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

August 24, 2012

Mr. M. E. Reddemann
Chief Nuclear Officer
Energy Northwest
P.O. Box 968 (Mail Drop 1023)
Richland, Washington 99352-0968

SUBJECT: COLUMBIA GENERATING STATION – INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) INSPECTION REPORT 05000397/2012008 AND 07200035/2012001

Dear Mr. Reddemann:

A routine inspection was completed of your dry cask storage activities associated with your Independent Spent Fuel Storage Installation (ISFSI) on July 24 - 25, 2012. An exit was conducted with your staff to discuss the findings of the inspection on July 25, 2012. The inspection reviewed the current storage activities associated with your ISFSI. Between 2002 and 2008, you completed several loading campaigns which placed twenty-seven Holtec HI-STORM 100 casks onto your pad. Since then, no additional loadings have been performed and none are scheduled for this year. The focus of this inspection was to review the status of the stored casks to verify ongoing compliance with the Holtec Certificate of Compliance No. 1014 and associated Technical Specifications; the Holtec Final Safety Analysis Report; the regulations in 10 CFR Part 20 and Part 72; and to review any changes that had been made to your ISFSI program since the last NRC inspection. Your ISFSI operations were found to be in compliance with the applicable NRC regulations and requirements and your storage casks were found to be in good physical condition. 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 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. Vincent Everett at (817) 200-1198.

Sincerely,

/RA/

D. Blair Spitzberg, Ph.D., Chief
Repository & Spent Fuel Safety Branch

Dockets: 50-397, 72-35
Licenses: NPF-21

Enclosure: Inspection Report 05000397/2012008;07200035/2012001

w/ Attachment:
1. Supplemental Information
2. Loaded Casks at the Columbia Generating Station Site

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DRAFT: S:\DNMS\RSFS\!JVE\CGS2012008-ISFSI-JVE
FINAL: R:\REACTORS\CGS\2012\CGS2012008-ISFSI-JVE

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

RIV:DNMS/RSFS	RIV:DNMS/RSFS	R-IV/C:RSFS		
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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	05000397, 07200035
Licenses:	NPF-21
Report Nos.:	05000397/2012008 and 07200035/2012001
Licensee:	Energy Northwest
Facility:	Columbia Generating Station Independent Spent Fuel Storage Installation (ISFSI)
Location:	Richland, Washington
Dates:	July 24 - 25, 2012
Inspectors	Vincent Everett, Senior Inspector Lee Brookhart, Inspector
Accompanied By:	Lynn Albin, Health Physicist Department of Health State of Washington
Approved By:	D. Blair Spitzberg, Ph.D., Chief Repository and Spent Fuel Safety Branch Division of Nuclear Materials Safety

EXECUTIVE SUMMARY

Columbia Generating Station
NRC Inspection Report 50-397/2012-08 and 72-35/2012-01

A routine inspection was conducted of the twenty-seven Holtec 100 casks currently stored at the Columbia Independent Spent Fuel Storage Installation (ISFSI). These casks had been loaded from 2002 to 2008. No additional loading campaigns had been performed since 2008. The current ISFSI staff was effectively monitoring the loaded casks and keeping the ISFSI program current by monitoring industry activities at other sites and applying lessons learned to the Columbia ISFSI program. This included updating procedures that would be implemented during future loading campaigns. The ISFSI staff was very knowledgeable of the requirements in the Certificate of Compliance and the Holtec Final Safety Analysis Report (FSAR) that applied to the Columbia ISFSI and effectively interfaced with the NRC inspectors during the inspection. Ongoing temperature monitoring of the storage casks had demonstrated that the casks were being maintained within the technical specification required temperature limits. Dosimeters along the ISFSI fence were providing radiological dose data within the expected levels for an ISFSI with twenty-seven casks in storage and showed decreasing radiation levels over time reflecting the radiological decay of the stored spent fuel.

<u>Operation of an ISFSI at Operating Plants (60855.1)</u>

- The audit and surveillance program implemented by the plant's Quality Services organization reviewed activities associated with the ISFSI including the ongoing temperature monitoring program, radiological surveys of the ISFSI, procedure compliance, records retention, worker training, and implementation of the 10 CFR 72.48 safety evaluation program. The audit and surveillance reports presented good information directed at improving organizational performance and providing management with feedback on the status of the various ISFSI related activities (Section 1.2.a).

- Environmental monitoring for the ISFSI included ten thermoluminescent dosimeters (TLDs) located on the security fence around the ISFSI. The fence was approximately 40 feet from the nearest casks. Radiation levels for 2011 measured on the ten TLDs ranged from 312 mrem/yr on the east side to 689 mrem/yr on the south side (Section 1.2.b).

- A radiological survey along the ISFSI fence found gamma radiation levels ranging from approximately 40 microR/hr (350 mrem/yr) to 80 microR/hr (700 mrem/yr). No measurable neutron radiation levels were found. These levels were consistent with the values measured by the TLDs (Section 1.2.b).

- Technical Specification 3.1.2 required air vent temperature surveillances and/or vent inspections to be performed daily on the storage casks to ensure the spent fuel was adequately cooled. Documentation reviewed provided sufficient evidence that the temperature monitoring requirements of Technical Specification 3.1.2 were performed as required (Section 1.2.c).

- An issue was identified by the licensee concerning the affect of aging on the ISFSI pad which results in hardening of the concrete. This was an issue because the hardness of the pad must not reach a point where the deceleration affect on the spent fuel during a cask tip over event could exceed 45g. Initial calculations by the licensee suggested that the pad hardness could reach a condition in less than the 20 year license of the ISFSI where the 45g limit could be exceeded. The cask vendor, Holtec Int., provided analysis specific to the Columbia pad that determined that the 45g limit would not be reached. Further evaluation by Entergy Northwest using the Holtec analysis and data developed by the Portland Cement Association determined that the pad service life would exceed 40 years (Section 1.2.d).

- Selected condition reports over the past four years were reviewed related to the ISFSI and the reactor building crane. A wide variety of issues had been documented including generic industry issues that could affect the Columbia ISFSI. Corrective actions taken to resolve the issues were found to be adequate. No significant trends were identified by the NRC inspectors from the list of condition reports provided (Section 1.2.e).

- Helium leak testing of canisters during fabrication had been discontinued for a period of time by Holtec. Subsequently, the NRC required Holtec to re-initiate the testing on canisters during fabrication. In the interim, a number of canisters had been shipped to licensees including twelve that had been loaded and placed on the Columbia ISFSI pad. NRC review of data provided by Holtec and Energy Northwest concluded that the twelve loaded canisters at Columbia were acceptable for continued use (Section 1.2.f).

- Industry related issues identified by the NRC and Holtec technical groups were reviewed for applicability to Columbia's ISFSI program and facility. The licensee adequately incorporated resolutions to the issues identified in NRC Information Notice 2011-10 and Holtec Information Bulletins (HIB) 45, 48, and 53 into their program (Section 1.2.f).

<u>Review of 72.212(b) Evaluations (60856.1)</u>

- The 72.212 Evaluation Report was being kept current and had included industry issues that affect the Columbia site, even though no new loading campaigns had been initiated since 2008 (section 2).

<u>Review of 72.48 Evaluations (60857)</u>

- All 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. Two safety evaluations had been performed since 2008. Both evaluations were in compliance with regulatory requirements (Section 2).

<u>Follow-up on Corrective Actions for Violations and Deviations (92702)</u>

- Notice of Violation 72-35/0801-01 issued during the last NRC ISFSI inspection concerning hydrogen monitoring during welding was adequately addressed and appropriate changes made to procedures. This item was closed (Section 4).

Report Details

Summary of Facility Status

A total of twenty-seven Holtec 100 casks were currently stored at Entergy Northwest's Columbia Independent Spent Fuel Storage Installation (ISFSI) located adjacent to the Columbia Generating Station. The casks were being monitored for temperature in compliance with Technical Specification 3.1.2 and for radiation levels in compliance with 10 CFR Part 20. The casks were in an ISFSI separate from the plant's Part 50 protected area. The ISFSI consisted of two concrete pads, each 30 feet by 135 feet. Each pad can hold 18 casks. The ISFSI pad was approximately 500 meters north-northwest of the Columbia reactor building. Three loading campaigns had been completed at the ISFSI starting in 2002 with five casks placed on the ISFSI pad. In 2004, ten more casks were added and in 2008, twelve more casks were added. The first and second loading campaigns were completed under Amendment 1 of the Holtec Certificate of Compliance No. 1014 and Revision 1 of the Holtec Final Safety Analysis Report (FSAR). The third loading campaign was completed under Amendment 2 of the Certificate of Compliance and Revision 4 of the FSAR. A tour of the ISFSI area found the casks to be in good physical condition, well secured in the ISFSI protected area. No significant ISFSI activities were underway except for the ongoing monitoring of the casks in storage. The next cask loading campaign was planned for 2014. Nine more casks will fill the current ISFSI pads. After that, the ISFSI will be expanded and additional pads will be constructed.

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

1.1 Inspection Scope

A review was performed of documentation related to the storage of the twenty-seven casks currently at the ISFSI. This review covered audits and surveillances conducted by the licensee, condition reports related to the ISFSI and the reactor building crane, environmental radiological data collected around the ISFSI for the past several years, compliance with Technical Specification 3.1.2 for temperature monitoring of the casks, and current issues that relate to the Holtec cask system that have been documented at other ISFSI sites relevant to Energy Northwest for future cask loadings. A tour of the ISFSI area was performed and radiological doses were measured by the licensee along the complete perimeter of the fence to verify appropriate placement of the environmental dosimetry.

1.2 Observations and Findings

a. Quality Assurance Audits and Surveillances

Three audits and several surveillances performed by the plant's Quality Services organization were reviewed. Audit Report AU-DC-08, documented the Quality Services audit conducted between March 11, 2008, and May 8, 2008. The audit reviewed ISFSI activities for the previous 24 month period. Specific areas reviewed included training and qualifications, procedures, design control, procurement documentation, ISFSI operations, and improvements in the ISFSI program. Issues identified during the audit were documented in the licensee's corrective action program. A review of the qualifications of ISFSI personnel found all personnel were qualified for the tasks they had been assigned, however, there were inconsistencies noted in the work orders where qualification requirements weren't always listed. The review of procedures and procedure use during loading noted that procedures were being followed. As procedural steps were completed, the steps were marked as complete. Several examples were noted during the observation of work activities where procedural deficiencies were found. In the area of design control, several issues were discussed. These included explosive gas monitoring and venting during welding, loading of fuel with decay times shorter than allowed by the FSAR, requirements related to annulus cooling for casks with heat loads greater than 21.52 kW, and the load limit setting on the 125 ton reactor building crane that had been set at 139 tons.

Audit Report AU-DC-10 documented the Quality Service audit conducted between April 9, 2010, and May 13, 2010. Audit areas included ISFSI operations, regulatory impact reviews, procedures, and continuous improvement. Overall, the audit found the ISFSI program to be effectively implemented. Several procedural recommendations were made to improve procedures. The review of 72.48 applicability determinations and screenings found the conclusions to be technically supported. Several issues were identified related to storage of records, documentation of daily temperature surveillances, and coordination between workers and the radiological protection organization for work performed outside the protected area.

Audit Report AU-DC-12 documented the Quality Services audit conducted between April 12, 2012 and May 3, 2012. The audit identified problems with the remote temperature monitoring system for the casks stored at the ISFSI as an ongoing issue. The temperature monitoring computer issues had been resolved, but the temperature detectors on two of the casks were not currently working. Daily vent inspections by an equipment operator performing a tour around the ISFSI were being implemented as a compensatory action. Problems with the temperature monitoring system had been ongoing for over 300 days. The audit also reviewed health physics surveys of the ISFSI and confirmed that the survey results demonstrated that the offsite dose to the public had not changed significantly in the past two years and was within regulatory limits. Review of 72.48 applicability determinations and screenings found that the conclusions were technically supported. Required records for the ISFSI program were found to be properly stored and controlled. The audit noted that recent evaluations related to the Holtec Information Bulletins concerning individual cell heat load calculations and canister lid minimum lifting capacities was thorough.

Several surveillances performed by the Quality Services organization were reviewed. The surveillance for the 12 month period from April 2007 to March 2008, issued in a report dated July 1, 2008, included a review of condition reports, audit reports, self-assessments, station trends, assessment reports by the Institute of Nuclear Power Operations (INPO), and station performance indicators. All of these areas were generic to site operations including the ISFSI. The only issue identified specific to the ISFSI related to control of keys for the ISFSI.

The surveillance for the period January through March 2010, issued as a report dated April 29, 2010, performed a number of oversight activities and audits including the engineering and fire protection audit and the emergency preparedness audit. Activities reviewed during the monitoring period included administrative procedure compliance, work controls, equipment reliability issues, availability of parts to support work activities, and observation of activities being performed by the various plant organization including operations, engineering, radiation protection, chemistry, security, training, and maintenance. No specific issues were identified during the surveillance related to the ISFSI, however, this appeared to be a thorough evaluation of the work activities and controls of the numerous plant organizations that were also responsible for implementing the ISFSI programs.

The surveillance for the period of April through June 2010, issued as a report dated July 8, 2010, reviewed various reports, self assessments, condition reports, and performance indicators. Attention was directed during this audit to identify common causes and cross cutting issues to provide focused feedback to station management. Areas reviewed included organizational effectiveness, administrative procedure compliance, emerging issues, training performance, timely reporting of “near miss” issues, developing trends based on performance indicators, awareness of industry events that could affect the plant, adequacy of corrective action resolutions, and observation of work activities being performed by the various plant organizations. Though no specific issues were identified related to the ISFSI, the surveillances covered a very broad spectrum of activities related to improving plant organizational performance and to appraise upper management of issues that needed their attention.

All the audit reports reviewed during this inspection reflected a thorough review of issues and presented good examples and data to support the conclusions reached in the audit. The issues were examined in a way to present good assessments that reflected the condition of the situation such that corrective actions could be identified that would contribute to improving performance.

In addition to the audits and surveillances performed of activities at the Columbia site, several recent surveillances performed at Holtec’s Manufacturing Division in Turtle Creek (Pittsburg), PA were reviewed. This facility fabricated the Holtec canisters used at several sites including Columbia. Reports of the surveillance conducted December 12 – 15, 2011, and two surveillances on January 24 – 27, 2012, were reviewed. The surveillances documented observation of fabrication processes underway, observation of the quality assurance inspectors performing their assigned quality control work activities during fabrication of several components, and verification of the qualifications of one of the Holtec non-destructive examination (NDE) inspectors.

b. Environmental Radiological Monitoring Program

The annual Radiological Environmental Operating Reports for 2009, 2010, and 2011 were reviewed. Data was provided for the ten thermoluminescent dosimeter (TLDs) locations on the ISFSI fence and two TLD locations that were placed near the ISFSI. Both quarterly and annual TLDs were stationed at each location. Since the last cask placed in the ISFSI was in 2008, the TLD results reflected the storage of the twenty-seven casks over the three year monitoring period. The ISFSI was located approximately 500 meters north-northwest of the reactor building. The ISFSI was rectangular shaped with two storage pads (a north pad and a south pad). The two pads were each 30 feet by 135 feet. Each pad can hold up to 18 casks. The north pad was fully loaded with 18 casks. The south pad had nine loaded casks. The ISFSI fence where the TLDs were located was approximately 40 feet from the nearest casks. The TLDs used by the licensee were Harshaw Model 8807 TLDs placed approximately three feet above the ground along the fence.

Data provided below in Table 1 shows the annual TLD data for 2009, 2010, and 2011. The TLDs can be influenced by radiation from the turbine building when the plant is operating. The TLDs on the south side of the ISFSI were closer to the turbine building than those on the north side. TLD 121 was approximately 0.1 miles north of the plant. The ISFSI was approximately 500 meters (0.31 miles) north-west of the reactor building. As can be seen, the radiation levels are decreasing over time. This can be attributed to the decay over time of the spent fuel stored in the casks.

Table 1: ISFSI Environmental TLD Data (mrem/year)

TLD #	LOCATION	2009	2010	2011
123	north fence – west of center	593	589	505
124	north fence – center	693	654	566
125	north fence – east of center	499	476	427
126	west fence – north of center	525	492	422
127	east fence – north of center	374	360	312
128	west fence – south of center	668	589	546
129	east fence – south of center	481	469	406
136A	south fence – west of center	764	698	598
137A	south fence – center	865	787	689
138A	south fence – east of center	688	621	550
	LOCATIONS NOT ON FENCE			
86	north-east of ISFSI	115	120	108
121	south of ISFSI about 300 m	298	356	225
122	north of ISFSI about 100 m	161	159	141

Source: Table 5-4 "Annual TLD Data Summary with Comparison to Preoperational and Operational Periods" from the 2009, 2010, and 2011 Annual Radiological Environmental Operating Reports

During the tour of the ISFSI area, a radiological survey was performed by the licensee of the fence line where the environmental TLDs were located. The survey was conducted to correlate radiological levels measured on the survey instruments with the annual TLD radiation results and to verify that the placement of the TLDs on the fence were representative of the radiation levels around the ISFSI. Both neutron and gamma readings were taken and are summarized in Table 2 below. The readings were taken using a Thermo-microrem meter for gamma and a remball for neutron. The first set of readings was taken at a building between the ISFSI and the plant prior to starting the survey. The gamma levels were 20 microR/hr. Neutron levels were at the instrument's minimum sensitivity level of 0.5 mrem/hr. Throughout the survey, no neutron levels were measured above the remball's minimum sensitivity.

Table 2: Radiological Survey Results of ISFSI

Side Along Fence (TLDs are on fence)	Predominant Readings
South	70 – 80 μR/hr
West	50 – 60 μR/hr
North	60 – 70 μR/hr
East	40 – 50 μR/hr

Source: Survey 1677712 "Columbia Health Physics Survey Map and Record" dated July 25, 2012

Using 8,760 hours in a year, an hourly radiation level of 40 μR/hr would equate to 350 mrem/yr, 60 μR/hr would equate to 525 mrem/yr and 80 μR/hr would equate to 700 mrem/yr. The location of the TLDs was found to be representative of the radiation levels around the ISFSI pad. No unusually high radiation levels were found during the ISFSI survey.

Data provided in the Energy Northwest 10 CFR 72.212 Evaluation Report, Section 3.1 "Background Radiation Levels Prior to Storing Fuel" provided a table of 1999 TLD results for three dosimeters located from 0.3 to 0.4 miles north of the plant that were used to represent background levels. These were Stations 71, 72, and 86. Background levels were listed in the table as 0.240 to 0.249 mrem/day representing pre-ISFSI operations. This would be approximately 10 μR/hr and 88 mrem/yr. The Annual Radiological Environmental Operating Report, Table 5-4 "Annual TLD Data Summary with Comparison to Preoperational and Operational Periods" for 2011 listed the annual TLD results for these same three TLD stations as 95.7, 91.8, and 107.7 mrem/yr, respectively. Station 86 was relative near to the ISFSI.

Holtec Report HI-2002448 "Radiation Shielding Analysis for the ISFSI at WNP-2," Revision 5 was reviewed concerning the offsite dose projections to the public from the ISFSI. The analysis had been performed based on the ISFSI being fully loaded with the projected 90 casks on the ISFSI pad and compared the calculated dose at the site boundary to the 25 mrem/yr limit in 10 CFR 72.104. The site boundary was 1,950 meters from the containment building. From the ISFSI, the closest site boundary would be 1,450 meters to the northwest. The nearest real resident was approximately four miles to the east north-east. The whole body dose at the 1450 meter site boundary was calculated to be 0.0364 mrem/yr (36 μR/hr) above background, well below the 25 mrem/yr limit. This value was shown in the table on page S5 of the report. The Holtec report also provided a table showing doses at various distances on page S4. The 0.0364 mrem/yr at 1450 meters was calculated to be 6.62 mrem/hr at 40 feet, which was the approximate distance between the ISFSI pad and the TLDs on the fence. The highest reading measured along the fence during the survey was 80 μR/hr (0.080 mrem/hr), which was a factor of over 80 times lower than the value in the Holtec report, demonstrated that the dose at the site boundary has been well below the 10 CFR 72.104 limit due to the current 27 casks in the ISFSI.

The main body of Holtec Report HI-2002448 calculated the site boundary dose based on 45,000 Megawatt Days/Metric Ton Uranium (MWD/MTU) and 13 year cooling of the spent fuel. Revision 4 of the report added Supplement 1 which calculated the site boundary dose based on spent fuel with a minimum of 5 years cooling time and a burnup of 45,000 MWD/MTU. Revision 5 revised Supplement 1 for 4.747 years cooling and 42,390 MWD/MTU. The tables on page S4 and S5 provided the calculated dose at 1450 meters. The discussion of the revision to the report stated on page S8 that administrative controls will be needed to require the fuel assemblies with maximum burnup of 42,390 MWD/MTU and 4.747 years cooling to be placed in Region 1 (inner region) of the canister. The licensee had incorporated this requirement in Plant Procedure Manual (PPM) 9.6.1 “Spent Fuel Selection for Cask Storage,” Revision 5. Several steps in the procedure specifically stated that spent fuel with cooling times between 4.747 years and 5 years must have an exposure less than or equal to 42,390 MWD/MTU and must be loaded in Region 1 (inner region) of the canister.

Certificate of Compliance 1014, Amendment 2 allowed for regional loading of fuel with cooling times as short as 3 years as specified in the Certificate of Compliance, Appendix B, Section 2.4.2 “Regionalized Fuel Loading Decay Heat Limit for Zr-Clad Fuel.” Figure 2.1-4 “Fuel Loading Regions-MPC 68/68F” provided a drawing showing where the inner region of the canister was located where the hotter fuel must be loaded. There were 32 out of the 68 slots assigned to the inner region. Fuel assembly selection records for canisters 170, 171, 172, and 173 (the 24th thru 27th casks loaded) were reviewed to verify that all the fuel loaded was above the 4.747 year requirement and if less than five years, had been placed in the inner region of the canister. The youngest spent fuel assemblies loaded in Canister 170 had been removed from the reactor on May 4, 2003. Canister 170 was placed on the ISFSI pad on May 5, 2008, five years after the spent fuel had been removed from the reactor. Canister 170 was loaded with spent fuel ranging from five years cooling time to twenty-one years. Thirty-two of the assemblies had five year cooling times. All were placed in the inner region of the canister.

Canisters 171, 172, and 173 were loaded with basically the same parameters as Canister 170 with spent fuel ranging from five years to twenty-one years. Thirty-two assemblies, placed in the inner region, had been cooled for 5 years prior to placing the three canisters on the ISFSI pad between May 10, 2008, and May 25, 2008. The individual decay heat of the assemblies loaded in the four casks were similar and ranged from 90 watts to 498 watts based on a date of May 13, 2007. The burnup values for the four casks ranged from 41,600 to 42,390 MWD/MTU.

c. Technical Specification 3.1.2 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 °F. All 27 HI-STORM casks at Columbia 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 10. The NRC inspectors requested daily temperature monitoring system data for the months of December 2008, April 2009, October 2009, June 2010, July 2011, and January 2012. This data was reviewed to verify that the 126 °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 ^oF 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 ^oF was an administrative limit to initiate corrective actions before the 126 ^oF 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, no cask exceeded the 114 ^oF delta T limit. For the months of June 2010 and January 2012, compliance with Technical Specification 3.1.2 was performed by vent inspections due to problems with the thermal monitoring system not working properly for all casks. Throughout the six months of documentation reviewed, the temperature monitoring system appeared to frequently malfunction, requiring a substantial number of times the operator was required to perform a daily visual inspection of the casks.

The delta T value between the average of the four outlet temperature detectors and the ambient temperature varied widely from day to day. The licensee had concluded that a primary factor for these variances was due to the affects of wind speed. Other factors such as sudden ambient temperature changes due to the transition of cold fronts or warm fronts and affects due to rain showers would also affect the delta T values. At Columbia, the temperature printout was generated daily at around the same time every day, at about 7:45 am. Below is a table with values associated with Cask #17 (21 kW) for a selected week during December 2008.

Table 3: Temperature Readings for Cask #17

Date	Delta T	Ambient Temperature	Average Wind Speed (mph) near the site	Average Wind Direction (0^o is North)
12/23/2008	89.6 ^oF	9.9 ^oF	1.300	240.907^o
12/24/2008	87.8 ^oF	13.3 ^oF	0.267	205.047^o
12/25/2008	69.3 ^oF	18.1 ^oF	7.877	166.447^o
12/26/2008	110.3 ^oF	9.3 ^oF	0.000	279.720^o
12/27/2008	70.6 ^oF	18.3 ^oF	4.833	323.467^o
12/28/2008	49.8 ^oF	19.8 ^oF	8.250	333.440^o
12/29/2008	65.5 ^oF	27.1 ^oF	6.140	321.857^o

Source: Temperature Records provided by licensee from Temperature Monitoring System

When the wind speed was low, the difference between the temperature detectors and ambient temperature tended to be high. Higher wind speeds caused more air flow through the vents, removing heat, and contributed to a lower delta T value. The 110.3 °F delta T value on December 26, 2008, was the highest recorded value during the six months of documentation reviewed. The surveillance operator recognized Cask #17 was approaching the alarm set-point and that all casks were reading 20 – 30 degrees greater than the previous day's values. Condition Report (CR) Number 190373 was initiated on December 26, 2008 to document the increase in temperature readings and stated that an equipment operator was dispatched to inspect the casks' vents. Upon inspection, the operator discovered snow exceeding six inches in height had accumulated and was partially blocking the vents on numerous casks. Action was taken to remove the snow from around the vents. The high delta T was attributed to the accumulation of snow partially blocking the vents and stagnant air speed on that day.

d. <u>ISFSI Pad Concrete Hardness Due to Aging</u>

The design basis for the ISFSI pad must provide for a pad with sufficient stiffness to meet the strength requirements set forth in the specific reinforced concrete code, but at the same time be sufficiently flexible (soft) so that the maximum load conditions the pad imposes on the cask during a drop or tipover event will not exceed the deceleration design limits established for the spent fuel stored in the cask. The determination of the effect of the pad on the cask must consider not only the pad, but also the soil under the pad. The Holtec FSAR, Revision 0, Table 3.1.2 "Design Basis Decelerations for Drop Events" established a deceleration limit of 45g due to gravity as the design basis value for both a vertical axis drop and a horizontal axis (side) drop. Holtec FSAR Section 3.4.10 "HI-STORM 100 Non-Mechanistic Tip-Over and Vertical Drop Event (Load Case 02.a and 02.c in Table 3.1.5)" stated that the 45g deceleration applied to the top of the fuel basket. Appendix 3.A "HI-STORM Deceleration under Postulated Vertical Drop Event and Tipover" evaluated the tipover event using the pad and subsoil values in Table 3.A.1 "Essential Variables for Reference ISFSI Pad Data." Table 3.A.1 was the same as Table 2.2.9 "Characteristics of Reference ISFSI Pad" with several additional input parameters. The key parameters included 4,200 pounds/square inch (psi) at 28 days for the concrete compressive break strength, a pad thickness of less than or equal to 36 inches, and a subgrade soil effective modulus of elasticity of less than or equal to 28,000 psi. Based on a pad built to these parameters, the calculated maximum acceleration at the top of the fuel basket was shown in Table 3.A.4 "Results" as 43.12g. For the top of the cask at 231.25 inches, the impact velocity was 341.3 inches/second resulting in a maximum acceleration of 48.41g. Since the fuel basket is 206 inches, the forces at the top of the fuel basket based on the geometry of the cask would be 206/231.25 = 89.09%. Applying this factor to the calculations resulted in an acceleration of 43.12g at the top of the basket holding the spent fuel. These results showed that the design basis acceleration of 45g was not exceeded for a pad built to the parameters in Tables 3.A.1 and 2.2.9.

The two concrete pads at the Columbia ISFSI were poured in October 2001. The haul road, which is mostly compacted dirt and gravel, included several concrete areas that were poured between September 2001 and February 2002. At that time, Certificate of Compliance 1014, Revision 0 and Holtec FSAR, Revision 0 were in effect. Appendix B of the Certificate of Compliance, Section 3.4 "Site Specific Parameters and Analysis" required the ISFSI pad concrete compressive strength to be less than or equal to 4,200 psi at 28 days. Appendix A of the Certificate of Compliance, Section 5.5 "Cask Transport Evaluation Program" also required the transport route to meet the 4,200 psi specification in Appendix B, Section 3.4.6. If the transportation route could not meet the

Section B.3.4.6 criteria, a site specific evaluation was allowed by Section A.5.5 (b) to demonstrate the impact loading due to a design basis drop event would not exceed 45g.

In the June 2000 time frame prior to pouring the ISFSI pad, Holtec issued Holtec Report HI-2002452 “Accident Analysis of the HI-STORM 100 Tipover and End Drop for WNP-2” and Report HI-2002454 “Evaluation of HI-STORM 100 Tipover Accident at the WNP-2 ISFSI Pad with an Assumed High Concrete Compressive Strength.” Approximately one month prior to that time, Energy Northwest had issued Calculation No. CE-02-00-04 “Long Term Concrete Strength of ISFSI Concrete Pad.” These three documents evaluated the ISFSI pad in relationship to the tipover accident. Holtec Report HI-2002452 provided a transient finite element analysis calculation to demonstrate that for a non-mechanistic tipover event and an end drop, the HI-STORM 100 cask used at Energy Northwest would not exceed the 45g design basis deceleration limit. The computer code LS-DYNA3D, Version 950 was used. Table 1 “Configuration and Properties of the ISFSI Pad at WNP-2” of Report HI-2002452 provided the subsoil parameters under the pad based on geotechnical evaluations of the pad site and the design values for the slab. These properties are listed below in Table 4.

Table 4: Columbia ISFSI Pad Design

ISFSI Pad Layer	Thickness (inches)	Young’s Modulus or Compressive Strength (psi)	Poisson’s Ratio
Concrete slab	24	4,200 @ 28 days	0.22
Engineering Fill Type B	28	28,000	0.35
Engineering Fill Type A	30	16,000	0.3
Hanford Deposit	---	16,000	0.3

Source: Table 1 “Configuration and Properties of the ISFSI Pad at WNP-2” from Holtec Report HI-2002452

For the pad calculations in Report HI-2002452 to verify compliance with the 45g limit, Holtec used more conservative values. Report HI-2002452, Table 2 “Data of ISFSI Pad Used in the Analysis” provided the following values that were used in the calculations.

Table 5: Data Used for Columbia Pad Analysis by Holtec

Thickness of concrete	36 inches
Nominal compressive strength of concrete	4,200 psi @ 28 days
Concrete mass density	2.097×10^{-4} lb-sec^2/inch^4
Concrete Poisson’s ratio	0.22
Mass density of soil	1.872×10^{-4} lb-sec^2/inch^4
Effective modulus of elasticity of the subgrade soil	28,000 psi
Poisson’s ratio of the soil	0.4

Source: Table 2 “Data of ISFSI Pad Used in Analysis” from Holtec Report HI-2002452

These values represented a stiffer soil under the pad and bounded the site specific characteristics for the Columbia pad described in Table 4 above. These values were identical to Table 3.A.1 of the Holtec FSAR. The calculations for the tipover were presented in Table 5 "Results" of Holtec Report HI-2002452 and were identical to the results presented in the Holtec FSAR, Table 3.A.4 listing the maximum acceleration for the top of the fuel basket as 43.12g. These results demonstrated that the design basis acceleration of 45g was not exceeded using the Table 5 values above for the concrete and subsoil hardness, and as such would be even lower for the actual soil values at the Columbia pad as shown in Table 4 above.

Holtec Report HI-2002454 evaluated the forces on the cask during a tipover for a concrete pad that had an initial concrete compressive strength of 4,200 psi at 28 days, but had aged over time and reached a postulated compressive strength of 6,100 psi at 20 years. Prior to this analysis, Energy Northwest had issued Calculation CE-02-00-04, Revision 0 which looked at the hardening of the concrete pad over time to verify that the 4,200 psi at 28 days was an appropriate criteria for their pad during the construction. Energy Northwest wanted to verify that over the 20 year life of the pad, the hardness of the pad would not reach a point where the 45g tipover limit would be exceeded. Energy Northwest decided to reduce the 28 day break strength design criteria for the pad from 4,200 psi to 3,800 psi to ensure aging of the pad would not present a problem. They also required Type III concrete to be used in the construction because that was a softer concrete mix. These two criteria were incorporated into the pad construction specification. Energy Northwest performed a calculation for the hardening of the concrete using the 3,800 psi and Type III concrete and used data from the Portland Cement Association (PCA) Bulletin RD102T "Evaluation of Long Term Properties of Concrete," Table 6 "Variation of Normalized Concrete Compressive Strength with Age." The hand calculations were performed using a statistical matrix concept from "Statistical Intervals, A Guide for Practitioners," by Hahn and Meeker. Based on the calculations, Energy Northwest determined that the upper bound concrete strength at 20 years would be 6,090 psi.

Holtec Report 2002454 was issued to evaluate a higher compressive strength of the pad concrete to account for the aging factor. The analysis used parameters more representative of the Columbia pad values of 24 inches in thickness and a subgrade effective modulus of elasticity of 16,000 psi. The calculations used the 20 year estimated compressive strength value of 6,100 psi. The calculations determined that the deceleration at the top of the fuel basket would be 39.57g. As such, a compressive strength of 6,100 psi at 20 years was determined to be acceptable for the Columbia ISFSI.

After the concrete pads at the ISFSI had been poured, Energy Northwest's Engineering Change (EC) 1665 "Change for CE-02-00-04 for the Evaluation of Long Term Effect of Concrete Over Strength" dated April 8, 2002, revised the calculations in CE-02-00-04 to account for the actual 28 day break tests for the concrete poured for the two ISFSI pads and the roadway. The roadway was mostly compacted soil and gravel with only a few locations where concrete was used. These included the slab to the east of the doors on the reactor building where the cask is removed from the building and brought outside, the turning pad east of these same doors, the pad over the duct bank, the south railroad crossing, and the north railroad crossing. The criteria for the 28 day break test was less than or equal to 3,800 psi. This proved difficult to do and resulted in several pours exceeding this value. Table 6 provides the average 28 day compressive break test results for the pads and roadway.

Table 6: Break Test Results for Columbia ISFSI at 28 Days

Location	# Samples	Results
South pad	16 samples	3,739 psi
North pad	14 samples	4,135 psi
Roadway slab east of doors	4 samples	4,170 psi
Turning pad east of doors	6 samples	3,905 psi
Roadway pad over duct bank	4 samples	4,168 psi
South railroad crossing	4 samples	3,518 psi
North railroad crossing	6 samples	3,865 psi

Source: Energy Northwest's Engineering Change (EC) 1665 "Change for CE-02-00-04 for the Evaluation of Long Term Effect of Concrete Over Strength" dated April 8, 2002

Even though several results showed average values that exceeded the 3,800 psi limit established by Energy Northwest, none exceeded the 4,200 psi limit established in Table 2.2.9 of the Holtec FSAR. Using the original CE-02-00-04, Revision 0, which correlated a 28 day break test value of 3,800 psi to a 20 year compressive strength value of 6,090, the licensee determined the service life of the concrete. The average break strength value for the pads and the concrete areas on the roadway were compared to the results of the original calculations to determine when the pad aging would result in a concrete strength of 6,090 psi. For the south slab and south railroad crossing, which were 3,739 psi and 3,518 psi respectively, the concrete was considered to have a 20 year service life. However, for the other locations, the service life was reduced. The north railroad crossing (3,865 psi) was 18.8 years. The turning pad east of the doors (3,950 psi) was 18.1 years. The north ISFSI pad (4,135 psi) was 13.75 years. The concrete pad over the duct bank (4,168 psi) and the slab east of the doors (4,170 psi) were 13.1 years.

On July 27, 2010, EC-9354 "Evaluation of Long Term Effect of Concrete Strength Calculation CE-02-00-04" was issued to further address the concrete hardening issue. The evaluation referenced Holtec Report HI-20002454 which had determined that 6,100 psi was acceptable and would result in a deceleration of 39.57g. This calculation took the actual 28 day break strength for the slab at the east of the doors, which was the highest value of 4,170 (only slightly higher than the north pad at 4,135 psi), and applied the predicted strength gain from the PCA Bulletin RD102T. For 20 years, the bulletin showed the concrete would gain in strength by a factor of 1.32 over the value measured at 28 days. The 20 year data included test results from cylinders tested at 20 years, 27 years and 34 years and represented a conservative aging factor for the 20 year value. At 10 years, the gain would be a factor of 1.24. Extrapolating the data at 10 years compared to the 20 year data, Energy Northwest estimated a gain at 30 years by a factor of 1.4 and at 40 years by 1.48. Using these values, EC-9354 determined the 4,170 psi at 28 days for the Columbia ISFSI design would be 6172 psi at 40 years. Since the 6,170 was only slightly above the 6,100 psi and the Holtec calculations of 6,100 psi had a margin of conservatism in it, Energy Northwest assigned the pad and concrete areas of the roadway a service life of 40 years. This premise was based on the

Holtec Report HI-2002454 calculation that showed the 6,100 psi represented 39.57g for the Columbia pad. EC-9354 also included an e-mail from Holtec which referenced Holtec Report HI-2002454 which had determined that a 6,100 psi concrete strength would result in a deceleration during a tipover of 39.57g, which would be below the 45g limit for a tipover. The Holtec e-mail also stated: "There are no regulatory limits or limits imposed in the FSAR on the lifetime compressive strength of the pad."

On November 20, 2009, Condition Report 208137 was issued concerning the need to revise Energy Northwest's "ISFSI 72.212 Evaluation." Section 2.0 "Cask Storage Pad and Area" provided design basis information related to the pad and it's acceptability. The condition report referenced the issue identified in EC-1665 related to the reduced service life of the ISFSI pads and roadway. The evaluation concerning this condition report noted that Revision 1 of the Holtec FSAR had revised Table 2.2.9 and included a new set of parameters for the ISFSI pad, referenced as Set "B." These Set "B" parameters reduced the pad thickness limit to 28 inches or less, the concrete compressive strength to 6,000 psi or less, and the subgrade effective modulus of elasticity to 16,000 psi or less. FSAR Section 3.4.10 "HI-STORM 100 Non-Mechanistic Tipover and Vertical Drop Event (Load Case 02.a and 02.c in Table 3.1.5)" calculated that for this pad with a smaller concrete thickness, the tipover deceleration value at the top of the fuel basket would be 39.91g. Condition Report 208137 stated that there was no lifetime maximum compressive strength limit for the ISFSI pad and that the calculations in EC-9354 showed that the service life of the Columbia pad was 40 years. Revision 6 of the ISFSI 72.212 Evaluation, issued March 2012, incorporated the 40 year design life discussion for the ISFSI pad in Section 2.1.2 "Pad Design Basis."

e. Corrective Action Program

Selected condition reports and operational experience reviews (OER) since the last NRC inspection in 2008 were reviewed. The licensee provided a list of condition reports related to the ISFSI and the reactor building crane from which the NRC inspectors selected a number for further review. These are summarized in the following table. The condition reports were categorized based on significance. Procedure SWP-CAP-01 "Corrective Action Program," Revision 25 defined Severity Level A as significant conditions adverse to quality, Severity Level B as conditions adverse to quality, Severity level C as an event or condition of minor consequences, and Severity Level D as a condition requiring no action assignment. The corrective actions taken for the various events were reviewed and found adequate to resolve the issue. The assigned severity levels were appropriate.

Table 7: Condition Reports Reviewed

Report # and Date	Severity Level	Description
CR 55779 09/24/07	B	The dose to the site boundary was evaluated prior to ISFSI loading based on 13 year old fuel. The analysis needs to be revised to reflect 4.747 year old fuel. [more discussion is provided in the section of this inspection report entitled "Environmental Radiological Monitoring Program"]
CR 179377 03/28/08	B	During downloading of a canister, the top of the canister came into contact with the transfer cask lid causing the crane to trip on overload.

Report # and Date	Severity Level	Description
CR 179813 04/07/08	D	Cask crawler right side track link master pin on one of the tread plates had backed out 1/8 inch. A similar problem had occurred on the left side.
CR 180048 04/13/08	C	After completing placement of the transfer cask in the reactor building, crane movement was lost. Several control relays required replacement.
CR 180076 04/14/08	C	Canisters were being stored outside and not covered. Procedures required the canisters to meet Level C storage requirements of being fully covered if stored outside.
CR 180077 04/14/08	C	Crane overload trip point limits were set at 139 tons instead of 125 tons, which was the maximum load rating of the lift yoke and lifting trunnions. Resolution was to leave the settings at 139 tons since the crane must account for dynamic loads on top of the 125 ton limit. A 15% factor was applied. Applying this to 125 tons equals 143.75 tons. As such the 139 ton trip point was appropriate.
CR 180210 04/16/08	C	During the 2008 NRC ISFSI Inspection, a review of information related to Casks 17 thru 22 indicated the canister heat loads exceeded 21.52 kW which required continuous annulus flushing. However, during loading, annulus flushing had not been performed. Calculation NE-02-08-07 was performed to remove conservatism from the heat load estimates to show actual heat loads at the time of loading were around 21 kW.
CR 180908 05/06/08	B	QA audit finding (Audit Report AU-DC-08) identified inadequate hydrogen monitoring during welding.
CR 181712 05/23/08	C	Cask crawler right side track developed slack in the chain with loss of chain tension during transport of Cask #27. Numerous parts were replaced and the track re-tensioned.
CR 187017 10/06/08 & CR 187201 10/09/08	D	Holtec identified possible fabrication nonconformance with loaded casks. The minimum weld length of a particular joint in the basket allowed by the fabrication drawings was less than allowed by the FSAR drawings. Canisters 93-100 and 162-173 were affected. Condition reports for each canister were issued. Further evaluation found that the welds were acceptable.
CR 190373 12/26/08	C	During review of the cask temperature monitoring data, it was observed that all 27 casks showed a sudden increase in temperature, including Cask 22, which reached 111.4 ºF delta T. An equipment operator was dispatched to check the cask vents and found partial blockage due to snow, which was removed. [more discussion is provided in the section of this inspection report entitled “Temperature Monitoring”]
CR 197487 05/19/09	C	Engineering changes to the cask design made by Holtec were reviewed and determined to not apply to Columbia. Follow-up review determined that three changes were applicable to Columbia and did require review.

Report # and Date	Severity Level	Description
CR 202299 08/04/09	C	During the fabrication of several canisters used at Columbia, Holtec had not performed a helium leak test of the canister. This issue related to Holtec Information Bulletin (HIB)-39. [more discussion is provided in the section of this inspection report entitled "Industry Issues Impacting Energy Northwest's ISFSI Program" under the heading HIB-39]
CR 225795 09/21/10	C	Analysis performed by Holtec indicated that under certain conditions, individual fuel assembly heat limits must be applied during loading in addition to the overall heat limit specified in the Certificate of Compliance. This issue related to Holtec Information Bulletin (HIB)-45. [more discussion is provided in the section of this inspection report entitled "Industry Issues Impacting Energy Northwest's ISFSI Program" under the heading HIB-45]
CR 227436 10/14/10	C	The Fire Hazards Analysis did not reflect current plant configuration. Buildings had been added and removed on the side of the plant where the ISFSI and ISFSI roadway were located including buildings to support the condenser replacement project. None of the changes affected the ISFSI pad area but did affect the roadway. Those buildings that will be in place during the next loading campaign will be added to the Fire Hazards Analysis.
CR 234820 02/25/11	C	This issue related to heat generation rates in the canister during vacuum drying when annulus flushing was not performed. The issue relates to HIB-45 and future cask loading at Columbia. The issue relates to CR 225795 which discussed casks that had already been loaded. [more discussion is provided in the section of this inspection report entitled "Industry Issues Impacting Energy Northwest's ISFSI Program" under the heading HIB-45]
CR 236670 03/29/11	C	NRC Region IV contacted Columbia concerning a generic issue related to seismic evaluations for the stack-up of the HI-TRAC transfer cask on the HI-STORM storage cask that had the potential to affect Columbia. Columbia reviewed the issue and determined that there was no immediate impact. If more specific NRC guidance was issued at a later date, further evaluation would be performed.
CR 236936 04/01/11	C	Operability of plant equipment during a design basis flood was evaluated in response to INPO Event Report (IER) 11-1 concerning short term response to Fukushima accident. The plant and ISFSI are above the worst case flood level.
CR 237071 04/04/11	C	During the evaluation of INPO IER 11-1, the flooding procedure was found to reference the upstream dam break as the worst case flood (design basis flood). The procedure needs revision to reflect that the design basis worst case flood is actually due to extremely high precipitation levels over a short period.

Report # and Date	Severity Level	Description
CR 237608 04/09/11	D	Crane block hoist not functioning and needs repair.
OER 240366 05/11/11	N/A	This Operational Experience Review (OER) reviewed the issues associated with NRC Information Notice 2011-10 and the potential affect on the Columbia dry cask storage program. [more discussion is provided in the section of this inspection report entitled “Industry Issues Impacting Energy Northwest’s ISFSI Program” under the heading IN 2011-10]
CR 248645 09/19/11	D	Three nuts on the auxiliary hoist case were found stripped. The bolts and nuts are not load bearing.
CR 256306 01/20/12	D	Snow and ice observed on lower vents on casks at ISFSI pad and need to be removed.
CR 257639 02/11/12	C	Temperature monitoring system at ISFSI has numerous faults and intermittent issues and is being neglected.
CR 264993 06/07/12	C	Holtec issued Information Bulletin (HIB)-56 concerning the round pins in several ISFSI lifting components including the yoke. Recent computer analysis indicated that the pins may need additional analysis to verify their design rating. Holtec had initiated the new analysis.
CR 265969 06/27/12	C	Holtec identified that several shell sections for the new canisters being fabricated exceeded the hardness requirements in the contract. Expected analysis results would be use-as-is.
OER 267094 07/18/12	N/A	This Operational Experience Review (OER) evaluated the issue related to Holtec Information Bulletin (HIB)-43 concerning when vacuum drying versus forced helium dehydration can be used. If total canister heat load (Q) for the MPC-68 is less than or equal to 23 kW, there is no vacuum drying limit. If Q is greater than 23 kW and less than or equal to 26 kW, then vacuum drying time is limited to 40 hrs. Over 26 kW, vacuum drying is not permitted. Forced helium dehydration must be used. Since all of the casks loaded at Columbia were below the 23 kW, the issue did not apply to the currently loaded casks. The issue also only applied to Amendments 4, 5, and 6 of the Certificate of Compliance. Columbia had loaded their casks to Amendments 1 and 2.

A wide variety of issues were identified in the corrective action program indicating that issues were being entered into the system to provide for tracking of resolution and for trending. No significant trends were found during the review of the condition reports, which covered a four year period.

f. Industry Issues Impacting Energy Northwest's ISFSI Program

(1) NRC Information Notice (IN) 2011-10

NRC's Information Notice (IN) 2011-10 "Thermal Issues Identified during Loading of Spent Fuel Storage Casks" (ADAMS Accession # ML111090200) was distributed to all holders of a Part 72 license on May 2, 2011. The purpose of the notice was to inform the addressees of an incident that occurred during the loading of spent fuel storage canisters at the Byron Generation Station. The NRC expected recipients to review the information for applicability to their facilities and take appropriate actions to avoid similar problems. Bryon, using the HI-STORM 100 system, experienced a canister cooling system malfunction. The circulating water in the annulus between the canister and transfer cask (annulus cooling) used to keep the fuel cladding temperatures below allowable limits, was found to be inoperable after being left unattended during the night shift. The annulus cooling system was required when loading higher kW canisters using the vacuum drying option. The information notice discussed six potential issues related to the incident. Columbia received the information notice and initiated Condition Report 240366 on May 11, 2011, to review applicability of the issues to the Columbia ISFSI and dry cask loading program. The following provides a discussion of the issues identified in the information notice.

- Issue #1: There was no means to prevent or mitigate air ingression into the canister containing fuel, which could cause fuel oxidation, if certain failures of the vacuum drying system occurred, such as a hose rupture. Columbia's Procedure PPM 6.6.7 "MPC Processing", addressed this issue. The procedure required maintenance personnel to constantly monitor and be present during all phases of canister processing including vacuum drying. The procedure contained minimal guidance for responding to an equipment failure, however, maintenance personnel were trained and qualified to respond appropriately to equipment failure. A review of training module IF000016-HO "Vacuum Drying System Student Handout," Revision 2 confirmed that training for vacuum drying abnormalities was included in the operator training.

- Issue #2: Cladding temperatures could exceed technical specification limits if the annulus cooling system was inoperable for an extended period of time. Columbia has not used annulus cooling.

- Issue #3: The Certificate of Compliance technical specifications for vacuum drying were non-conservative for the particular heat load of spent fuel being loaded. This issue was applicable to Columbia and is discussed below in the section concerning Holtec Information Bulletins HIB-45 and HIB-48.

- Issue #4: The Certificate of Compliance, FSAR, and technical specifications did not address the need for a vacuum drying time limit when loading higher heat load (kW) canisters. This issue was applicable to Columbia and is discussed below in the section concerning Holtec Information Bulletin HIB-48.

- Issue #5: The Certificate of Compliance, technical specifications, and FSAR did not address requirements for annulus cooling when the decay heat of an individual fuel assembly reached a certain heat load (kW) limit. This issue was applicable to Columbia and is discussed below in the section concerning Holtec Information Bulletins HIB-45 and HIB-48.

- Issue #6: The FSAR allowed the use of either helium or nitrogen during the canister blow-down operations; however no evaluation was performed by Holtec to justify the use of nitrogen. Columbia used nitrogen for all blowdown operations and as such, this issue was applicable. This is discussed below in HIB-48.

(2) Helium Leak Testing of Canisters during Fabrication (HIB) - 39

The NRC issued a non-cited violation to Holtec International in a letter dated August 5, 2009 (ADAMS Accession # ML092180140) concerning a modification to the Holtec FSAR which eliminated the requirement for the shop helium leak rate test during fabrication of the canister. The leak rate test was designed to demonstrate that the canister shell and baseplate welds were leaktight. The NRC disagreed with Holtec that the leak test could be eliminated and required Holtec to re-instate the leak test. Between the time the FSAR was revised to remove the test until the time the testing was re-initiated, several canisters had been manufactured and sent to reactor sites. One of the sites affected was Columbia. When Energy Northwest became aware of the issue, they initiated Condition Report 202299 to document the problem and track corrective action of the issue. Twelve loaded casks at Columbia were affected from the 2008 loading campaign. These were the 16^{th} through 27^{th} casks loaded. By letter dated October 27, 2010 (ADAMS Accession # ML103070299), Energy Northwest provided information to the NRC related to the twelve affected casks currently stored at the Columbia ISFSI. Energy Northwest stated that a review of environmental radiological data since the casks had been placed on the ISFSI pad showed no indication of any increases in doses that could be related to leaks in the loaded cask welds. The NRC responded to Energy Northwest by letter dated July 7, 2011 (ADAMS Accession # ML111880244) stating that the NRC had reviewed information provided by Holtec and the information provided by Energy Northwest in the October 27, 2010 letter and had determined that the affected casks currently stored at the Columbia ISFSI were acceptable for continued use and no further actions were necessary.

(3) Holtec Information Bulletin (HIB) - 45

Holtec issued a letter to the NRC entitled "Transmittal of Information Regarding Holtec Corrective Action Report #175 Concerning Vacuum Drying of HI-STORM 100 Systems," dated March 7, 2011 (ADAMS Accession # ML110690020). This letter was written to address a Holtec, self-identified concern, regarding specific assumptions used in the thermal analysis supporting vacuum drying for both standing water in the annulus and annulus flushing. This concern was identified by Holtec after review of information related to the NRC inspection of Bryon's equipment failure problem, which later resulted in the issuance of Information Notice 2011-10. Holtec Information Bulletin HIB-45 stated that the thermal models for vacuum drying the Multi-Propose Canister (MPC) used the assumption that the total cask heat load was equally distributed in each of the fuel cell storage locations. This was not specified in the FSAR. Neither the technical specifications nor the FSAR indicated the specific heat limit for each cell location for conditions of vacuum drying with standing water in the annulus or flushing the annulus. Rather, only a limit for the total canister heat load permitted for vacuum drying was given in Technical Specification Appendix A, Table 3-1. As a result, some users were meeting the total canister heat load limit by placing hotter fuel with cooler fuel, unaware that the thermal analysis assumed the heat load of the fuel was equally distributed. Individual fuel assembly heat loads greater than the average limit were outside the FSAR's thermal model. Since vacuum drying was sensitive to cell heat load, the concern was that the peak cladding temperature values under these conditions may not be bounded by the original analysis. Holtec provided its users with HIB-45 to alert them to this concern. Holtec stated that users with loaded canister(s) having individual heat values greater than the heat value assumed for the thermal analysis could submit the canisters' data to Holtec and an analysis would be performed to determine if the peak cladding temperature limits were exceeded.

Columbia had loaded all 27 canisters to less than 21.52 kW choosing not to use the annulus cooling system. Per HIB-45, the assumed heat load (used for the thermal model) for each fuel cell location was limited to 0.316 kW. Columbia had stayed below the 21.52 kW total heat limit for all 27 casks, but had exceeded the 0.316 kW individual fuel assembly limit in five of their already loaded canisters. Fuel assemblies as high as 0.498 kW had been loaded into the five canisters. Pursuant to HIB-45, Columbia provided the heat load data for the five canisters to Holtec for analysis. Columbia also generated Condition Report CR 225795 to document the issue. While in compliance with the technical specifications and FSAR, the five selected Columbia casks had been in an unanalyzed condition during loading operations while undergoing vacuum drying. Holtec provided Columbia with analysis HI-2104751 "Evaluation of BWR Plants Vacuum Drying under the No-Annulus Flush Threshold Heat Load Condition," Revision 0, which concluded that no peak cladding temperature limits had been exceeded for any of the five casks. The Holtec analysis, HI-2104751, stated that using a bounding 24 hour vacuum drying duration, the maximum fuel drying temperature computed based on a bounding Columbia specific cask heat load configuration was 956 ^{o}F, which was below the 1058 ^{o}F peak cladding temperature limit. The bounding vacuum drying heat load scenario described in Table 1 of the HI-2104751 report accurately bounded all of Columbia's five canisters. None of the five canisters had exceeded 23 hours to complete the vacuum drying process.

To avoid future deviations from the original thermal model, Columbia changed their 72.212 Evaluation Report in Section 1.3.2 to state: "If the annular gap between the MPC and HI-TRAC is not continuously flushed with water during vacuum drying operations, no individual fuel bundle shall exceed 316 watts (0.316 kW) unless the total cask heat load is no more than 21.52 kW and a specific analysis has been performed that indicates that the spent fuel peak cladding temperature limits will not be exceeded for the heat load distribution of that cask." Also a change to Procedure PPM 9.6.1 "Spent Fuel Selection for Cask Storage" was in progress to state the same verbiage as Section 1.3.2 of the 72.212 Report. This was being tracked for completed by October 11, 2012, by Condition Report 225795-07. Columbia was pursuing a specific analysis from Holtec to allow greater heat loads than 316 watts for future loading campaigns.

(4) Holtec Information Bulletin (HIB) - 48

Holtec Information Bulletin HIB-48, Revision 1 was issued on April 18, 2011, to the Holtec user group. The bulletin provided information related to vacuum drying. The Holtec information bulletin contained the same heat load discussion as HIB-45 plus other guidance information. Holtec required all users to adhere to the operational practices discussed in the bulletin regardless of which amendment they were loading to. This included the requirement for vacuum drying time limits. In the FSAR, no limitation was placed on the vacuum drying duration if the heat load was sufficiently low. For the MPC-68 canister, this limit was 21.52 kW. A vacuum drying time limit of 40 hours was added to Certificate of Compliance, Amendment 5 for heat loads exceeding 23 kW as a precautionary measure to protect the fuel cladding. This limit addressed an NRC staff concern that cladding temperatures during steady state vacuum drying at the full design basis heat load (28.19 kW for the MPC-68) were too close to the limits specified in Interim Staff Guidance (ISG) – 11 "Cladding Considerations for the Transportation and Storage of Spent Fuel," Revision 3. Users of the MPC-68 canister were required by HIB-48 to implement vacuum drying time limits of ≤ 40 hrs if loading a canister > 23 kW. Columbia did not use annulus flushing during vacuum drying and had not loaded any canisters over the 21.52 kW limit, which would require annulus flushing, or over 23 kW which would limit vacuum drying to 40 hours. Procedure PPM 9.6.1 limited the total cask heat load to 21.52 kW. The licensee stated, if and when Columbia adopts annulus flushing and the procedure is revised to allow total cask heat loads greater than 21.52 kW and if Columbia chooses to exceed total cask heat loads of 23 kW, at that time vacuum drying time limits will be imposed in Procedure PPM 6.6.7 "MPC Processing."

Holtec Information Bulletin HIB-48 also discussed the use of nitrogen to blow-down the canister. The FSAR allows the use of helium or nitrogen during blow-down operations. Holtec acknowledged that the FSAR does not describe or analyze specifically the use of nitrogen in the thermal chapter. However, in all cases, the acceptance criterion for the fuel was that the peak cladding temperature remained below the limits of ISG-11, Revision 3 (1058 ^{o}F) and that the cladding remained in an inert atmosphere. Both of those criteria were met with the use of nitrogen as a blow down gas. Below is a table that was provided in HIB-48 of the peak cladding temperature analytical data for the operational condition of blow-down using helium or nitrogen. For the twenty-seven casks currently loaded at Columbia, all used nitrogen for blow-down and all were below 28.74 kW.

Table 8: Peak Cladding Temperature During Blow-Down

Heat Load (kW)	Condition	Peak Cladding Temperature (^{o}F)	Peak Cladding Temperature Limit (^{o}F)
≤ 28.74	N2 @ 1 atm	894	1058
≤ 28.74	He @ 1 atm	927	1058

Source: Holtec Information Bulletin (HIB)-48, Table on page 4 of 4

(5) Holtec Information Bulletin (HIB) - 53

Holtec Information Bulletin HIB-53 was issued to the Holtec users group on December 6, 2011. The bulletin described an issue that was observed by NRC inspectors at the Waterford nuclear plant (ADAMS Accession # ML12124A387). While Waterford was loading their first canister, operators isolated the canister by closing both the vent and drain port caps during installation of the remote valve operating actuators (RVOAs). Having both port caps closed at the same time isolated the canister while filled with water and spent fuel. This could result in unintentional pressurization of the canister, resulting in a dangerous situation. HIB-53 reminded users that the vent and drain port caps should not be closed simultaneously and that the RVOAs must be installed one at a time in the open position when the canister is filled with water. Columbia initiated Operating Experience Review (OER) 267090 to evaluate the information in HIB-53. The operating experience review stated: At Columbia, Plant Procedure Manual (PPM) 6.6.7 "MPC Processing," controls all closure operations relative to the canister. This includes all lid welding, hydrostatic testing, water blow-down, vacuum drying, and helium backfill operations. Procedure PPM 6.6.7, Revision 22, as written, contains sufficient guidance to preclude inadvertent pressurization caused by having the vent and drain port caps being inappropriately closed or the RVOAs installed at the same time in the closed position. However, an assignment AR00267090-02 was opened to enhance the PPM 6.6.7 procedure to include a caution statement to more strongly reflect the lessons learned identified in HIB-53. The report stated a similar event had not occurred at Columbia.

1.3 Conclusions

The audit and surveillance program implemented by the plant's Quality Services organization reviewed activities associated with the ISFSI including the ongoing temperature monitoring program, radiological surveys of the ISFSI, procedure compliance, records retention, worker training, and implementation of the 10 CFR 72.48 safety evaluation program. The audit and surveillance reports presented good information directed at improving organizational performance and providing management with feedback on the status of the various ISFSI related activities.

Environmental monitoring for the ISFSI included ten thermoluminescent dosimeters (TLD) located on the security fence around the ISFSI. The fence was approximately 40 feet from the nearest casks. Radiation levels for 2011 measured on the ten TLDs ranged from 312 mrem/yr on the east side to 689 mrem/yr on the south side.

A radiological survey along the ISFSI fence found gamma radiation levels ranging from approximately 40 microR/hr (350 mrem/yr) to 80 microR/hr (700 mrem/yr). No measurable neutron radiation levels were found. These levels were consistent with the values measured by the TLDs.

Technical Specification 3.1.2 required air vent temperature surveillances and/or vent inspections to be performed daily on the storage casks to ensure the spent fuel was adequately cooled. Documentation reviewed provided sufficient evidence that the temperature monitoring requirements of Technical Specification 3.1.2 were performed as required.

An issue was identified by the licensee concerning the affect of aging on the ISFSI pad which results in hardening of the concrete. This was an issue because the hardness of the pad must not reach a point where the deceleration affect on the spent fuel during a cask tip over event could exceed 45g. Initial calculations by the licensee suggested that the pad hardness could reach a condition in less than the 20 year license of the ISFSI where the 45g limit could be exceeded. The cask vendor, Holtec Int., provided analysis specific to the Columbia pad that determined that the 45g limit would not be reached. Further evaluation by Entergy Northwest using the Holtec analysis and data developed by the Portland Cement Association determined that the pad service life would exceed 40 years.

Selected condition reports over the past four years were reviewed related to the ISFSI and the reactor building crane. A wide variety of issues had been documented including generic industry issues that could affect the Columbia ISFSI. Corrective actions taken to resolve the issues were found to be adequate. No significant trends were identified by the NRC inspectors from the list of condition reports provided.

Helium leak testing of canisters during fabrication had been discontinued for a period of time by Holtec. Subsequently, the NRC required Holtec to re-initiate the testing on canisters during fabrication. In the interim, a number of canisters had been shipped to licensees including twelve that had been loaded and placed on the Columbia ISFSI pad. NRC review of data provided by Holtec and Energy Northwest concluded that the twelve loaded canisters at Columbia were acceptable for continued use.

Industry related issues identified by the NRC and Holtec technical groups were reviewed for applicability to Columbia's ISFSI program and facility. The licensee adequately incorporated resolutions to the issues identified in NRC Information Notice 2011-10 and Holtec Information Bulletins (HIB) 45, 48, and 53 into their program.

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

2.1 Inspection Scope

The 72.212 Evaluation Report was reviewed to verify site characteristics were still bounded by the Holtec HI-STORM 100 design basis.

2.2 Observations and Findings

The current version of the 72.212 Evaluation Report was Revision 6 (March 2012). Revision 5 had been reviewed during the June 17, 2008, NRC inspection and included in Inspection Report 72-35/08-001 (ADAMS Accession # ML081690841). Changes incorporated into Revision 6 included information related to Holtec Supplier Manufacturing Deficiency Report (SMDR) 1025-1712, Revision 1 concerning the number of bolts on the HI-TRAC transfer cask required to be installed during loading and unloading operations; information related to Holtec Information Bulletins (HIB) 45 and 48 concerning decay heat limits for individual spent fuel assemblies, vacuum drying requirements, cooling requirements for the annulus; and the design life issue related to the ISFSI pad concrete hardening over time.

2.3 Conclusions

The 72.212 Evaluation Report was being kept current and had included industry issues that affect the Columbia site, even though no new loading campaigns had been initiated since 2008.

3 Review of 10 CFR 72.48 Evaluations (60857)

3.1 Inspection Scope

The licensee's 72.48 screenings and evaluations since the 2008 NRC inspection were reviewed to determine compliance with regulatory requirements.

3.2 Observations and Findings

Procedure SWP-LIC-02 "Licensing Basis Impact Determinations," Revision 11 defined the licensee's 72.48 and 50.59 safety evaluation process. This included performing an applicability determination review as a pre-screening to determine if the issue required a 72.48/50.59 screening. If a screening was required, the screening evaluation reviewed the document to determine if a 72.48/50.59 evaluation was required. If an evaluation was required, the evaluation determined if the change or activity could proceed without prior NRC approval.

Applicability determinations issued since the last NRC inspection in 2008 were reviewed related to the ISFSI. Thirty-seven had been issued. The review evaluated whether issues were being processed through the applicability determination process correctly and moving to the screening level as appropriate. Of the 37 applicability determinations reviewed, twelve involved administrative or editorial changes, one corrected errors made during incorporation of previous changes, one change corrected a non-conforming issue related to combustible gas monitoring, and the remaining applicability determinations moved to the screening level. No examples of issues that should have been screened but were not, were identified during the review of the applicability determinations.

Since 2008 the licensee had conducted one hundred and sixty-one 72.48 screenings. One hundred and thirty one of those screenings were performed in 2009. The majority of screenings performed during the four year period were reviews of Holtec's Supplier Manufacturing Deviation Reports (SMDRs) and Holtec's Engineering Change Orders (ECOs). These documents were supplied by Holtec as a package associated with a particular canister or HI-STORM storage cask when shipped to the site. The documentation described any change or deviation from the original design of the equipment. The SMDRs and Holtec's ECOs had been previously review by Holtec and found acceptable. The Columbia 72.48 screenings also found that the SMDRs and ECOs did not identify any issues inconsistent with the programs being implemented at Columbia.

The NRC selected ten screenings for additional review. None of the screening required a full safety evaluation. The issues discussed in the screenings are summarized in the following table. All screenings were determined to be adequately evaluated.

Table 9: Safety Screening

Screen # and Date	Description
08-0003 05/15/08	A revision to Procedure PPM 6.6.6 “MPC Fuel Loading” changed the amount of water removed from the canister prior to removing the canister from the spent fuel pool. An additional 30 gallons was drained to provide additional space below the bottom surface of the canister lid during welding operations. More freeboard space limited hydrogen generation and assured that the weld zone for the lid-to-shell weld would remain well clear of water intrusion during the heat-up of the water caused by the decay heat of the spent fuel. Discussions with the licensee determined that the change, which now resulted in removing 90 to 100 gallons from the canister in Procedure PPM 6.6.6, Step 7.6.30, still left water covering the top of the fuel to preclude oxidation.
09-0017 09/11/09	Holtec SMDR 1021-895 documented that the material procured to support the fabrication of the canister’s vent/drain cover plates and closure rings were purchased to the wrong ASME code subsection with a less stringent quality class. The material was re-evaluated and ultrasonic tested to meet the correct quality assurance procurement and non-destructive examination testing requirements. The material was accepted “as is” and was documented as capable of performing its design function. This issue applied to nine casks used at Columbia.
09-0029 09/09/09	Holtec SMDR 1021-1085 documented the disposition and resolution of a fabrication issue relative to the roundness of the lid openings of some canisters. The canister openings were reworked to achieve sufficient roundness to allow lid insertion. This applied to four canisters provided by Holtec to Columbia.
09-0032 09/09/09 and 09-0060 11/04/09	Both screenings addressed Holtec SMDR 1021-1516 and SMDR 1021-1519 that documented the replacement of canister vent and drain port caps made of SA 479 Type 304 material with caps made of Nitronic 60. The change in the material had not impact the design function of the caps. This change applied to four canisters loaded at Columbia.
09-0041 09/11/09	Holtec SMDR 1021-165 documented the accept-as-is disposition for two types of non-conformances in the Boral panels received and utilized in the fabrication of some canisters. The Boral panels were outside the design tolerances, but were evaluated to meet the design function to limit reactivity. Holtec accepted the product as is. This applied to seven different canisters used at Columbia.
09-0130 11/13/09	Holtec ECO 1024-137 documented numerous small drawing changes affecting several HI-STORM storage casks. The drawing changes included reducing the size of non-structural welds and changing weld styles (fillet to groove). Holtec evaluated and approved the change. This applied to eight HI-STORMs used at Columbia.

10-0002 12/06/10	The hydrogen gas tube trailer and gas cylinder configuration used by the hydrogen gas vendor (who supplies hydrogen to Columbia) had changed from those used previously. Since there was no change in the hydrogen storage location, no increase in individual hydrogen gas cylinder size, and the change did not involve any increase in hydrogen storage capacity or quantities, there was no increase in fire or explosion hazard and no adverse effects due to the change.
12-0001 01/04/12	The purpose of this screen was to provide additional evaluation information relative to the Holtec Earthquake Response Mitigator (HERMIT). The purpose of the HERMIT was to minimize the transfer of seismic induced forces to sufficiently minimize cask rocking and lateral motion to preclude tip-over during the cask stack-up and transfer of the canister from the HI-TRAC transfer cask into the HI-STORM storage cask. This was previously evaluated in screen 02-0010.
12-0002 01/19/12	This screening evaluated Revision 6 to the 72.212 Evaluation Report to reflect new information regarding the useful life of Columbia's ISFSI concrete pads and maximum fuel bundle heat limitations relative to notifications from Holtec in HIB-45 and HIB-48.

Source: Screenings Selected by NRC from a List Provided by the Licensee

As a result of the applicability determination reviews and the safety screening reviews of issues identified since the last NRC inspection in 2008, two issues were identified as requiring 72.48 evaluations. Evaluation 7248-07-0001, Revision 0 related to the cooling time of the casks that were loaded versus the assumption for the cooling time used in the offsite dose calculations. This issue related to Condition Report 55779. The ISFSI site boundary dose was computed prior to the original ISFSI loading campaign using fuel with a cooling time of 13 years or greater. However, almost all the fuel loaded in the 2002 and 2004 cask loading campaigns (15 casks) had been cooled for 5.7 to 10 years rather than the 13 years. Evaluation 7248-07-0001 reviewed the change from 13 years to 5 years for the cooling period against the criteria of 10 CFR 72.48 and determined that the screening criteria were met and no change to the license or technical specifications was required. Additional dose modeling was performed and determined that the offsite dose at the site boundary was basically unchanged.

Evaluation 7248-07-002 related to the spent fuel with cooling times of 4.747 years and the need to use this value in the site boundary dose calculations. This 72.48 evaluation was reviewed during the 2008 inspection issued as Inspection Report 72-35/08-001 on June 17, 2008, (ADAMS Accession # ML 081690841).

3.3 Conclusions

All 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. Two safety evaluations had been performed since 2008. Both evaluations were in compliance with regulatory requirements.

4 Follow-up on Corrective Actions for Violations and Deviations (92702)

<u>(Closed) NOV 72-35/0801-01 Failure to Monitor the Combustible Gas Concentration While Performing the MPC Lid-to-Shell Weld.</u> On June 17, 2008, the NRC issued Inspection Report 72-35/08-001 (ADAMS Accession # ML081690841) which included Notice of Violation (NOV) 72-35/0801-01. On July 16, 2008, Energy Northwest responded to the NRC via letter (ADAMS Accession # ML082250259) providing a discussion of the cause of the event and corrective actions to prevent recurrence. The primary corrective action was to revise procedures to strengthen and add instructions for the hydrogen monitoring during cask welding. On August 6, 2008, the NRC responded to Energy Northwest (ADAMS Accession # ML082190553) stating that the NRC found the July 16, 2008 letter response to the issue identified in the Notice of Violation. The NRC also noted that during the loading of Canisters 169 and 170, the NRC Resident Inspector observed the welding operations, which occurred without further incident.

5 Exit Meeting

The inspector reviewed the scope and findings of the inspection during an exit conducted on July 25, 2012.

ATTACHMENT 1:

SUPPLEMENTAL INSPECTION INFORMATION

PARTIAL LIST OF PERSONES CONTACTED

Licensee Personnel

R. Fuller, ISFSI Project Manager
T. Jones, Radiation Protection Technician
C. Moore, Reactor and Major Maintenance Supervisor
M. Rowe, Radiation Protection Craft Supervisor
R. Sanker, Radiation Operations Supervisor
D. Suarez, Licensing Engineer
L. Woosley, Principal Engineer, Safety Analysis

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
IP 92702	Follow-up on Corrective Actions for Violations and Deviations

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Discussed

None

Closed

72-35/0801-01	NOV	Failure to monitor the combustible gas concentration while performing the MPC Lid-to-shell weld

LIST OF ACRONYMS

ANSI	American National Standards Institute
BWR	boiling water reactor
CFR	Code of Federal Regulations
CoC	Certificate of Compliance
CR	condition report
DCS	dry cask storage
Delta T	delta temperature

EC	engineering change
ECO	Holtec's engineering change orders
F	Fahrenheit
FSAR	Final Safety Analysis Report
HIB	Holtec information bulletin
IER	INPO event report
INPO	Institute of Nuclear Power Operations
IP	inspection procedure
ISFSI	Independent Spent Fuel Storage Installation
kW	killo-watt
MPC	multi-purpose canister
mR	milliRoentgen
micro(μ)R	microRoentgen
MPC	multipurpose canister
mrem	milliRoentgen equivalent man
MWD/MTU	megawatt days/metric ton uranium
NOV	notice of violation
NRC	Nuclear Regulatory Commission
OER	operational experience review
PCA	Portland Cement Association
PPM	plant procedure manual
psi	pounds per square inch
psig	pounds per square inch gauge
RVOA	remote valve operating actuators
SMDR	Holtec's supplier manufacturer deviation reports
TLD	thermoluminescent dosimeter
WNP	Washington Nuclear Project
Zr	zircalloy

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

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 individual “initial” enrichment per cent of U-235

Casks 1 – 15 were loaded under Amendment 1 of the Certificate of Compliance and Revision 1 of the Holtec FSAR
Casks 16 – 27 were loaded under Amendment 2 of the Certificate of Compliance and Revision 4 of the Holtec FSAR