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

April 25, 2014

Mr. M. E. Reddemann
Chief Executive Officer
Energy Northwest
P.O. Box 968, Mail Drop 1023
Richland, WA 99352-0968

SUBJECT: COLUMBIA GENERATING STATION AND INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) INSPECTION REPORT 05000397/2014007 AND 07200035/2014001

Dear Mr. Reddemann:

This letter refers to a routine inspection conducted on March 24 - 27, 2014, of the dry cask storage activities associated with your Independent Spent Fuel Storage Installation (ISFSI). The enclosed inspection report documents the inspection results which were discussed on March 27, 2014, with you and members of your staff.

The inspection 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 inspection reviewed compliance with the requirements specified in the Technical Specifications associated with Holtec International HI-STORM 100 Certificate of Compliance 1014, the HI-STORM 100 Final Safety Analysis Report (FSAR), and Title 10 of the Code of Federal Regulations (CFR) Part 72, Part 50, and Part 20. Within these areas, the inspection included a review of radiation safety, cask thermal monitoring, quality assurance, your corrective action program, safety evaluations, observations of cask loading activities, and how you addressed industry issues that affected your facility's programs. Also reviewed were changes made to your ISFSI program since the last routine ISFSI inspection that was conducted by the U.S. Nuclear Regulatory Commission (NRC). No violations of NRC regulations were identified.

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 Agencywide Document Access Management 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 817-200-1287 or Mr. Lee Brookhart at 817-200-1549.

Sincerely,

/RA/

Linda Howell, Deputy Director
Division of Nuclear Materials Safety

Dockets No.: 05000397, 07200035
Licenses No.: NPF-21

Enclosure:
Inspection Report 05000397/2014007
and 07200035/2014001

w/attachments:
1. Supplemental Information
2. Loaded Casks at the Columbia ISFSI

cc w/attachments: Electronic Distribution to Columbia Generating Station

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See next page

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FINAL: R:_Reactors_COL\2014\CGS 2014007-ISFSI-LEB.docx

SUNSI Rev Compl.	**X Yes** ☐ No	ADAMS	**X Yes** ☐ No	Reviewer Initials	LEB
Publicly Available	**X Yes** ☐ No	Sensitive	☐ Yes **X No**	Sens. Type Initials	

RIV:DNMS/RSFS	RIV:DNMS/RSFS	R-IV/DD:DNMS		
LEBrookhart	EJSimpson	LLHowell		
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04/25/14	04/25/14	04/25/14		

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Letter to M. E. Reddemann from L. Howell, dated April 25, 2014.

SUBJECT: COLUMBIA GENERATING STATION AND INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) INSPECTION REPORT 05000397/2014007 AND 07200035/2014001

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	050-00397; 072-00035
Licenses:	NPF-21
Report Nos.:	05000397/2014007; 07200035/2014001
Licensee:	Energy Northwest
Facility:	Columbia Generating Station and Independent Spent Fuel Storage Installation (ISFSI)
Location:	P.O. Box 968 Richland, WA 99352
Dates:	March 24–27, 2014
Inspector:	Lee Brookhart, Senior ISFSI Inspector Repository & Spent Fuel Safety Branch
Accompanying Personnel:	Eric Simpson, Health Physicist, Inspector-in-Training, Repository & Spent Fuel Safety Branch
Approved By:	Linda Howell, Deputy Director Division of Nuclear Materials Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000397/2014007; and 07200035/2014001; 03/24–03/27/2014; Columbia Generating Station and Independent Spent Fuel Storage Installation; Routine ISFSI Inspection Report

The report covers an announced inspection by one regional inspector and one inspector-in-training. No findings or violations associated with Nuclear Regulatory Commission (NRC) regulations were identified. The significance of any Part 50 findings are indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process." The cross-cutting aspect is determined using IMC 0310, "Components Within the Cross-Cutting Areas." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after the 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. In accordance with the NRC Enforcement Policy, all of the Part 72 ISFSI inspection findings follow the traditional enforcement process and are not dispositioned through the Reactor Oversight Process or the Significance Determination Process.

A. NRC-Identified Findings and Self-Revealing Findings

No findings were identified.

B. Licensee-Identified Violations

None.

PLANT AND ISFSI STATUS

Columbia Generating Station's (CGS) Independent Spent Fuel Storage Installation (ISFSI) stored thirty loaded Holtec HI-STORM 100S (243) casks at the time of the routine inspection. CGS was in the middle of a nine cask loading campaign and was performing the loading operations associated with cask number 31at the time of the routine inspection. CGS utilized a general Part 72 license in accordance with the Holtec HI-STORM 100 System, approved under Certificate of Compliance 1014, License Amendment 2 and Final Safety Analysis Report (FSAR), Revision 4. The version of the Holtec systems used at CGS included the MPC-68, a 68 fuel bundle multi-purpose canister (MPC), designed to hold 68 boiling water reactor (BWR) fuel assemblies. The ISFSI at CGS consisted of two pads that were approximately 147 feet long by 30 feet wide and were designed to accommodate 18 storage casks each. At the end of the current loading campaign, CGS will have filled the two pads and will begin construction in the coming years to build additional pads to accommodate more casks. The storage casks were located outside the Part 50 facility's protected area (PA) within its own PA. A tour of the ISFSI pad and adjacent area found the loaded storage casks to be in good physical condition. A review of the environmental monitoring program demonstrated that radiological exposures to offsite locations due to the ISFSI were low and within regulatory limits.

The reactor at CGS is a BWR with a core containing 764 fuel assemblies. Refueling outages are performed approximately every 24 months. The spent fuel pool capacity at CGS is 2,658 spaces. After the current loading campaign is complete, spent fuel pool will contain 943 spaces that are open and available for use.

REPORT DETAILS

4. OTHER ACTIVITIES

4OA5 Other Activities

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

a. Inspection Scope

(1) Quality Assurance Audits and Surveillances

An onsite review of Quality Assurance (QA) audit reports and surveillances related to dry cask storage activities at the CGS ISFSI was performed. Since the last inspection in July 2012, the Energy Northwest had not issued any audit reports of ISFSI related activities. The plant's Quality Services organization had been performing biennial audits of ISFSI related activities. Those audits typically reviewed areas including training and qualifications, procedures, design control, procurement documentation, and ISFSI operations. The timing of the current NRC inspection was between auditing periods. The last audit was conducted April 12–May 3, 2012 and was reviewed during the last inspection. The NRC inspectors were onsite March 24–27, 2014, before the next biennial audit report will be performed or issued.

CGS had performed one vendor QA audit at the Areva Federal Services LLC (AFS) facility located in Richland, WA. That audit was performed to evaluate the effectiveness of the AFS quality program as it related to providing helium leak testing services to support the Energy Northwest dry cask storage program. One deficiency was noted during the audit. However the audit team determined that the finding was administrative in nature and was not considered significant.

(2) Radiological Conditions Related to Stored Casks

The CGS ISFSI is located approximately 500 meters NNW of the reactor building. It consisted of two concrete pads, each 30 feet wide by 147 feet long with the capacity to hold 18 HI-STORM 100S spent fuel storage casks. The two ISFSI pads were protected on all sides by three fences. The ISFSI pads were positioned just left of center in the inner fence. No flammable or combustible materials were observed inside any of the fenced ISFSI areas. Thirty out of a possible capacity of 36 casks were loaded with spent fuel and placed on the ISFSI at the time of the inspection. Pad 1 was fully loaded with 18 HI-STORM 100S casks. Pad 2 was partially loaded with 12 casks, but was scheduled to reach full capacity by the end of the current loading campaign. The inspectors found the 30 loaded HI-STORM 100S casks to be in good physical condition.

Radiological conditions at the CGS ISFSI were determined from onsite ISFSI pad measurements. Inspectors reviewed the most recent quarterly radiological survey and quarterly ISFSI thermoluminescent dosimeter (TLD) monitoring results. A recent radiological survey was provided to the NRC inspectors prior to their arrival onsite. For the ISFSI inspection, a radiation protection (RP) technician accompanied the NRC inspectors through the PA fencing to perform a walk-down of the ISFSI pad. A radiological survey was performed by the RP technician with a tissue-equivalent organic scintillation detector for gamma radiation. This detector provided dose-equivalent readings in micro-rem per hour (μrem/h). The RP technician took radiological measurements as directed by the NRC to confirm the levels reported on the most recent ISFSI survey. Survey measurements were taken around the perimeter of the ISFSI pad, at selected storage cask locations, and at environmental TLD monitoring locations on the ISFSI PA fence.

General area background readings before reaching the ISFSI were 9 μrem/h. Approaching the ISFSI pad from the south, the dose rate approached 20 μrem/h outside the first of three fences. Proceeding into the PA, readings obtained along the innermost fence were 50–130 μrem/h, with the highest readings along the broad sides of the ISFSI pads. At the outer edges and corners of the ISFSI pads, the readings ranged from 100–500 μrem/h on northernmost Pad 1 and 400–500 μrem/h on the southern ISFSI Pad 2. At the center of the ISFSI pads the readings were on the order of 1 milli-rem per hour (mrem/h). The highest readings measured on the ISFSI pad were obtained at lower HI-STORM 100S vent locations. The inspected lower vent openings measured from 1.3–4.0 mrem/h. Those measurements confirmed earlier site survey records. Neutron measurements were not made during the ISFSI inspection. The radiological conditions of the ISFSI were as expected for the age, burn-up levels, and heat-load of the spent fuel currently loaded.

(3) Environmental Radiological Monitoring Program

The primary purpose of the Energy Northwest Radiological Environmental Monitoring Program is to evaluate the radiological impact that CGS operation may have on the environment. The environmental monitoring program was focused on airborne (gaseous and particulate), liquid effluent, and direct radiation monitoring onsite, at the site boundary, and at offsite locations. By design, there were no airborne or liquid effluent releases from the ISFSI, which is located in the site's NNW sector, approximately 500 meters from the reactor building.

NRC inspectors reviewed the CGS 2012 Annual Radiological Environmental Operating Report (ML13137A108), issued April 30, 2013, and ISFSI direct radiation monitoring data from 2012 and 2013. The operating report also included the direct radiation monitoring data for the ISFSI. Twelve site TLD monitoring locations were directly influenced by the ISFSI, ten of which were directly positioned on the ISFSI isolation fence. Collectively, along with other monitoring locations, the ISFSI TLDs were described as being part of a collection of TLDs that were deployed to track specific site activities, but not used to assess offsite dose.

Figure 1 show the locations of the ten TLD stations (STD.123–STD.138A) located around the ISFSI fence. This arrangement of TLDs, in conjunction with the radiological surveys conducted by the CGS Radiation Protection Department, served as the radiological monitoring program for the ISFSI and satisfied the monitoring requirements of 10 CFR 72.44(d)(2).

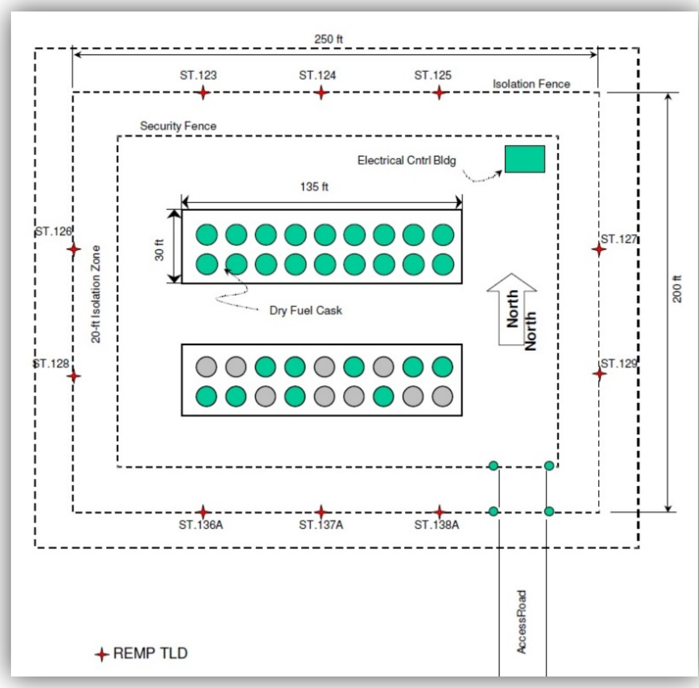

Figure 1 Columbia Generating Station ISFSI

The ISFSI TLD monitoring locations are identified in documents provided to the NRC as location numbers 121–129 and 136A–138A. Table 1, below, displays the yearly monitoring results for the monitoring stations positioned directly on the ISFSI fence: 123–129 and 136A–138A.

Table 1, Direct Monitoring Results for Columbia ISFSI in mR[1]/yr

TLD#	Station and Location	2012	2013
123	ISFSI North Fence	457	432
124	ISFSI North Fence	539	499
125	ISFSI North Fence	403	373
126	ISFSI West Fence	405	377
127	ISFSI East Fence	306	288
128	ISFSI West Fence	504	471
129	ISFSI East Fence	396	367
136A	ISFSI South Fence	578	530
137A	ISFSI South Fence	635	586
138A	ISFSI South Fence	509	472
Control	WSW sector, 28 miles offsite	81	N/A

The TLD monitoring results are reported in units of exposure, milliroentgen per year (mR/yr), and not in dose-equivalent (rem)[1]. For the period beginning in January 2012 and ending on December 31, 2013, the annual TLD gamma exposure rates at the ten ISFSI monitoring locations ranged from 288 to 635 mR/yr, the highest being from location 137A in 2012. Those values converted to exposure rates between 33 and 72 μR/h. The exposure rates for all ten monitoring locations trended downward from 2012 to 2013. This was a continuation of a downward trend that began at the end of the last CGS spent fuel loading campaign in 2008. Those levels may increase this year with the planned loading of nine additional casks.

A site survey performed at the direction of NRC inspectors confirmed the most recent TLD results (see Table 2). It should be noted that three additional spent fuel casks were on the ISFSI pad at the time of the survey that were not present during the 2012 or 2013 monitoring periods. Also, the survey meter used during this inspection provided dose-equivalent readings in micro-rem per hour (μrem/h).

Table 2, ISFSI Fence TLD Location Survey Results

TLD #	Survey Results
123	50 μrem/h
124	50 μrem/h
125	40 μrem/h
126	50 μrem/h
127	40 μrem/h
128	70 μrem/h
129	50 μrem/h
136A	60 μrem/h
137A	80 μrem/h
138A	60 μrem/h

At the time of the inspection, CGS was in the midst of its fourth ISFSI loading campaign, the first since the last loading campaign in 2008 that brought the number of casks on the pad to 27. By the end of the NRC inspection, four additional casks

[1] For the purposes of making comparisons between NRC regulations based on dose-equivalent and measurements made in roentgens, it may be assumed that one roentgen equals one rem (http://www.nrc.gov/about-nrc/radiation/protects-you/hppos/qa96.html).

were loaded and three had been placed onto Pad 2. This brought the total number on both pads to 30. Five casks remained to be loaded and six remained to be placed on the pad.

The CGS site boundary is considered to be a circle with a radius of 1.2 miles, centered on the reactor building. The closest environmental monitoring program direct radiation locations to the ISFSI were at site boundary locations identified with Station Numbers 16, 17, 18, and 49. Those locations are in the WNW, NNW, N, and NW sectors, respectively, and at distances of 1.21, 1.19, 1.16, and 1.19 miles, respectively, from the plant. Table 3, below, lists the 2012 exposure rates presented in the environmental operating report for those site boundary monitoring locations.

Table 3, Exposure Rates for REMP Locations Close to CGS ISFSI in mR/yr

Station ID	Sector	Distance from Plant	2012 Exposure Rate
16	WNW	1.21 miles	84
17	NNW	1.19 miles	87
18	N	1.16 miles	85
49	NW	1.19 miles	85
Control	WSW	28.35 miles	81

Average background (control) dose measurements for 2012 were 81mR/yr. Correcting for background shows only a slight increase in ambient radioactivity at the site boundary monitoring locations closest to the ISFSI, between 3 and 6 mR/yr. These values were actually lower than the measured pre-operational values for those locations and were below the 10 CFR 72.104 (a) ISFSI dose requirement not to exceed 25 mrem per year. Offsite doses due to the CGS ISFSI were within regulatory limits.

(4) Records Related to Fuel Stored in the Casks

A site review of dry fuel storage records at CGS was performed to determine if adequate descriptions of spent fuel was documented as a permanent record as required by 10 CFR 72.212(b)(12). The spent fuel contents of every MPC loaded at CGS and stored in a HI-STORM 100S overpack was recorded in DIC 1814.5, Cask Loading Plan. The Cask Loading Plans included MPC fuel assembly loading maps, fuel assembly identification information, fuel inspection records, and other fuel assembly information. The forms also contained Technical Specification related information about each fuel assembly, including assembly decay heat (kW), cooling time (years), initial Uranium 235 (U-235) enrichment (%), and burn-up values (MWd/MTU). A complete set of forms was reviewed for three random casks from past loading campaigns and three casks from the current campaign. CGS was in compliance all applicable regulations related to fuel storage records.

(5) Cask Loading Observations

Various loading activities were observed by the NRC inspectors during the course of the routine inspection. CGS was in the process of loading canister #31 at the time of the inspection. The NRC inspectors observed the fuel movement activities to place spent fuel assemblies into canister #31. The licensee's staff was experienced in moving the spent fuel assemblies and was proficient in locating the correct assembly, verifying the assembly, moving the assembly from the rack to the canister, and inserting the assembly into the assigned canister slot. The time from grappling the assembly, placing the assembly into the assigned canister slot, and returning to the next assigned spent fuel assembly was approximately five to six minutes.

Selected welding and non-destructive examination activities were observed during the loading associated with canister #31. An automatic welding process was used to weld the canister lid. The automated welding machine utilized two weld heads to weld the lid to shell weld. The welders operated the equipment remotely in a low dose rate area. Hydrogen monitoring was performed during the welding of the root weld and subsequent passes. The licensee monitored for combustible gas every ten minutes utilizing a hand-held monitor around the open weld area or the MPC vent port after the root pass had been completed. Additionally, the NRC inspectors observed the non-destructive dye penetrant exams conducted on the various welds. No weld indications were identified from the various non-destructive examinations.

Other activities that were observed by NRC inspectors during the loading of canister #31 included: lifting the HI-TRAC transfer cask and fully loaded MPC out of the spent fuel pool, vacuum drying the canister, helium backfill of the canister, welding of the vent and drain port covers, and transfer of the canister from the HI-TRAC to the HI-STORM.

(6) Technical Specification 3.1.2, Cask Temperature Monitoring

Technical Specification 3.1.2 required either a daily inspection of the inlet and outlet vents for blockage or daily verification that the temperature difference (delta T) between the HI-STORM outlet temperature and ISFSI ambient temperature was ≤ 126 degrees F. All HI-STORM casks at CGS were equipped with remote temperature monitoring equipment which provided daily temperature readings that were used to verify compliance with the technical specification limit. If the temperature monitoring equipment malfunctioned, the licensee's procedure required a daily visual inspection of the vents for the affected cask(s). The temperature surveillances and/or vent inspections were performed using Procedure OSP-SFS-D101 "Spent Fuel Storage Cask Heat Removal System Daily Checks," Revision 12. The NRC inspectors requested daily temperature monitoring system data for the months of December 2012 and June 2013. This data was reviewed to verify that the 126 degree F delta T limit had not been exceeded.

Procedure OSP-SFS-D101 required the user to obtain the daily temperature printout for the casks and attach the printout to the surveillance procedure for document retention. Step 7.3 of the procedure required the user to either verify the change in temperature between the outlets and ambient temperature was less than 114 degree F or verify the HI-STORM's inlet and outlet air ducts were free of blockage if one of the temperature detectors were malfunctioning. The 114 degree F was an administrative limit to initiate corrective actions before the 126 degree F technical specification limit was reached. Step 7.2.4 required the user to submit a work request for repair of temperature detectors that were found to be in a fault status.

The HI-STORM casks stored on the ISFSI pad were equipped with a temperature detector just outside each of the four outlet vents on the cask. The computed difference between the ambient air and the average of the four temperature detectors (delta T) was recorded for each cask on a temperature printout and was attached to the back of the surveillance procedure for each day. For all the days of the selected months reviewed the licensee had either performed the temperature difference verification or performed a vent surveillance check in accordance with the technical specification. No documentation reviewed showed a cask exceeding the 114 degree F delta T limit or the vents being blocked.

(7) Corrective Action Program

A list of condition reports issued since the last NRC inspection in July of 2012 was provided by the licensee for the fuel building overhead crane and the ISFSI. Issues were processed in accordance with Procedure SWP-CAP-01 "Corrective Action Program," Revision 29. When a problem was identified the licensee documented the issue as an Action Request type Condition Report (AR-CR) in the licensee's corrective action program.

Of the list of AR-CRs provided relating to the ISFSI and the fuel building overhead crane, approximately 24 AR-CRs were selected by the NRC inspectors for further review. The AR-CRs related to a number of different topics including: evaluation of the concrete pad hardening over time, cask thermal couple malfunctions, a new evaluation regarding the stack-up seismic stability, servicing of the helium leak detector, use of tungsten shielding on the MPC, MPCs exceeding hardness requirements, re-evaluation of the cask lifting devices, and crane maintenance issues.

The AR-CRs reviewed were well documented and properly categorized based on the significance of the issue. The corrective actions taken were appropriate for the situations. Based on the level of detail of the corrective action reports, the licensee demonstrated a high attention to detail in regard to the maintenance and operation of their ISFSI program and the fuel building overhead crane. No NRC concerns were identified related to the condition reports reviewed.

(8) Self-Assessment Report

In January of 2014, CGS conducted a self-assessment of the ISFSI Program. The ISFSI self-assessment report AR-SA #299169 dated January 29, 2014, was completed as a readiness review for the nine cask loading campaign that was set to begin March 3, 2014. Two standard deficiencies were identified through the assessment.

The first deficiency identified that there were no time limits provided in the ISFSI loading procedures regarding closure of the mating device (MD) gate adapter drawer with a loaded canister in the HI-STORM overpack. Since normal convection cooling of the canister is inhibited with closure of the MD drawer, heat-up of the canister and associated spent fuel inside the canister could exceed allowable temperature limits. Per a discussion with Holtec and CGS, a specific analysis for the closed MD drawer on a loaded over pack had not been performed for CGS nor had it been performed as part of vendor licensing basis documents (the general license CoC, TS, or FSAR). A review of CGS's ISFSI loading procedures concluded that an allowed time limit to be in this configuration was not specified in their procedures. CGS placed this issue in their CAP as AR-CR 00300588 dated January 7, 2014.

Once a canister is lowered through the MD gate adapter, the MD drawer is closed and the inflatable bags are inflated to allow bolt-up of the transfer cask bottom lid. The transfer cask is then unbolted from the MD gate adapter and lifted back to the refuel floor. The MD drawer is closed and partially opened for various periods of time while bolted to an overpack with the loaded canister to remove rigging equipment. Holtec Technical Specifications and FSAR are silent on this loading operation and configuration for time limits. With the MD drawer bolted on the top of the HI-STORM in the closed position it would restrict airflow because there are no outlet vents associated with the MD like that on a HI-STORM lid.

CGS requested Holtec to perform a site specific analysis to determine the duration for the mating device to remain atop of the loaded HI-STORM in the closed configuration and in a half-open position. Holtec provided CGS with Response to Request for Technical Information (RRTI) 2028-004 Revision 1 on February 19, 2014. The evaluation stated while the mating device is atop a HI-STORM overpack with the drawer closed, the normal cooling airflow through the overpack will be interrupted. During this short-term condition the canister and overpack temperatures will rise, eventually exceeding component temperature limits. A transient evaluation of a HI-STORM overpack containing a MPC-68 had been performed for this scenario. The evaluation bounded the CGS decay heats (all below 22 kW) using the design basis decay heat of 28.19 kW and bounded the presence of a mating device by enforcing a zero flow condition at the outlet vents. The evaluation indicated that in excess of 24-hours existed before any licensing-basis component temperature limit would be exceeded. Based on that evaluation, the mating device must be opened or removed within 24-hours of closure to ensure that the short-term temperature limits of FSAR Rev. 4 Table 2.2.3 are not exceeded. The evaluation further stated that CGS does not perform stack up in a recessed vault, like a cask transfer facility. The HI-STORM is located atop of a low profile transporter on the flat surface of the truck bay. Therefore, keeping the mating device open 50% or greater will keep the MPC/HI-STORM system within the normal temperature range and allow for full air cooling of the MPC. No time limit was required for a MPC/HI-STORM in the 50% open configuration.

CGS placed a Caution note into loading Procedure 6.6.5, "Movement and Transfer Operations of HI-TRAC and HI-STORM in the Reactor Building," Revision 14, to limit the closure time of the MD drawer to under 24-hours and specified the minimum percent to open the drawer to restore normal cooling. A review was completed of the 27 previously loaded canisters which determined that none had exceeded the 24-hour time limitation.

The second deficiency related to an error in the Holtec supplied equation for calculating the minimum allowable mass of helium to be backfilled into a MPC with measurement uncertainty to ensure compliance with the Technical Specifications. CGS placed this issue into the CAP program as AR-CR #00300522 dated January 6, 2014. The self-assessment identified that Holtec had used some non-conservative measurements of uncertainty to obtain their calculation for users to meet the helium backfill technical specification. The evaluation identified that if a user was to place the minimum allowable helium in the canister per the Holtec equation, it may lead to the licensee not meeting the Technical Specification if the calculated measurement uncertainty had occurred. The result of not using truly conservative measurements of uncertainty was that the helium backfill amount was off by a very small fraction. The analysis that supported this conclusion was documented in Report # 20350-R-001 “Discussion of Minimum Required MPC Backfill Helium Mass,” dated February 12, 2014.

CGS reviewed the final helium pressures that were placed in the 27 previously loaded canisters. For all canisters, more than the minimum target mass of helium was placed into the MPCs. All canisters were still within the correct tolerance band, and therefore, all canisters were in compliance with the helium backfill technical specification. CGS revised their loading procedure Plant Procedure Manual 6.6.7 “MPC Processing,” Revision 25 to contain the correct equation for helium backfill.

(9) Preparation of Loading Activities

The inspectors requested documentation related to maintenance of the fuel building cask handling crane, calibration of various gauges associated with the loading activities, and the annual maintenance of the special lifting devices.

Documents were reviewed that demonstrated that the fuel building crane was inspected on an annual basis in accordance with American Society of Mechanical Engineers (ASME) B30.2 prior to the 2014 loading campaign. CGS utilized Procedure 10.4.5 “Reactor and Turbine Building Overhead Traveling Crane Inspection, Maintenance, and Testing,” Revision 23 and Work Order (WO) 02036654 to perform the annual maintenance in October of 2013.

The annual maintenance as required by American National Standards Institute (ANSI) N14.6 for special lifting devices was completed for the following special lifting devices: the lift yoke, HI-STORM brackets, and the HI-TRAC lifting trunnions. Documentation reviewed included WO 02036364, WO 02037517, WO 02052336, and associated non-destructive examination (NDE) documentation associated with the testing. All equipment passed the magnetic particle, liquid penetrant, and dimensional testing.

Calibration documentation of the hydrostatic pressure gages and the vacuum drying gage was reviewed to ensure the equipment had been properly calibrated prior to the loading operations. All calibration certificates reviewed demonstrated that the gages were properly calibrated prior to the loading campaign.

b. Findings

No findings were identified.

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

a. Inspection Scope

The 10 CFR 72.212 Evaluation Report was reviewed to verify site characteristics were still bounded by the Holtec HI-STORM 100 cask system's design basis.

CGS's 10 CFR 72.212 Evaluation Report at the time of the inspection was Revision 7, dated November, 2012. One revision had been performed to the 72.212 Evaluation Report since the last NRC routine ISFSI inspection and was reviewed during this inspection. The associated 10 CFR 72.48 screening for the revision was also reviewed.

In Revision 7, the licensee added changes to reflect CGS's use of non-destructive examination testing in lieu of load testing to verify continuing compliance for their Holtec supplied special lifting devices. This allowance is allowed by the Holtec FSAR and ANSI N14.6 standard for special lifting devices used at nuclear facilities. Other changes made to the report were editorial in nature, which included date changes, labeling, etc.

b. Findings

No findings were identified.

.3 Review of 10 CFR 72.48 Evaluations (60857)

a. Inspection Scope

The licensee's 10 CFR 72.48 screenings and evaluations since the last NRC routine ISFSI inspection were reviewed to determine compliance with regulatory requirements. A list of modifications to the ISFSI programs and changes to the fuel building overhead crane was provided by the licensee. The licensee had not performed any 72.48 ISFSI evaluations or any 50.59 crane evaluations since the last NRC inspection in July 2012. From the list of 72.48 screens provided by the licensee, 38 screenings were selected for further review. The licensee utilized Procedure SWP-LIC-02, "Licensing Basis Impact Determinations," Revision 12 to perform the 10 CFR 72.48 or 10 CFR 50.59 safety screenings or evaluations. None of the screenings reviewed required a full 10 CFR 72.48 or 10 CFR 50.59 safety evaluation. All screenings were determined to be adequately evaluated.

b. Findings

No findings were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On March 27, 2014, the inspectors presented the inspection results to Mr. M. Reddemann, Chief Executive Officer, and other members of the licensee staff. The licensee acknowledged the inspection details presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

SUPPLEMENTAL INSPECTION INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

P. Dugen, Reactor and Major Maintenance Supervisor
R. Fuller, ISFSI Project Manager
C. Moore, Reactor and Major Maintenance Supervisor
D. Suarez, Licensing Engineer
R. Torres, Quality Manager

Contractor Personnel

J. Kelly, Cask Loading Supervisor
J. Pavetto, Cask Loading Supervisor
D. Williams, Project Manager

INSPECTION PROCEDURES USED

IP 60855.1	Operations of an ISFSI 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

None

Discussed

None

Closed

None

LIST OF DOCUMENTS REVIEWED

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

4OA5.1 Other Activities

Calculations

NUMBER	TITLE	REVISION / DATE
NE-02-00-08	Appendix B – Assembly Data	Rev. 2
HI-2125144	Columbia Generating Station Vacuum Drying Evaluation without Annulus Flushing	07/19/12

Procedures

NUMBER	TITLE	REVISION / DATE
DIC 1814.5	Cask Loading Plan Number 2014-U01	11/05/13
DIC 1814.5	Cask Loading Plan Number 2014-U02	11/05/13
DIC 1814.5	Cask Loading Plan Number 2014-U03	11/05/13
DIC 1814.5	Cask Loading Plan Number 2002-4	11/12/02
DIC 1814.5	Cask Loading Plan Number 2004-8	04/07/04
DIC 1814.5	Cask Loading Plan Number 2004-10	04/15/04
PPM 6.6.6	MPC Fuel Loading	Rev. 14
PPM 6.6.5	Movement and Transfer Operations if HI-TRAC and HI-STORM in the Reactor Building	Rev. 14
PPM 6.6.12	Vacuum Drying System Operations	Rev. 6
PPM 6.6.13	Helium Backfill System Operation	Rev. 4
PPM 6.6.14	MPC Alternate Cooling Water System Operation	Rev. 1
PPM 6.6.15	Spent Fuel Cask Loading Verification	Rev. 6
PPM 6.6.7	MPC Processing	Rev. 25
PPM 9.6.1	Spent fuel Selection for Cask Storage	Rev. 6
PPM 6.6.4	HI-STORM System Site Transportation	Rev. 12
OSP-SFS-D101	Storage Cask Heat Removal System Daily Checks	Rev. 12
SWP-CAP-01	Corrective Action Program	Rev. 29
PPM 10.4.5	Reactor and Turbine Building Overhead Traveling Crane Inspection, Maintenance, and Testing	Rev. 23
SWP-LIC-02	Licensing Basis Impact Determinations	Rev. 12

Design Basis Documents

NUMBER	TITLE	REVISION / DATE
	Independent Spent Fuel Storage Installation 10 CFR 72.212 Evaluation Docket Number 72-35	Rev. 7
	Certificate of Compliance (CoC 72-1014 HI-STORM 100 Cask System	Amendment 2
	Holtec International Final Safety Analysis Report for the HI-STORM 100 Cask System	Rev. 4

Miscellaneous Documents

NUMBER	TITLE	REVISION / DATE
AR-SA #299169	ISFSI Self-Assessment Report	01/29/14
RRTI 2028-004	Response to Request for Technical Information	Rev. 1
RRTI 2028-005	Response to Request for Technical Information	Rev. 0
Report # 20350	Discussion of Minimum Required MPC Backfill Helium Mass	02/12/14
N/A	Columbia Generating Station 2012 Annual Environmental Operating Report	5/2013
Survey #2352613	Columbia Generating Station Health Physics Survey Map and Record	12/10/13
N/A	ISFSI Campaign 4/2014 ALARA Plan	N/A
HI-2125454	Structural Evaluation of HI-STORM Lifting Bracket	Rev. 0
HI-2125246	Structural Analysis of Lifting Pins	Rev. 0
HI-992272	Calculation Package for Cask Miscellaneous Item	Rev. 14
SMDR 2191	Supplier Manufacturing Deviation Report	Rev. 1
CE-02-00-12	Analysis of the Reactor Building Railroad Bay	Rev. 1
CE-02-13-02	Seismic Stability Analysis of Loaded HI-TRAC on Reactor Building 606' Elevation for ISFSI Operations	Rev. 0
CE-02-13-24	ISFSI Stacked Cask ANSYS Analysis	Rev. 0
LDCN 72212	Licensing Document Change Notice Form	09/07/12
N/A	Calibration Certificate #1394048565	03/06/14
N/A	Calibration Certificate #1394564608	03/12/14
N/A	Calibration Certificate #1395159708	03/19/14
QAP 9.16	Procedure for High Temperature Liquid Penetrant Examination	Rev. 7
QAP 9.6	Procedure for Liquid Penetrant Examination	Rev. 15

72.48 Screens

12-0002	13-0004	13-0014	13-0024
12-0003	13-0005	13-0015	13-0025
12-0004	13-0006	13-0016	13-0026
12-0005	13-0007	13-0017	13-0027
12-0006	13-0008	13-0018	13-0028
12-0007	13-0009	13-0019	14-0002
12-0008	13-0010	13-0020	
13-0001	13-0011	13-0021	
13-0002	13-0012	13-0022	
13-0003	13-0013	13-0023	

Work Orders

WO 02036654	WO 02036364	WO 02037517	WO 02052336
WO 02021596	WO 02004016		

Condition Reports

AR-CR 00264993	AR-CR 00265969	AR-CR 00267587	AR-CR 00278590
AR-CR 00287994	AR-CR 00288520	AR-CR 00289370	AR-CR 00290507
AR-CR 00291797	AR-CR 00291807	AR-CR 00294399	AR-CR 00297522
AR-CR 00298443	AR-CR 00298603	AR-CR 00300522	AR-CR 00300588
AR-CR 00301398	AR-CR 00301722	AR-CR 00302115	AR-CR 00303958
AR-CR 00267573	AR-CR 00271229	AR-CR 00286654	AR-CR 00295430
AR-CR 00289324	AR-CR 00271577	AR-CR 00298499	

LIST OF ACRONYMS

ADAMS	Agencywide Documents Access and Management System
AFS	Areva Federal Services LLC
ANSI	American National Standards Institute
AR-CR	Action Requested Condition Report
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CAP	Corrective Action Program
CGS	Columbia Generating Station
CoC	Certificate of Compliance
CFR	Code of Federal Regulations
DNMS	Division of Nuclear Material Safety
FSAR	Final Safety Analysis Report
IMC	Inspection Manual Chapter
ISFSI	Independent Spent Fuel Storage Installation
kW	kilo-watt
mR/h	milliroentgen per hour
mrem/h	milli-rem per hour
μR/h	microroentgen per hour
MPC	Multi-Purpose Canister
MWd/MTU	megawatt days/metric ton uranium
NRC	U.S. Nuclear Regulatory Commission
PA	Protected Area
PPM	Plant Procedures Manual
QA	Quality Assurance
RP	radiation protection
TLD	thermoluminescent dosimeter

ATTACHMENT 2:

LOADED CASKS AT THE COLUMBIA GENERATING STATION ISFSI

LOADING ORDER	MPC SERIAL No.	HI-STORM No.	DATE ON PAD	HEAT LOAD (kW)	BURNUP MWd/MTU (max)	MAXIMUM FUEL ENRICHMENT %	PERSON-REM DOSE
1	MPC-68-028	Serial No. 018	09/20/02	10.81	32,299	2.72	0.385
2	MPC-68-031	Serial No. 019	10/07/02	11.10	32,416	2.72	0.341
3	MPC-68-022	Serial No. 020	10/28/02	11.30	32,541	2.72	0.315
4	MPC-68-039	Serial No. 021	11/18/02	11.42	33,045	2.72	0.298
5	MPC-68-033	Serial No. 022	12/09/02	11.20	32,804	2.72	0.245
6	MPC-68-091	Serial No. 119	02/25/04	12.00	32,318	2.72	0.390
7	MPC-68-092	Serial No. 120	03/05/04	17.10	38,607	2.92	0.298
8	MPC-68-093	Serial No. 121	03/11/04	17.10	38,738	2.92	0.320
9	MPC-68-094	Serial No. 122	03/18/04	17.00	38,732	2.92	0.304
10	MPC-68-095	Serial No. 123	03/24/04	17.00	38,772	2.92	0.276
11	MPC-68-096	Serial No. 124	03/31/04	17.10	38,729	2.92	0.253
12	MPC-68-097	Serial No. 125	04/06/04	17.20	39,121	2.92	0.251
13	MPC-68-098	Serial No. 126	04/14/04	17.10	39,002	2.92	0.237
14	MPC-68-099	Serial No. 127	04/20/04	17.00	39,008	2.92	0.208
15	MPC-68-100	Serial No. 128	04/25/04	16.80	38,982	2.92	0.199
16	MPC-68-162	Serial No. 225	02/22/08	11.91	39,172	3.22	0.260

LOADING ORDER	MPC SERIAL No.	HI-STORM No.	DATE ON PAD	HEAT LOAD (kW)	BURNUP MWd/MTU (max)	MAXIMUM FUEL ENRICHMENT %	PERSON-REM DOSE
17	MPC-68-163	Serial No. 226	02/29/08	21.00	43,302	3.56	0.458
18	MPC-68-164	Serial No. 227	03/07/08	21.02	43,181	3.56	0.426
19	MPC-68-165	Serial No. 228	03/14/08	21.02	43,010	3.56	0.343
20	MPC-68-166	Serial No. 070	03/21/08	21.00	43,330	3.56	0.379
21	MPC-68-167	Serial No. 071	04/04/08	21.03	42,827	3.56	0.587
22	MPC-68-168	Serial No. 072	04/11/08	21.01	43,020	3.56	0.503
23	MPC-68-169	Serial No. 073	04/29/08	20.88	42,269	3.87	0.362
24	MPC-68-170	Serial No. 074	05/05/08	20.73	42,281	3.87	0.333
25	MPC-68-171	Serial No. 075	05/10/08	20.96	42,390	3.87	0.306
26	MPC-68-172	Serial No. 081	05/16/08	20.97	42,349	3.87	0.279
27	MPC-68-173	Serial No. 082	05/25/08	20.98	41,600	3.87	0.315
28	MPC-68-366	Serial No. 587	03/08/14	13.53	38,702	2.92	0.304
29	MPC-68-367	Serial No. 588	03/15/14	20.58	44,031	3.86	0.479
30	MPC-68-368	Serial No. 589	03/22/14	20.57	43,884	3.86	0.476
31	MPC-68-369	Serial No. 590	03/29/14	20.55	43,860	3.86	0.435

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

Casks 1 – 15 were loaded to CoC Amendment 1, FSAR Revision 1
Casks 16 – 31 were loaded to CoC Amendment 2, FSAR Revision 4