted in one condition report being issued. Condition Report (CR)-2009-4256 was initiated on September 14, 2009 to document a safety hazard in the lay-down area, also known as the Room 68 platform. Netting was installed in the opening of the platform where workers would perform decontamination, drying, and lid welding activities on a loaded canister. The netting presented a slip, trip, or fall hazard. The condition report was closed on October 14, 2009 with the removal of the netting and completion of a re-engineered platform surface that did not present a trip hazard.

b. Radiological Conditions of the ISFSI

A radiological survey of the ISFSI pad and perimeter was performed by Fort Calhoun personnel and observed by the NRC inspectors during a tour of the ISFSI. Gamma survey instrument readings were taken of the ISFSI pad perimeter and from three specific cask locations. The survey results were consistent with the most recent radiological survey performed by the licensee. Typical radiation levels around the ISFSI pad were from 7 μR/hr (background) to 15 μR/hr. Radiation levels around the HSM lower vents were 5 to 12 mR/hr (5,000 to 12,000 μR/hr) on contact.

Four thermoluminescent dosimeters (TLD) were located in close proximity to the ISFSI pad. The four TLDs were mounted on the ISFSI perimeter fence, located roughly north, east, west, and south of the ISFSI pad. The closest TLDs were at the west and south fence locations, approximately 75 and 125 feet, respectively, from the HSMs. The other two TLDs were at the north and east fence monitoring locations, approximately 300 and 600 feet away, respectively. Radiological data for the period of 2009 thru the first half of 2012 was reviewed. As expected, the closest (west) TLD showed higher readings than the other TLDs. Over the three year period, the west TLD averaged 10 mrem/yr above background. The north TLD consistently recorded only background levels. The east TLD recorded 18 mrem for 2010, but showed zero for 2011 and 2012. The south TLD recorded 9 mrem in 2010 and 6 mrem in 2011, but zero in 2012. This indicated that other site activities were influencing the east and south TLD's readings.

Environmental data reviewed in support of this inspection included offsite monitoring data from the 2009, 2010, and 2011 annual environmental reports, TLD monitoring data from 1998 through 2012, and Procedure CH-ST-RV-0003 “Environmental Sample Collection – Quarterly Environmental Dosimeters (TLDS).” The licensee maintained environmental TLDs around the site in all 16 sectors. The background or control sample location was located in Sector L, 19.6 miles southwest of the site. Table 2.0 “Radiological Environmental Monitoring Program Summary” in each of the annual reports showed the control TLD mean dose rate as 1.3 mrem/week (7.7 μrem/hr). The mean dose rate for all the environmental TLDs, excluding the control, was 1.4 mrem per week (8.3 μrem/hr) consistent for all three years, ranging from 0.7 to 1.8 mrem/week. The nearest environmental TLDs to the ISFSI pad were in Sector P and Sector Q. Sample Station Nos. 1 and 55 were located 0.5 and 0.6 miles from the plant in Sector P. Sample Station No. 56 was located 0.7 miles from the site in Sector Q. The 2009 thru 2011 average value for these three TLDs was 1.3 mrem/week (7.7 μrem/hr). This was the same value as the control TLD. Based on these data points, the casks loaded at the ISFSI did not have a measurable impacting on offsite doses. The annual environmental report for 2012 had not been completed and issued. A review of the data that had been collected showed an increase in the doses measured at the three locations. The average value had increased to 1.5 mrem/week (9.1 μrem/hr). This value was within the statistical variation (0.7 to 1.8 mrem/week) of all offsite environmental monitoring data for the previous three years.

c. Radiological Information Related to Cask Loading

Documents and records related to the 2006 and 2009 ISFSI cask loading campaigns provided by the licensee’s dosimetry and ALARA groups were reviewed. The information reviewed included personnel electronic alarming dosimetry records, survey records, and estimated neutron doses to workers during cask loading activities. The dose records for the 2006 loading campaign (Casks 1 thru 4) were approximations. The dose records for the 2009 loading campaign (Casks 5 thru 10) were presented by the ALARA personnel as being more accurate. The 2006 loading campaign had the highest person-rem doses and ranged from 201 to 534 mrem per cask, averaging 341 mrem. For the 2009 loading campaign the doses ranged from 64 to 111 mrem per cask, averaging 84 mrem. This represented a 75 percent reduction in personnel dose and was comparable to dose values at other Region IV plants with low overall doses. The lower 2009 doses were attributed to the use of the OS197H transfer cask. The 2006 campaign had used the OS197L transfer cask which was a lighter cask with less shielding. The OS197L transfer cask was replaced with the OS197H transfer cask when the auxiliary building HE-2 crane was upgraded to 106 tons in 2009.

During both loading campaigns, worker gamma doses were tracked using electronic alarming dosimetry. Neutron doses were estimated using stay time calculations and remball neutron survey readings. The ALARA staff provided gamma and neutron dose information for eight job functions during the loading campaign in 2006. Of those eight job functions (see Table 1 below), seven documented neutron doses. Neutron doses accounted for an estimated 45 percent of the worker dose. During the 2006 campaign, the highest neutron and overall doses were received during canister lid welding activities and radiation protection support operations.

Table 1: Gamma vs. Neutron Dose Estimates (mrem) by Task During 2006

Cask Loading Job Function	Gamma	Neutron	Total	% Neutron
1) Spent fuel pool work	119	34	153	22%
2) Welding and Room 68 work	82	320	402	80%
3) Cask transport to/from HSM	267	10	277	4%
4) RP and Operations Support	301	185	486	38%
5) Quality Control Support	22	57	79	72%
6) Regulatory Oversight	5	11	16	69%
7) Management Oversight	10	42	52	81%
8) Fuel Movement	10	0	10	0%
Total:	816	659	1475	45%

For the 2009 loading campaign, the number of cask loading job functions being tracked increased (see Table 2 below). Five activities documented neutron doses. The neutron dose accounted for an estimated 30 percent of the dose. In 2009, the highest neutron and overall doses were seen during canister welding and canister transfer activities.

Table 2: Gamma vs. Neutron Dose Estimates (mrem) by Task During 2009

Cask Loading Job Function	Gamma	Neutron	Total	% Neutron
1) Visual Fuel Inspection	32	0	32	0%
2) Radiation Protection Support	0	0	0	0%
3) Preps for Dry Cask Activities	16	0	16	0%
4) DSC Preparation work	9	0	9	0%
5) Transfer from SFP to Rm. 68	42	11	53	21%
6) Decon of the cask pit, and DSC	54	21	75	28%
7) DSC sealing and welding activities	81	37	118	31%
8) DSC transfer and storage	135	87	222	39%
9) Routine Decontamination	0	0	0	0%
10) Management Oversight	2	1	3	33%
Total:	371	157	528	30%

An ALARA Post-Job Review FC-RP-301-10, Revision 0 provided additional neutron and gamma survey information for the six canisters loaded during the 2009 campaign. There were measureable neutron dose rates associated with all six canisters. In several cases, the neutron dose rates equaled or exceeded gamma dose rates at some locations on the canister. The highest measured occupational exposures occurred when workers were required to perform operations in close proximity to the canister. The ALARA records indicated that TLD gamma results recorded on the dose of legal record (DLR) for individuals generally agreed with the electronic alarming dosimetry gamma measurements.

Despite the presence of measureable neutron dose rates during cask loading activities, the TLDs (Panasonic Model 802) used as the dose of legal record, which contained a lithium neutron chip, did not record any neutron dose for any of the workers. Based on neutron estimates using the remball, the neutron component was contributing roughly 30 percent to 45 percent of the overall personnel dose during cask loading activities. This disparity was documented as CR-2009-3827 on August 19, 2009. Numerous reasons were identified by the licensee that could result in inaccurate neutron dose estimates, including: (a) the neutron dose calculation used by the TLD processing facility assumed an average neutron energy of 600 keV, while the average neutron energy exposures during the loading campaign may have been closer to 70 keV; (b) the standard neutron dose rate instrument used during the loading activities (remball) under-responded to thermal energy neutrons; (c) a detailed gamma and neutron dose calculation was performed prior to the first loading campaign in 2006 and not repeated for the different transfer cask used in 2009; and (d) field measurements capable of accurately determining the neutron energy were not performed for either dry cask loading campaign.

The licensee initiated several activities to evaluate the situation. Testing was performed on dosimeters used during the loading of the first two casks to verify their accuracy; a test was performed using a remball to determine a gamma to neutron ratio; and the use of neutron sensitive electronic dosimeters was initiated. The remball test involved taking readings around Cask #9 (FCS32PT-S100-HZ05) after the annulus was drained on August 21, 2009. The data was documented as Survey #09-0716. The remball measured neutron levels of 4 mrem/hr on contact and 4 mrem/hr out to three feet. When the detector probe was removed from the remball's 9" poly sphere, 80 mrem/hr was measured on contact and 75 mrem/hr at three feet. The licensee believed the readings indicated that the neutron field being measured was thermalized. The OS197H transfer cask has a three inch neutron water shield. Gamma readings were also taken at the same locations and recorded 9 mR/hr on contact and 4 mR/hr at three feet.

Since the issuance of CR-2009-3827, the licensee has initiated the use of a neutron sensitive electronic dosimeter capable of better quantifying neutrons including those in the thermal energy range. The newer electronic dosimeter may be more useful for tracking personnel neutron doses in future loading campaigns than the estimation method used in the 2006 and 2009 loading campaigns.

The neutron dose estimates were recorded onto an unofficial form called "Platform Entry Neutron Tracking." The dose estimates were not included in the official personnel dosimetry file. This issue was documented in Condition Report CR 2009-3721 on August 13, 2009. As of this inspection, no neutron doses have been added to the official personnel dosimetry files as the licensee continues to evaluate whether the neutron values determined using the record keeping process accurately reflected the neutron doses received by the workers.

d. <u>Technical Specification 1.3 Surveillance and Monitoring</u>

Technical Specification 1.3 required either a daily inspection of the inlet and outlet vents for blockage at each HSM, per Technical Specification 1.3.1, or temperature measurements of the thermal performance for each HSM, on a daily basis, per Technical Specification 1.3.2. The licensee had equipped all HSMs with temperature monitoring equipment and was implementing the daily temperature monitoring requirements in compliance with Technical Specification 1.3.2. If the temperature monitoring equipment malfunctioned or was not in operation, the licensee performed vent inspections in accordance with Technical Specification 1.3.1. The licensee implemented Procedure OP-ST-SHIFT-001 "Operations Technical Specification

Required Shift Surveillance", Revision 108 to comply with the Technical Specification. The procedure required operators to verify that each HSM temperature was < 180 degrees Fahrenheit (F) and the temperature increase within the last 24 hours was < 11 degrees F. If the acceptance criteria was not met, the operators were required to check the backup temperature element for the affected HSM, inspect the vent openings of the affected HSM, and remove any blockages, if found, at the affected HSM.

Temperature monitoring documentation was reviewed for the months of September 2010, October 2011, February 2012, and June 2012 to verify compliance with the Technical Specification. For all four months selected for review, temperature monitoring or vent inspections were performed daily, as required. For all the days of the selected months reviewed, no cask vents were reported as being blocked or exceeded temperature monitoring criteria.

e. Records Related to Fuel Stored in the Casks

Permanent records describing the spent fuel stored in the casks are required by 10 CFR 72.212(b)(12). A review of the licensee's records was performed to determine if an adequate description of the spent fuel loaded in the casks was contained in permanent records. The contents of each loaded cask were documented in a number of locations. The procedures were maintained by the reactor engineering group. Administrative Procedure RE-AD-0005 "Fuel Selection and DSC Planning for Dry Cask Storage," Attachment 8.1 – "DSC Fuel Selection Worksheet," dated April 30, 2009, contained specific fuel bundle identifications for each fuel assembly in every loaded cask. Procedure RE-ST-DFS-0001 "Unit 1 Surveillance Test/Fuel Selection Verification for Placement in Dry Fuel Storage" Attachments 2 and 3, dated February 2, 2006, contained fuel bundle specifications including maximum U-235 enrichment, burnup, uranium content, cooling time, discharge date, decay heat, and supporting fuel selection criteria from the UFSAR. The quality assurance program had reviewed the records for the spent fuel as documented in the Unit No. 1 Technical Data Book/Dry Fuel Storage Layout (TDB-I.B.5), dated November 16, 2009 for all casks loaded during 2006 and 2009. Quality Assurance Audit Report No. 48/Nuclear Fuel 11-QUA-084, issued November 15, 2011, found that the records had adequately documented fuel bundle information for all spent fuel assemblies in each of the loaded casks.

f. Corrective Action Program

A list of condition reports issued since the last NRC inspection in August 2009 was provided by the licensee for the auxiliary building HE-2 crane and the ISFSI. The condition reports were processed in accordance with Procedure SO-R-2 "Condition Reporting and Corrective Action", Revision 53a. Section 2.2 "CR Significance Levels" provided a description of the safety categorization for the condition reports. Those classified as "Level 1 and Level 2" were considered adverse conditions, with the "Level 1" condition reports considered significant. "Level 3" conditions were non-significant issues that needed to be investigated and corrected. "Level 4" condition reports were minor issues and "Level 5" were low level problems typically closed to immediate actions taken. Forty selected condition reports were identified by the NRC for further review. The condition reports related to a number of topics including performance of 10 CFR 72.48 screenings for maintenance related activities, recalibration of the load cell on the auxiliary building HE-2 crane, malfunction of the HSM temperature monitoring equipment that required repair, evaluation of whether neutron monitoring should be required around the ISFSI pad, replacement of a coupling on the auxiliary building HE-2 crane, updating the ISFSI fire hazards analysis and/or associated drawings due to the movement of equipment, replacement of the failed filter unit in the compressed air system of the auxiliary building HE-2 crane, updating the 72.212 report due to the rise in Fort Calhoun's maximum probable flood level, 72.48 screening requirements to justify the storage of diesel generator security lights on the ISFSI pad, installation of scaffolding on the side of the ISFSI so operators could view the outlet vents as an alternative to using an extension ladder, flooding of the ISFSI's temperature monitoring equipment shed, and minor concrete defects found on the HSMs and ISFSI pad.

The condition reports reviewed were well documented and properly categorized based on the significance of the issue. The corrective actions taken were appropriate for the situation. Based on the level and detail of the corrective action reports, the licensee demonstrated a high attention to detail in regards to the maintenance and operation of the ISFSI program and auxiliary building HE-2 crane. No NRC concerns were identified related to the condition reports reviewed.

g. <u>ISFSI Design Basis Flood Level</u>

A design basis flood of an ISFSI was analyzed in the UFSAR, Section 8.2.4 "Flood." The analysis considered a flood resulting from high river water or a dam failure resulting in a 50 foot high hydrostatic head of water producing external pressure on the canister. Section 8.2.7 "Blockage of Air Inlet or Outlet Openings" discussed situations where the vents on the HSM were blocked, including from a flood condition. Section P.3.7.3 "Flood" discussed the forces on the HSM-H from the design basis 15 feet/second flow rate of the flood water. During a flood, the spent fuel temperature for a 24 kW canister will not reach a level that could damage the fuel cladding or the canister integrity. However, after approximately 40 hours of vent blockage, the HSM-H concrete could reach a temperature of 350 degree F, which could degrade the concrete. As such, Technical Specification 1.3 was established to limit any blockage of the vents to 40 hours. UFSAR Figure 8.2-16 "HSM Roof Internal Concrete Temperatures Following Vent Blockage" provided a graph showing the temperature affect on the HSM concrete over a five day period with both vents blocked. With an ambient temperature of 125 degree F, the temperature of the concrete in the HSM reaches 350 degree F in 40 hours and 450 degree F in 5 days.

The Fort Calhoun Station Updated Safety Analysis Report (USAR), Revision 12, Section 2.7.1.2 "River Stage and Flow" stated that the Army Corps of Engineers estimated the maximum probable flood resulting from the maximum probable rainstorm at the site would yield a peak water discharge of 550,000 cubic feet/second, yielding a peak flood stage estimate of approximately 1009.3 feet (roughly 1009 feet 4 inches) Mean Sea Level (MSL). A follow-on calculation of the flood resulting from the failure of an upriver dam concurrent with the events giving rise to the maximum probable flood was estimated by the Army Corps of Engineers to yield a water discharge of approximately 1,200,000 cubic feet/second at the plant site and a flood elevation of 1013 to 1014 feet MSL.

The top of the ISFSI pad base mat was at elevation 1009 feet 10 inches. This is higher than the maximum probable flood height of 1009 feet 4 inches without a dam failure by a half foot. During 2011, flood waters reached as high as 1006 feet 11 inches. The ISFSI pad was not flooded, but the pad was surrounded by water. After the 2011 flood, the licensee began the process to re-evaluate the maximum possible flood level at the site and how flooding involving an upstream dam failure or a flood exceeding the previously accepted maximum flood height could affect the ISFSI.

A review of documents and technical drawings related to the NUHOMS storage system design showed that the HSM-H measured 178 inches (14 feet 10 inches) from base to top edge. The HSM-H roof block added 44 inches (3 feet 8 inches) of concrete to the top of the HSM-H structure to bring the overall installed height to 222 inches (18 feet 6 inches). The HSM-H, when specified for use with the 32PT canister, holds the canister at a centerline height of 106 inches. The canister has a radius of 33.595 inches. Based on this information, the bottom of a loaded canister will be at a height of 72.405 inches or slightly over 6 feet above the ISFSI pad. The top of the canister will be at approximately 11 feet 8 inches above the ISFSI pad. The top of the HSM-H would be at 1028 feet 4 inches.

The HSM-H has a lower vent with a front facing screen opening that rises 30 inches (2 feet 6 inches) above the ISFSI pad leading to an opening into the annulus region between the canister and the HSM-H. This opening into the annulus region behind the screen extends to a height 8 inches above the ISFSI pad and is 12 feet 4 inches wide. As such, 8 inches of water or mud/silt on the ISFSI pad that gets past the lower vent screen would result in full blockage of the opening into the annulus region and prevent air flow to the inside of the HSM-H and the canister. This would equate to a flood level of 1010 feet 6 inches. From this point until the water came into contact with the canister at 1015 feet 10 inches, the canister would be in the worst case thermal condition for heating. Once the water reached the canister, cooling of the canister would be re-established. Until then, there would be some cooling effect on the canister and HSM-H by the water due to an evaporative process resulting from the heat generated by the canister. Any flood less than 1010 feet 6 inches would allow for air to continue flowing through the HSM-H lower vent, albeit restricted flow as the waters approach that elevation. As discussed in the UFSAR, Section 8.2.7, the negative effect of the blocked lower vent is on the limits applied to the concrete, not on the stored spent fuel or canister. Once the flood water has receded, any mud or debris blocking the 8 inch vent opening would dry out and would result in the worst case thermal condition for the internal concrete portions of the HSM-H. Table 3 below provides a summary of the heights associated with the HSM-H and the canister at Fort Calhoun.

Table 3: HSM-H Component Heights and Design Bases Flood Impacts

HSM-H Component	Height	Height Plus Elevation	Impacted by 1014' Flood Water?
Top of ISFSI pad	0"	1009' 10"	Yes
Lower inlet vent blocked	8"	1010' 6"	Yes
DSC lower shell edge	6'	1015' 10"	No
DSC upper shell edge	11' 8"	1021' 6"	No
HSM-H upper vents	14' 2"	1024'	No
HSM-H top (excluding roof block)	14' 10"	1024' 8"	No
HSM-H with roof block	18' 6"	1028' 4"	No

h. Auxiliary Building (HE-2) Crane

When the Fort Calhoun nuclear station was constructed, a 100 ton Harnischfeger crane trolley was installed in the auxiliary building. The Harnischfeger trolley did not meet the criteria of NUREG 0554 "Single Failure Proof Cranes for Nuclear Power Plants." In 1986, installation and acceptance testing of a new Ederer trolley was completed replacing the Harnischfeger trolley. The Ederer trolley was rated at 75 tons and was a single failure proof trolley. Between March 13 and May 5, 2006, the NRC conducted a preoperational inspection of the licensee's preparations to begin loading spent fuel into dry cask storage at the Fort Calhoun ISFSI. Inspection Report 72-54/06-02 was issued July 19, 2006 (Adams Accession No. ML062000421). This inspection included a review of the Ederer 75 ton crane (HE-2) installed at Fort Calhoun to confirm it's classification as a single failure proof crane.

During the 2006 inspection, the NRC identified concerns with the licensee's plans to use the light weight Transnuclear OS197L transfer cask because of the 75 ton weight limitation of the crane. Use of the OS197L transfer cask would result in high dose rates near the cask which would require remote operations for a portion of the cask movement from the spent fuel pool. The NRC determined that the licensee had performed an inadequate safety evaluation to conclude that the OS197L transfer cask was acceptable. The NRC approved an exemption for Fort Calhoun to use the OS197L transfer cask on July 19, 2006 (Adams Accession No. ML062000153), but limited the loading to four casks.

In 2007, Ederer Nuclear Crane Division performed an Engineering Up-Rate Study dated October 9, 2007 (Document No. 70587330) and a Structural Analysis, dated October 8, 2007, which included calculations to up-rate the crane to 105 tons. These were included in Calculation FC07263 and evaluated the new loads on the mechanical crane components. Calculation FC07262 "Auxiliary Building Crane (HE-2) Support Structure Evaluation" dated February 18, 2008 analyzed the auxiliary building support structures for the crane including the effects of environmental factors on the crane structures such as tornados, earthquakes, high winds, and seismic induced pendulum and swinging load effects of the suspended load. Both dead loads and live loads were analyzed. A 3-D crane model was used which evaluated the forces and moments in the bridge girders, end ties, end trucks and trolley frame structural members, main hoist rope forces, trolley and crane wheel reactions and maximum main hook displacements. Several trolley positions on the bridge girder, hook positions, and bridge girder positions on the concrete runway girders were evaluated. Components on the crane requiring modifications for the new 105 ton load were identified.

After completion of the analysis to up-rate the crane to 105 tons, it was realized that the weight of the loaded OS197H transfer cask required a load rating of 106 tons. The licensee determined that that the original F-1224 main hoist lower block, weighing 11,600 pounds, could be replaced with a newer and lighter lower block, weighing less than 9,000 pounds. This would reduce the weight associated with the trolley by one ton, thereby allowing a heavier load of 106 tons on the hook. By doing this, the extensive building and crane structural analysis would still be valid. Purchase Order 116712 was issued to purchase the new lower block. The weight of the trolley, block, hook and load used in the analysis discussed above was 283,000 pounds (141.5 tons). Subtracting the trolley, lighter block, and hook weight of 71,000 pounds (35.5 tons) left 106 tons (141.5 tons minus 35.5 tons) available for the weight of the cask. Revision 1 to the

Ederer Engineering Up-Rate Study was issued February 1, 2008, which revised the reference to the crane rating from 105 tons to 106 tons. Ederer issued new revisions to Topical Report EDR-1, Appendix B and Appendix C. Appendix B was revised on May 23, 2008 (Revision 3), and Appendix C on February 19, 2008 (Revision 1), to incorporate changes for the upgrade of the auxiliary building HE-2 Ederer X-SAM crane from 75 tons to 106 tons. A cross-reference was provided showing the regulatory requirements for a single failure proof crane compared to the Fort Calhoun crane rated at 106 tons. Calculation FC07473 "Auxiliary Building Crane (HE-2) 106 Ton Replacement Lower Block," dated April 15, 2008, evaluated the new lower block at 9,000 pounds. The Ederer Engineering Up-Rate Study (Document No. 70587330) was revised February 1, 2008 changing the 105 ton rating of the crane to 106 tons based on the use of the new lighter lower block.

Fort Calhoun completed the modifications and further analysis to up-rate the auxiliary building HE-2 crane to 106 tons to allow the use of the heavier OS197H transfer cask and to maintain the status as a single failure proof crane. The modifications, which included selected component replacements, were documented in Engineer Change EC-41654 "Upgrade of the Auxiliary Building Crane HE-2," Revision 0. The component replacements included: (a) changing the hoist gear ratio to reduce the maximum hoisting speed to 3.6 feet/minute; (b) replacing four gears in the hoist failure detection system to match the new ratio in the hoist speed reducer; (c) replacing the main hoist wire rope with a higher strength rope; (d) increasing the weld size on the main hoist drum bearing pillow block, the shear bars, and the hand brake dead end mounting bracket; (e) installing larger fasteners in the drum pinion pillow block support base; (f) installing a stronger sheave frame in the main hoist upper block; (g) replacing the upper block load cell sheave pin; (h) replacing the main hook hoist overspeed switch and speed responsive switch; (i) replacing the lower block and main hook; and (j) providing new drum brake actuators. The licensee also performed an analysis of the effect of the heavier cask load on the rail bay floor, including the weights associated with the 125 percent load test to ensure the floor could support the weight. This was documented in Calculation FC07484 "Evaluation of Cask Transfer Options (for 106 Ton Upgrade) in the Auxiliary Building Rail Bay at Elevation 1004 Feet," dated May 4, 2009.

The licensee performed the auxiliary building crane support structure seismic evaluations in Calculation Number FC07262 to verify the auxiliary building walls could hold the 106 ton load during a seismic event when using the OS197H transfer cask. This analysis used seismic methodology EA-FC-94-003 "Alternative Seismic Criteria and Methodology." The alternate methodology was reviewed and approved by the NRC in a letter to Omaha Public Power District (OPPD) entitled "Safety Evaluation of Alternate Seismic Criteria and Methodologies – Fort Calhoun Station (TAC No. M71408)," dated April 16, 1993. In that letter, the NRC specifically approved the use of the EA-FC-94-003 alternate seismic methodology for evaluation of heating, ventilation, and air conditioning (HVAC) equipment and piping. The EA-FC-94-003 methodology was subsequently used in numerous other structural evaluations within the reactor facility, including the auxiliary building. The NRC's Inspection Manual Chapter (IMC) 350 inspection team performed an evaluation of the use of the EA-FC-94-003 seismic methodology for other structural evaluations, including the auxiliary building crane support structures. The results of that evaluation are documented in Inspection Report 50-285/2013-008.

i. <u>Load Tests for the Auxiliary Building (HE-2) Crane and Ancillary Equipment</u>

Fort Calhoun's site acceptance test of the upgraded 106 ton auxiliary building HE-2 crane was performed in accordance with Ederer Procedure 70587543 "Site Acceptance Test OPPD Fort Calhoun Aux Building 106 Ton X-SAM Crane Upgrade," Revision 1. This procedure was performed under Work Order (WO) 00263046 "Testing MOD-EC 41654 HE-2 Crane Upgrade," Revision 0. The 125 percent load test was completed on May 16, 2009, and was documented in Procedure 70587543 as successfully lifting the test weight of 266,014 pounds (133 tons). The weight was recorded in Step 6.1.2 of Procedure 70587543. The required weight for the 125 percent load test was 132.5 tons to 136.5 tons. The weight was lifted approximately two feet and the drum brake set. The main hoist brake was released and the load was successfully maintained by the drum brake. The load was moved two to five feet in the east, west, north, and south directions, then lowered to one foot and held for one minute. This completed the 125 percent load test. The test weight value recorded in Procedure 70587543 differed slightly from a value documented in Work Order CWO-263046-07 "Auxiliary Building Crane, HE-2, Upgrade (EC Post Mod Testing)" Attachment 1 "Test Weight Verification" which listed each of the individual 13 weights that had been used for the 125 percent load test and their individual measured weight. The sum of these values added to 266,460 pounds (133.23 tons). Each of the individual 13 test weights had been measured in Section 4.0 "Work Instructions" of Attachment 1 using a digital 50,000 pound dynamometer (M&TE No. 67496). An e-mail dated April 3, 2009, from M. Newland, Fort Calhoun Station to K. Henry stated that the accuracy of the dynamometer was 0.1 percent full scale which equated to +/- 50 pounds. The weight of the rigging, which included the slings, shackles, and turnbuckles was estimated to be 4,784 pounds. The added rigging weight had been zeroed out prior to lifting each of the 13 test weights to document their actual weight. An e-mail from K. Henry to D. Stransky dated April 6, 2009, provided the rigging weights and confirmed that the rigging weights were not included in the weights for each of the 13 test weights. The slight difference between the weights as measured with the dynamometer and the weight recorded at the time of the 125 percent load test is not significant. The weight recorded at the time of the 125 percent load test (266,014 pounds) appears to be the value from the crane's load cell.

The 100 percent load test followed the 125 percent load test and was also performed in accordance with Ederer Procedure 70587543 and Work Order (WO) 00263046. The 100 percent load test was completed on May 18, 2009. The recorded weight in Step 6.2 of Procedure 70587543 was 219,074 pounds (109.5 tons). An acceptable weight for the 100 percent load test was 106 tons to 111 tons. The load was lifted two feet and held for one minute, lifted three feet and held for one minute, then lowered three feet and held for one minute to verify operations of the brakes. The weights were then transported on the trolley the full length of the bridge with the bridge close to the south end of the building. The test weights were then transported by the bridge the full length of the runway with the trolley close to the west end of the bridge. The test weights were returned to their starting point and the test was completed. After the 125 percent lift and 100 percent lift, the accessible structural welds were inspected by a structural engineer. No issues were found.

A 200 percent load test was performed on the new hook that was installed on the upgraded auxiliary building HE-2 crane. The hook replacement was part of Engineering Change EC-41654. The hook load test was performed by Forjas Irizar S.L. on October 8, 2008. The load test was documented in Forjas Irizar Certificate Number Q08/2664/PC08/88 "Load Testing" dated October 8, 2008. The hook was placed in a load testing machine and was loaded with an equivalent pressure to simulate a 212 ton load. The hook was dimensionally inspected prior to the load test and after the load test. The hook passed both magnetic particle non-destructive testing and ultrasonic non-destructive testing prior to the load test and after the load test. All dimension inspections and non-destructive examinations were found to be satisfactory.

In 2009, Fort Calhoun replaced their transfer cask lifting yoke (rated at 75 tons) with a higher capacity version to accommodate the heavier OS197H transfer cask. The new lifting yoke was rated at 110 tons and was compatible with the OS197H transfer cask. The yoke was designed to accommodate the space restrictions of Fort Calhoun's spent fuel pool. The new lifting yoke was similar in configuration to the previous one. The lifting yoke employed a spreader beam with remotely-actuated lifting arms. The new lifting yoke's remote operations were performed pneumatically instead of hydraulically, as with the original lifting yoke. The lifting arm openings that engaged the cask trunnions had less vertical length on the new yoke. This was necessitated by the need for bulkier lifting arms for the increased load, yet still being able to remain within the approximate 8 foot square footprint available in the spent fuel pool. The 300 percent load test was performed on the new lifting yoke on April 9, 2009. A load of 660,000 pounds (330 tons) was applied to the lifting yoke and held for ten minutes. This was witnessed by Fort Calhoun quality assurance personnel during a quality assurance surveillance at the test facility in Greensboro, North Carolina. The 300 percent load test was documented in NSS-NPS-09-008 "OPPD Surveillance No. FCS 09-VEN/S-005 Transnuclear (Columbian Hi Tech, Greensboro, NC)," dated April 17, 2009. After the load test, magnetic particle testing of each of the lifting yoke swing arms' critical load bearing welds were performed. No issues were found.

j. Wire Rope Break Testing for Auxiliary Building (HE-2) Crane

NUREG 0554 "Single Failure Proof Cranes for Nuclear Power Plants," issued May 1979, stated in Section 4.1, "Reeving System" that the maximum load on each individual wire rope in the dual reeving system with the maximum critical load attached should not exceed 10 percent of the manufacturer's published breaking strength.

The auxiliary building HE-2 crane used a dual reeving system containing two ropes of 1-1/4 inch, 6 x 37 class independent wire rope core, made of extra improved plow steel with a breaking strength of 159,800 pounds and a yield strength of 127,840 pounds. The HE-2 crane's dual reeving system contained 16 parts of rope. The crane was rated at 106 tons (212,000 pounds). As such, each reeve of rope would be expected to hold a maximum of 13,250 pounds (212,000/16). Ten percent of the manufacturer's published break strength (159,800 pounds) would be 15,980 pounds. In compliance with NUREG 0554 Section 4.1, the maximum load on a wire rope part of 13,250 pounds did not exceed 10 percent of the manufacturer's published breaking strength of 15,980 pounds.

Break tests were performed on the two new wire ropes at the University of Washington to determine the actual breaking strength of the ropes. Rope sections were load tested by the University of Washington Structural Engineering Laboratory to failure. The actual breaking strength of the wire ropes was documented in a report from the University of Washington under Purchase Order 4160044295, dated November 7, 2008. The wire rope tension tests were witnessed by personnel from PaR Nuclear, Inc. Ederer Nuclear Crane Division. The actual wire rope breaking strengths were found to be 177,370 pounds and

180,030 pounds. The actual breaking strengths were greater than the manufacturer's published breaking strength of 159,800 pounds.

k. Minimum Operating Temperature for the Auxiliary Building (HE-2) Crane

NUREG 0554, Section 2.4 "Material Properties" and NUREG 0612 "Control of Heavy Loads at Nuclear Power Plant," issued July 1980, Section C-2 (8) established a default minimum crane operating temperature of 70 degrees F if cold proof testing had not been performed on the crane. Cold proof testing was performed on the 75 ton crane prior to the upgrade. That testing allowed the licensee to lift up to 75 tons if the temperature in the auxiliary building was greater than 50 degrees F. Fort Calhoun did not perform cold proof testing when upgrading the crane from 75 tons to 106 tons. For loads greater than 75 tons, the auxiliary building must be greater than 70 degrees F for compliance with NUREG 0554/0612. The temperature operating restrictions were incorporated into Procedure GM-OI-HE-2 "Auxiliary Building Crane HE-2 Normal Operation," Revision 24. The crane operator was required by the procedure to record the ambient auxiliary building temperature in Attachment 6 "HE-2 Shift Log" prior to performing lifting operations. The restrictions on operating temperatures were specified in the procedure in Steps 3.2 and 5.4.

l. Maximum Loads on Auxiliary Building (HE-2) Crane

The maximum loads the upgraded 106 ton auxiliary building (HE-2) crane would be subjected to were evaluated in Transnuclear Calculation TN-11212-2 "NUHOMS 32PT-S100 Operational Lift Weight Calculation with OS197-3 (OS197H) Transfer Cask," Revision 1. Four different critical lifts were evaluated and the weight of the loaded transfer cask calculated. The maximum lifts occurred during the lifting of the transfer cask, with a fully loaded canister, from the spent fuel pool to the cask decon room and from the cask decon room to the transfer trailer. The calculation used the measured and certified weight of 104,208 pounds for the OS197H transfer cask (serial number OS197-3) in the evaluation. The OS197H transfer cask and heavy haul transporter used for the Fort Calhoun 2009 loading campaign were owned by Southern California Edison Company. The yoke weight used in the calculations was for the 110 ton capacity yoke used at Fort Calhoun. The yoke weight was 5,900 pounds. The weight used for the fuel assemblies included the fuel spacers and poison rods. For 32 assemblies, this equated to 38,400 pounds for the fuel, 1,312 pounds for the fuel spacers, and 2,500 pounds for the poison rods. Both the canister cavity and the annulus between the transfer cask and canister were assumed to be filled with water.

The maximum weight of the transfer cask containing a canister fully loaded with spent fuel and water, water in the neutron shield, water in the transfer cask annulus, and carried by the lifting yoke with the canister top shield plug installed was listed in Calculation TN-11212-2, Table 2 "Critical Lift 1 Fuel Pool to Cask Decon Room" as 213,890 pounds (106.95 tons). Since this was above the 106 ton operating limit of the crane, the licensee needed to remove a minimum of 227 gallons of water from the canister prior to fully removing the transfer cask and canister from the spent fuel pool. Licensee Procedure RE-RR-DFS-0002 "Dry Shielded Canister Sealing Operations," Revision 10 required the removal of 400 gallons to be conservatively under the 106 ton limit. With the administratively controlled procedure removing the 400 gallons of water, the total weight lifted by the crane was calculated to be 210,552 pounds (105.28 tons).

Once the canister had been welded, dried, and backfilled with helium, the transfer cask containing the canister was lifted from the cask decon room and lowered onto the heavy haul transport trailer. Calculation TN-11212-2, Table 3 “Critical Lift 2 Cask Decon Room to Transfer Trailer” listed the total weight of the transfer cask and canister, loaded with spent fuel and ready for transport to the ISFSI pad as 206,991 pounds (103.5 tons).

The values calculated in Transnuclear Calculation TN-11212-2 were consistent with the Transnuclear UFSAR, Table M.3.2-1 “Summary of the NUHOMS-32PT System Component Nominal Weights” when adjusted for the Fort Calhoun specific information.

m. <u>Possible Fuel Cladding Exposure to Air During Cask Movement from Spent Fuel Pool</u>

The removal of 400 gallons of water from the canister prior to lifting the transfer cask completely out of the spent fuel pool was performed by injecting helium into the canister per Steps 7.1.31 thru 7.1.33 of Procedure RE-RR-DFS-0002. This process displaced the water through the drain line into the spent fuel pool. Prior to moving the canister to the cask decon room, the helium supply line was disconnected. At this time, the canister lid was on the canister, but was not welded. As such, a pathway for the helium to leak out of the canister existed. With 400 gallons of water removed, the top portions of the spent fuel assemblies were above the water level in the canister. The helium supply lines were re-connected approximately 1/2 hour later and helium injection re-initiated prior to welding of the canister lid. Technical Specification 1.2.19 requires the use of helium for blowdown to ensure fuel cladding is not exposed to oxidizing atmospheres at high temperatures. The technical specification allows eight hours to correct the problem if the helium or water environment cannot be maintained. The technical specification specifically applies to the 61BTH and 32PTH1 canisters but does not state that it applies to the 32PT canister. The UFSAR included several references (Sections 1.3.2.2, 4.7.3.1, and 5.1.1.3) related to the use of air for blowdown of the canister to remove water. In these situations, the spent fuel would be exposed to an oxidizing atmosphere for short periods of time prior to the initiation of vacuum drying.

The NRC has issued guidance concerning the exposure of spent fuel to air. Interim Staff Guidance (ISG)-22 “Potential Rod Splitting Due to Exposure to an Oxidizing Atmosphere During Short-Term Cask Loading Operations in Light Water Reactor of Other Uranium Oxide Based Fuel” (Adams Accession No. ML061080663) issued May 8, 2006 provided guidance to licensees concerning the potential problem of allowing air to come into contact with spent fuel. The NRC recommended three possible options to address the potential for fuel oxidation due to air contact with the fuel pellets inside the fuel rod. These were (1) maintain the fuel in an environment of helium, argon, nitrogen, or other suitable inert gas; (2) assure there are no cladding defects in the fuel rods, including hairline cracks and pinhole leaks; or (3) determine a time-at-temperature profile of the fuel rods while they are exposed to an oxidizing atmosphere and calculate the expected oxidation to determine if a gross breach could occur. The ISG-22, Appendix A provided a discussion on the limitations of the data currently available to fully understand the potential for damage to the spent fuel in an air atmosphere. The licensee’s position concerning the practice of removing the helium supply from the canister during movement from the spent fuel pool to the decon pit was that the intact spent fuel currently being loaded was not adversely affected during the short time the helium was disconnected. Since the current UFSAR allowed for the use of air in the canister for blowdown and the fuel currently being loaded at Fort Calhoun had a low heat load (i.e.

low kW), the licensee concluded that no potential damage was occurring to the fuel assemblies due to possible contact with air.

n. Horizontal Storage Module Maintenance

On an annual basis, the licensee performed a walk-down and visual inspection of the ISFSI pad and accessible HSM surfaces for visual signs of degradation or structural cracking. Work Order #00409497, dated August 16, 2011, documented that for the 2011 inspection, no new degradation or cracking was observed of the ISFSI pad or HSM surfaces since the 2009 inspection. Condition Report 2009-4648 had documented various defects observed during the 2009 walk-down performed on September 30, 2009. The observations included: the concrete finish layer on one section of the ISFSI pad was flaking off; two small cracks were noted at the ground vents between HSM #5 and HSM #3; one small crack was noted at the ground vent between HSM #4 and HSM #6; one small crack was noted at the corner of the ground vent between HSM #8 and HSM #10; and two cracks were noted in the section of the HSM pad east of the HSMs. All documented defects were evaluated to be superficial surface issues which did not affect the function of the HSM or ISFSI pad.

o. Decommission Funding Plan

Federal Register Notice 76FR35512, dated June 17, 2011, included a new rulemaking requirement that affected Part 72 licensees. The Federal Register documented a change to 72.30(b) which required Part 72 licensees to submit to the NRC for review and approval an ISFSI decommissioning funding plan. The final rule made changes to the financial assurance requirements for Part 72 licensees to provide greater consistency with similar decommissioning requirements in the 10 CFR Part 50 regulations. Financial assurances are financial arrangements provided by the licensee to ensure funds for decommissioning will be available when needed. The effective date of the new rule was December 17, 2012. The new rule required licensees to submit a decommissioning funding plan to the NRC by the effective date of the rule. Contrary to this requirement, Fort Calhoun had not submitted their ISFSI decommissioning funding plan by December 17, 2012. The NRC has determined that this is a Severity Level IV violation of 10 CFR 72.30(b). Since the licensee entered the issue into their corrective action program, and because the issue was not a repetitive violation or willful, this violation is being treated as a NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. Subsequent to this inspection, the licensee submitted the ISFSI Decommissioning Funding Report on March 6, 2013 (Adams Accession No. ML13067A319).

1.3 Conclusions

The Fort Calhoun Quality Department had included ISFSI related activities in their audit and surveillance program. Quality assurance audits and surveillances performed in 2011 reviewed activities and documentation associated with the ISFSI program. No significant conditions adverse to quality were found.

Radiological conditions at the ISFSI were evaluated which included conducting a survey of the area around the ISFSI and the HSMs. Radiation levels recorded on the dosimeters around the ISFSI pad showed low radiation levels, as expected for an ISFSI with ten casks. Offsite monitoring data from the 2009, 2010, and 2011 annual environmental reports documented that there were no offsite radiological impacts attributable to ISFSI operations.

Documents and records related to the 2006 and 2009 ISFSI cask loading campaigns were reviewed. Information included personnel EAD records, canister survey records, and estimated neutron doses to workers during cask loading activities. Worker doses to load a cask have continued to improve, with the last campaign averaging 0.084 person-rem/cask.

Technical Specification 1.3.2 temperature monitoring requirements for the HSMs were performed daily as required.

Required records were maintained that described the specific fuel parameters for the spent fuel stored in each of the licensee's loaded casks.

Selected condition reports were reviewed for the period 2009 thru 2012. A wide range of issues had been identified and resolved. Resolution of the issues was appropriate for the safety significance of the issue. No adverse trends were identified during the review.

The ISFSI pad base mat was built at an elevation of 1009 feet 10 inches. This is higher than the Army Corps of Engineer's calculated maximum probable flood height without an upstream dam failure of 1009 feet 4 inches. During 2011, flood waters reached 1006 feet 11 inches. Though the ISFSI pad was not flooded, it was surrounded by water. After the 2011 event, Fort Calhoun began the process to re-evaluate the maximum possible flood level at the site, including a possible upstream dam failure, and the affect on the stored fuel and HSMs.

The licensee upgraded the existing 75 ton single failure proof auxiliary building HE-2 crane to a rated load capacity of 106 tons to support the 2009 cask loading campaign. An NRC in-office review of the crane upgrade documentation was performed after the onsite inspection was completed. NRC inspectors reviewed documentation related to the replaced components, the basis for the crane's designation as single-failure proof, the 125 percent and 100 percent crane load tests, the 200 percent hook load test, the 300 percent lift yoke load test, the newly replaced wire ropes' breaking strengths, the crane's revised operating procedures, and the calculated maximum weight of the loaded cask that will be lifted by the crane. No issues related to the trolley were identified during the review. Concurrent with this inspection, the NRC's IMC 350 inspection team was in the process of reviewing the seismic methodology that had been used for various structures at Fort Calhoun. This review included the adequacy of the auxiliary building crane support structures and is documented in Inspection Report 50-285/2013-008.

On an annual basis, the licensee performed a visual inspection of the ISFSI pad and accessible HSMs' surfaces for signs of degradation or structural cracking. No new degradation or cracking has been observed at the ISFSI pad or on the HSM surfaces since 2009 when a number of small cracks were documented. All documented defects were evaluated as superficial surface issues which would not affect the function of the HSMs or ISFSI pad.

Each holder of a license under 10 CFR Part 72 must submit a decommissioning funding plan for NRC review and approval in accordance with 10 CFR 72.30(b). Per Federal Register Notice 35573 dated June 17, 2011, this new rule took effect on December 17,

2012. Contrary to this requirement, Fort Calhoun had not submitted their ISFSI decommissioning funding plan by December 17, 2012. The NRC has determined that this is a Severity Level IV Violation of 10 CFR 72.30(b). Since the licensee entered the issue into their corrective action program, and because the issue was not a repetitive violation or willful, this violation is being treated as a NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. Subsequent to this inspection, the licensee submitted the ISFSI Decommissioning Funding Report on March 6, 2013.

2 Review of 10 CFR 72.212(b) Evaluations at Operating Plants (60856.1)

2.1 Inspection Scope

The 10 CFR 72.212 Evaluation Report was reviewed to verify site characteristics were still bounded by the Transnuclear NUHOMS design basis.

2.2 Observations and Findings

The Fort Calhoun's 10 CFR 72.212 Evaluation Report was currently Revision 1 issued June of 2009. The revision evaluated the use of the OS197H transfer cask, the up-rating of the auxiliary building HE-2 crane to 106 tons, and the 2009 loading campaign operations using Certificate of Compliance 1004, Amendment 9. Two new appendices were added to the 72.212 report. Appendix B "10CFR72.212 Evaluation for the 2009 Loading of Six NUHOMS-32PT DSCs Using the OS197 Transfer Cask" and Appendix C "Certificate of Compliance Evaluation" provided information related to the 2009 loading campaign and evaluated the loading activities to verify they were in compliance with the NUHOMS Certificate of Compliance, Amendment 9 and the NUHOMS UFSAR, Revision 10. Appendix B, Table 9.1 "ISFSI Activities Evaluated Under 10CFR50.59" listed the various engineering changes that had been performed to address the changes with the auxiliary building crane and transfer cask along with a number of other topics ranging from security, criticality control, and procedure changes.

2.3 Conclusions

The licensee was maintaining the 10 CFR 72.212 Evaluation Report current as required. No changes to the 10 CFR 72.212 report had been made since the last NRC inspection in 2009.

3 Review of 10 CFR 72.48 Evaluations (60857)

3.1 Inspection Scope

The licensee's 10 CFR 72.48 screenings and evaluations since the 2009 NRC inspection were reviewed to determine compliance with regulatory requirements.

3.2 Observations and Findings

A list of modifications to the ISFSI program and changes to the auxiliary building HE-2 crane was provided by the licensee. Three 10 CFR 72.48 screenings and two 10 CFR 50.59 screenings for the crane were performed since the last NRC inspection in August 2009. The licensee utilized Procedure NOD-QP-3 "10 CFR 50.59 and 10 CFR 72.48 Reviews," Revision 33 to perform the 10 CFR 72.48 safety screenings or evaluations.

None of the screenings required a full 10 CFR 72.48 or 10 CFR 50.59 safety evaluation. The issues discussed in the screenings included the following: a scaffold being constructed near the HSMs that allowed the operators to view the outlet vents more conveniently; temporary lighting towers being staged within the fenced ISFSI boundary; an engineering change to modify the platform that was placed around the transfer cask; the replacement of one emergency drum brake actuator on the auxiliary building HE-2 crane; and an engineering change to modify a guardrail on the HE-2 crane. All screenings were determined to be adequately evaluated.

3.3 Conclusions

All required safety screenings had been performed in accordance with procedures and 10 CFR 72.48 requirements. All screenings reviewed were determined to be adequately evaluated. No safety evaluations were performed since the last inspection in 2009.

4 Exit Meeting

The inspector reviewed the scope and findings of the inspection during an exit conducted on April 17, 2013.

ATTACHMENT 1:

SUPPLEMENTAL INSPECTION INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee Personnel

M. Braden, Licensing Engineer, Regulatory Assurance
T. Bussey, Reactor Engineering
E. Durboraw, Radiation Protection
K. Erdman, Supervisor, Programs Engineering
D. Little, Radiation Health Specialist, Dosimetry
P. Turner, Jr., ISFSI Program Engineer, Nuclear Engineering Division

INSPECTION PROCEDURES USED

IP 60855.1	Operations of an ISFSIs at Operating Plants
IP 60856.1	Review of 10 CFR 72.212(b) Evaluations at Operating Plants
IP 60857	Review of 10 CFR 72.48 Evaluations

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

72-54/1301-01	NCV	Failure to submit 10 CFR 72.30(b) ISFSI Decommissioning Funding Report by December 17, 2012.

Discussed

None

Closed

72-54/1301-01	NCV	Failure to submit 10 CFR 72.30(b) ISFSI Decommissioning Funding Report by December 17, 2012.

LIST OF ACRONYMS

ADAMS	Agencywide Document Access and Management System
ALARA	As low as reasonably achievable
CE	Combustion Engineering
CFR	Code of Federal Regulations
CoC	Certificate of Compliance
CR	Condition report
DLR	Dose of legal record
DSC	Dry shielded canister
EAD	Electronic alarming dosimeter
F	Fahrenheit
FCS	Fort Calhoun Station
HSM	Horizontal Storage Module
HVAC	Heating, Ventilation, and Air Conditioning
IMC	Inspection Manual Chapter
IP	Inspection procedure
ISFSI	Independent Spent Fuel Storage Installation
kW	Kilo-watt
mR	MilliRoentgen
micro(μ)R	MicroRoentgen
micro(μ)rem	MicroRoentgen equivalent man
MPC	Multipurpose canister
mrem	MilliRoentgen equivalent man
MSL	Mean Sea Level
MWD/MTU	Megawatt days/metric ton uranium
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
NUHOMS	Nuclear Horizontal Modular Storage
OPPD	Omaha Public Power District
QA	Quality Assurance
TLD	Thermo-luminescent dosimeter
TN	Transnuclear
UFSAR	Updated Final Safety Analysis Report (Transnuclear)
USAR	Updated Safety Analysis Report (Fort Calhoun Station)
WO	Work Order

ATTACHMENT 2:

LOADED CASKS AT THE FORT CALHOUN NUCLEAR STATION ISFSI

ORDER	DSC SERIAL No.	NUHOM No.	DATE ON PAD	HEAT LOAD (kW)	BURNUP MWd/MTU (max)	MAXIMUM FUEL ENRICHMENT %	PERSON-REM DOSE
1	FCS32PT-S100-A-HZ02	DFS-HSM-002	07/29/2006	9.88	39578	3.511	0.534
2	FCS32PT-S100-A-HZ01	DFS-HSM-001	08/04/2006	10.10	41115	3.511	0.380
3	FCS32PT-S100-A-HZ06	DFS-HSM-003	08/10/2006	9.37	42047	3.037	0.201
4	FCS32PT-S100-A-HZ08	DFS-HSM-004	08/17/2006	10.52	42049	3.511	0.247
5	FCS32PT-S100-A-HZ04	DFS-HSM-006	07/29/2009	10.60	42251	3.509	0.108
6	FCS32PT-S100-A-HZ07	DFS-HSM-008	08/06/2009	11.13	41120	3.601	0.073
7	FCS32PT-S100-A-HZ10	DFS-HSM-010	08/11/2009	10.73	42251	3.511	0.082
8	FCS32PT-S100-A-HZ03	DFS-HSM-005	08/16/2009	11.48	44699	3.599	0.067
9	FCS32PT-S100-A-HZ05	DFS-HSM-009	08/22/2009	10.22	44834	3.599	0.111
10	FCS32PT-S100-A-HZ09	DFS-HSM-007	08/29/2009	10.35	42251	3.595	0.064

NOTES:
- Heat load (kW) is the sum of the heat load values for all spent fuel assemblies in the cask
- Burn-up is the value for the spent fuel assembly with the highest individual discharge burn-up
- Fuel enrichment is the spent fuel assembly with the highest average “initial” enrichment per cent of U-235

HSMs # 1-4 were loaded to NUHOMS CoC 1004, License Amendment 8, and UFSAR, Revision 9
HSMs # 5-10 were loaded to NUHOMS CoC 1004, License Amendment 9, and UFSAR, Revision 10

Casks are maintained to the Amendment and Revision they were loaded to.