QA: N/A

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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ARLINGTON, TEXAS 76011-4005

April 23, 2004

Mr. J. V. Parrish
Chief Executive Officer
Energy Northwest
P.O. Box 968; MD 1023
Richland, Washington 99352-0968

SUBJECT: NRC INSPECTION REPORT 050-00397/04-007; 072-00035/04-001

Dear Mr. Parrish:

On March 31, 2004, the NRC completed an inspection at your Columbia Generating Station Independent Spent Fuel Storage Installation (ISFSI). The enclosed report presents the results of that inspection.

The inspection included reviews of changes made to your ISFSI operations and procedures, safety evaluations conducted under 10 CFR 72.48, radiological control practices, spent fuel selection and loading documentation, technical specification surveillances, personnel training and certification records, problem reporting and resolution, quality assurance audits, and closure of one open unresolved item. 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 and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

D. Blair Spitzberg, Ph.D., Chief
Fuel Cycle and Decommissioning Branch

Docket Nos.: 050-00397; 072-00035
License No.: NPF-21

Enclosure:
NRC Inspection Report
050-00397/04-007; 072-00035/04-001

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket Nos.:	050-00397; 072-00035
License No.:	NPF-21
Report No.:	050-00397/04-007; 072-00035/04-001
Licensee:	Energy Northwest
Facility:	Columbia Generating Station
Location:	Richland, Washington
Date:	March 30-31, 2004
Inspector:	J. Vincent Everett, Senior Health Physicist
Accompanied by:	Scott P. Atwater, Health Physics Inspector-In-Training Lynn Albin, Radiation Physicist Washington State Department of Health
Approved by:	D. Blair Spitzberg, Ph.D., Chief Fuel Cycle and Decommissioning Branch
Attachments:	Supplemental Inspection Information
ADAMS Entry:	IR05000397-04-07; 07200035-04-01; on 04/30/04;Columbia Generating Station ISFSI Report.

EXECUTIVE SUMMARY

Columbia Generating Station
NRC Inspection Report0 50-00397/04-007; 072-00035/04-001

An inspection of the Columbia Generating Station Independent Spent Fuel Storage Installation (ISFSI) facility in Richland, Washington, was conducted on March 30-31, 2004. This was a routine operational inspection of ISFSI activities to verify that the ISFSI was being maintained in conformance with the commitments and requirements contained in the Facility Safety Analysis Report (FSAR), Certificate of Compliance (CoC), Technical Specifications, Quality Assurance (QA) Program and 10 CFR Part 72. Columbia is using the Hi-Storm 100S cask storage system and had completed their initial loading campaign with casks 1-5 in December of 2002. During this inspection Columbia was conducting their second loading campaign with casks 6-15. The inspectors observed the transfer of the 11th spent fuel canister into its storage cask and movement of the storage cask out to the ISFSI pad. Ms. Lynn Albin, Radiation Physicist from the State of Washington Department of Health, was also onsite performing a state review of ISFSI activities and observed this inspection.

Operation of an ISFSI

- Changes made to the ISFSI operations and procedures had been appropriately screened through the 10 CFR 72.48 process and were bounded by the licensing basis documents (Section 1.2.a).

- The personnel exposure data for casks 1-5 was very similar to casks 6-10, even though casks 6-10 were considerably hotter. The loading campaigns reflected a learning curve with decreasing exposures as experience was gained. The ALARA and radiological control practices for the ISFSI operation were effective in minimizing personnel exposure (Section 1.2.b).

- The thermo-luminescent dosimeters (TLDs) located on the ISFSI area fence provided adequate representation of the ISFSI radiation levels. The environmental TLD data and licensee surveys were consistent with the survey data obtained independently by the NRC inspectors (Section 1.2.b).

- After revising its procedure for conducting contamination surveys of the canister, the licensee was meeting the Technical Specification requirements for removable contamination on the exterior surfaces of the canister (Section 1.2.c).

- The fuel selected for loading the 6th and 11th canisters was verified to meet the requirements for approved contents. The documentation package for the loaded spent fuel canister selected for review contained the required information (Section 1.2.d).

- Loaded Hi-Storm cask temperatures and/or air inlets and outlets were being monitored as required by Technical Specifications. The loaded cask surface dose rates were documented as being maintained within the limits of the Technical Specifications (Section 1.2.e).

- Training and certification records for selected personnel assigned to the ISFSI project were reviewed. The licensee had sufficiently trained and certified their ISFSI personnel to meet the applicable requirements (Section 1.2.f).

- The licensee's use of the problem evaluation request (PER) system to track problems related to the ISFSI provided an effective system to document, evaluate and determine corrective actions for the various issues being identified for the ISFSI related work (Section 1.2.g).

Review of 72.212 Evaluations

- Since the previous inspection, no changes had been made to the licensee's evaluation of conditions of a general license performed pursuant to 10 CFR 72.212.

Unresolved Items

- On November 1, 2002, the NRC issued Inspection Report 50-397/02-08;72-35/02-01 which included Unresolved Item (URI) 72-35/0201-01 related to the observation of hydrogen bubbles being generated in a Holtec canister placed in the Columbia spent fuel pool. As a result of the reviews conducted by the Spent Fuel Project Office, and an inspection of the vendor, a violation related to this issue was issued to Holtec, Inc. The violation addressed Holtec's failure to provide for adequate design measures to ensure compatibility of the Hi-Storm materials as required by 10 CFR 72.146 "Design Controls." No further follow-up is required related to this issue at Columbia. Unresolved Item 72-35/0201-01 has been closed (Section 3).

Report Details

Summary of Facility Status

Columbia Generating Station is a General Electric boiling water reactor owned by Energy Northwest. The facility is located approximately 12 miles northwest of Richland, Washington, on the Department of Energy's Hanford Reservation. Energy Northwest had selected the Holtec HI-STORM 100S cask system as the storage system for their spent fuel. The licensee was loading spent fuel under a general license to Certificate Of Compliance 1014, Amendment 1. The initial Independent Spent Fuel Storage Installation (ISFSI) loading campaign was completed in December 2002 and consisted of five casks. The second loading campaign was in progress at the time of this inspection and consisted of 10 more casks. During this inspection, the licensee was in the process of placing the 11th cask in the ISFSI.

1 Operation of an ISFSI (60855)

1.1 Inspection Scope

An inspection of the Columbia Generating Station was conducted to verify the licensee was operating the ISFSI in conformance with the commitments and requirements contained in the Facility Safety Analysis Report (FSAR), Certificate of Compliance (CoC), Technical Specifications, Quality Assurance (QA) Program and 10 CFR Part 72.

1.2 Observations and Findings

a. Changes to the ISFSI Operation and Procedures

The licensee had submitted their 2003 Annual Operating Report to the NRC in accordance with Part 50 Technical Specification 5.6.1. Section 8.0 of the report titled, "10 CFR 72.48 Changes, Tests, and Experiments," indicated no activities were conducted during 2003 requiring reporting pursuant to the 10 CFR 72.48 requirements.

Although no safety evaluations were performed, there were 16 safety screenings, of which 4 involved changes to the ISFSI operation. The four ISFSI changes were related to additional passivation of the canister internals (Screen 02-0016), nitrogen purging of the canister during drying (Screen 02-0017), and the use of de-icers on the ISFSI pad during freezing conditions (Screens 03-0005 and 03-0008). These changes to the ISFSI operation were reviewed and found to be bounded by the license basis documents. The changes had been appropriately screened.

The licensee had made a significant number of revisions to their ISFSI loading procedures based on lessons learned from the initial loading campaign. Changes made to the following procedures were reviewed in detail and found to be bounded by the licensing basis documents:

- Procedure 6.6.4, "HI-STORM System Site Transportation."
- Procedure 6.6.5 "Movement and Transfer Operations of HI-TRAC and HI-STORM in the Reactor Building"
- Procedure 6.6.6 "MPC Fuel Loading"
- Procedure 6.6.7 "MPC Processing"

b. Radiological Controls

The licensee was conducting this loading campaign using the following six radiation work permits (RWP):

- 3000-1175 Ancillary Equipment Maintenance
- 3000-1176 Install/Remove MPC HI-TRAC - Fuel Handling
- 3000-1177 MPC Sealing Activities
- 3000-1178 Transfer MPC From HI-TRAC to HI-STORM
- 3000-1179 Dry Cask (HI-STORM) Transport and Storage
- 3000-1180 HI-TRAC Decon Activities

The RWPs were reviewed. The exposure and contamination control measures were found to be commensurate with the operations covered. As of the date of this inspection, the licensee had not encountered any hot particle or airborne activity problems.

The radiological exposures received by the workers for the loading of the casks were being tracked by the licensee. This data was reviewed to evaluate the licensee's ALARA efforts for the first 10 casks that had been loaded. The first loading campaign involved the loading of five casks and was completed in December 2002. A 14-month period elapsed between the first and second loading campaign, during which a significant turnover of ISFSI personnel occurred. Approximately 75 percent of the personnel working on the second campaign were new to the ISFSI project. During the first loading campaign, the licensee recorded a total dose of 0.385 person-rem to load the first cask and place it into the ISFSI. As the other four casks were loaded, efficiencies developed and personnel exposures decreased. The total dose recorded for the fifth cask was 0.245 person-rem. When the second campaign started, the doses recorded for the first cask loaded (i.e. sixth cask) were 0.389 person-rem. This was consistent with the doses received for the first cask loaded during the first loading campaign and reflected the new staff for the ISFSI project. As work continued toward the loading of the 10th cask, the staff was again able to develop efficiencies in their radiological work process, eventually reaching a dose of 0.276 person-rem for the loading of the 10th cask. This was achieved while working on significantly hotter casks averaging 17 kW compared to the first loading campaign where the casks were averaging 11-12 kW. Attachment 2 to this inspection report provides a summary of the person-rem doses per cask.

The licensee had submitted their 2003 Radioactive Effluent Release Report to the NRC in accordance with 10 CFR 72.44(d)(3) and Part 72 Technical Specification 5.4. No new casks were placed on the ISFSI pad during 2003. The licensee monitored the environmental radiation levels around the ISFSI with approximately 10 TLDs placed on the outer ISFSI security fence. Data from the TLDs was reviewed for the second quarter of 2002. This time frame was prior to placement of any casks on the ISFSI pad and reflected only shine from the reactor facility plus natural background. Background levels recorded on the 10 TLDs ranged from 0.274 mrem/day (11 microR/hr) to 0.330 mrem/day (14 microR/hr). The TLD results from the fourth quarter 2003, which reflected five casks stored at the ISFSI, ranged from 0.369 mrem/day (15 microR/hr) to 0.669 mrem/day (28 microR/hr). During this inspection, a radiological survey of the perimeter of the ISFSI along the boundary of the outer fence was conducted by the NRC inspectors using a microR meter. Readings were taken at the various TLD stations plus any areas between the TLDs that gave higher readings. Readings ranged from 36 microR/hr to 115 microR/hr. These readings reflected the radiation levels associated with 11 casks now located at the ISFSI. Higher readings were directly related to being in-line with the vents on the storage casks. The highest reading during the NRC survey of 115 microR/hr was between two TLD locations, with the nearest TLD location (#124) reading 110 microR/hr. Based on the NRC's independent radiological surveys, the TLD placements were determined to provide adequate representation of the ISFSI radiation levels at the fenced boundary for the casks loaded to date.

A survey was conducted of the casks on the ISFSI pad. A contact reading for the 11th cask, which was one of the hotter casks at 17 kW, found a dose rate of 2.8 mR/hr on contact with the vent screen. Moving to the side of the screen, the reading dropped to 0.95 mR/hr on contact with the concrete side of the cask. The licensee had properly posted the inner ISFSI fence as a radiologically controlled area, radiation area and radioactive materials area.

c. <u>ISFSI Operations</u>

The licensee was required by Section 8 of the Final Safety Analysis Report to have procedures for the unloading of a canister, should removal of the spent fuel become necessary. FSAR Section 8.3.3, Step 7, required sampling of the gas in the canister prior to removal of the lid to determine if any damage to the fuel cladding had occurred during storage. The licensee had provided instructions in Section 7.8 of Procedure 6.6.9, "MPC Cooldown and Weld Removal Systems," Revision 0, for gas sampling of a canister prior to lid removal. The procedure provided a schematic showing the proper connection of the gas sample bottle and included detailed valving instructions for taking the sample. Step 7.8.7 of the procedure required the sample to be sent to chemistry for isotopic radioactivity analysis to identify any gases that could indicate failure of the fuel cladding. Step 7.8.10 required authorization from health physics before venting the gases from the canister. The procedure included adequate instructions for performing the gas sampling of the canister.

Technical Specification 3.2.2 required that removable contamination on the exterior surfaces of the transfer cask and accessible portions of the canister not exceed 1,000 dpm/100 cm^2 beta gamma contamination and 20 dpm/100 cm^2 alpha contamination. Surveillance Requirement 3.2.2.1 stated, "Verify that the removable contamination on the exterior surfaces of the transfer cask and accessible portions of the canister containing fuel is within limits." Columbia was performing the required survey of the canister lid and the upper portions of the canister down to several inches below where the annulus seal had been installed. This survey was being performed after these areas were decontaminated. Columbia was also taking a sample of the water in the annulus gap below the seal to verify contamination levels were below 1×10^{-6} μCi/ml.

Chapter 12 of the Final Safety Analysis Report provided the basis for performing the survey and discussed the need to verify that loose contamination would not be spread to the ISFSI. However, the basis statements were not clear as to how the survey of the accessible portion of the canister would be related to the contamination levels on the inaccessible portions of the canister. The basis statement in Section B.3.2.2 of Chapter 12 in the Limiting Condition of Operation (LCO) section could be interpreted to only apply the contamination limits to the accessible portions of the upper circumferential portion of the canister. The LCO states that, "The objective is to determine a removable contamination value representative of the entire upper circumference of the MPC." In this case, decontamination prior to surveying in order to ensure that any contamination on the upper portion of the canister is below the technical specification limit could be considered appropriate. Since the annulus seal was used to prevent contaminated spent fuel pool water from reaching the majority of the canister, one could assume that only the upper accessible portions of the canister that were in contact with the spent fuel pool water may have become contaminated and needed to be cleaned and surveyed prior to movement to the ISFSI.

However, past experience at other sites loading canisters and the experience at Columbia with the first canister had shown that contamination has occasionally been found under the annulus seal area. For this reason, surveying the area under the annulus seal after the water is removed and before being decontaminated would provide reasonable verification that the inaccessible portions of the canister were also within the limits of the technical specification. Columbia issued Condition Report (CR) 2-04-01121 to revise their procedure to require a contamination smear below the annulus seal prior to decontamination.

d. <u>Fuel Selection and Loading Documentation</u>

Certificate of Compliance 1014, Amendment 1, Appendix B, Section 2.0, "Approved Contents," provided fuel parameter requirements for the MPC-68 canister. These requirements were compared to the fuel placed in the 6th and the 11th canisters. The sixth canister had a heat load of 12.0 kW with spent fuel cooling times ranging from 7.8 years to 13.7 years. Table 2.1-5 of Appendix B for uniformly loaded canisters limited the maximum decay heat per assembly to 350 watts for a cooling time of $\geq$13 years and 363 watts for a cooling time of $\geq$7 years. The hottest spent fuel assembly placed in the 6th canister was 241.9 watts.

The licensee switched to a regionalized fuel loading concept with the 7th canister. This concept was allowed by Technical Specification 2.1 and provided for placement of a limited number of hotter fuel assemblies in the inner storage locations of the canister.

The 11th canister had been loaded using the regionalized concept. This canister had a heat load of 17.1 kW with spent fuel cooling times ranging from 5.7 years to 13.7 years. Table 2.1-7 of Appendix B for regionally loaded canisters limited the maximum decay heat per assembly for spent fuel placed in Region 1 to 397 watts for a cooling time of ≥13 years and 500 watts for a cooling time of ≥5 years. The hottest assembly placed in Region 1 of the 11th canister was 340.4 watts. No failed fuel was placed in either of the canisters.

Fuel enrichment for the spent fuel assemblies stored in the 6th canister ranged from 2.19 to 2.72 percent. For the 11th canister, enrichment ranged from 1.76 to 2.92 percent. Table 2.1-3 of Appendix B to the technical specifications for the MPC-68 canister for the types of arrays/classes of fuel at Columbia limited the enrichment to no more than 5 percent.

The maximum assembly burnup of the spent fuel stored in the 6th canister was 32,318 megawatt days/metric ton of Uranium (MWD/MTU). Table 2.1-4 of Appendix B limited the burnup for a uniformly loaded canister with spent fuel of 7 years or more cooling time to a maximum burnup of 42,300 MWD/MTU. For the 11th canister, Table 2.1-6 provided limits for the two loading regions in the canister. The licensee loaded spent fuel with a minimum cooling time of 5.7 years and a maximum burnup of 38,729MWD/MTU into Region 1. Table 2.1-6 limits fuel with this cooling time to 45,100 MWD/MTU. For Region 2, the licensee loaded spent fuel with a minimum cooling time of 9.7 years and a maximum burnup of 33,710 MWD/MTU. Table 2.1-6 limits fuel with this cooling time to 37,600 MWD/MTU. Attachment 2 to this report provides a summary of the characteristics of the spent fuel that had been loaded at the time of the inspection.

10 CFR 72.234(d)(2) "Conditions of Approval," required the licensee to include the following information in each loaded spent fuel canister documentation package: 1) NRC CoC number; 2) spent fuel storage cask model number; 3) spent fuel storage cask identification number; 4) fabrication certifications; 5) inspection certifications; and 6) name and address of the licensee using the spent fuel storage cask. The documentation package for MPC-006, Serial No. 091 was selected for review and found to be complete. The licensee had met the requirements of 10 CFR 72.234(d)(2) for this canister.

e. Technical Specification Surveillances

Technical Specification 3.1.2, "Heat Removal System," required the licensee to check for blockage of the inlet and outlet air ducts on the loaded casks every 24 hours or provide a check of the air outlet temperature versus ambient temperature every 24 hours to verify the difference was 126°F or less. The licensee's Procedure OSP-SFS-D101, "Spent Fuel Storage Cask Heat Removal System Daily Checks," Revision 3, provided instructions for performing the daily temperature surveillances of the casks in compliance with Technical Specification 3.1.2. Procedure OSP-SFS-D101 provided instructions to perform temperature verification using the installed temperature monitoring system with an acceptance value of not more than 114°F, which was more conservative than the technical specification limit of 126°F. If the monitoring system was not available, visual monitoring of the vents was required. If visual monitoring was implemented, Procedure OSP-SFS-D101, Attachment 9.2, "SFSC Ventilation Duct Inspection," provided a form to log the surveillance of the four inlet and four outlet vents to verify they were free from blockage on a daily basis. A review was completed of selected records for the period of January through March 7, 2004. The licensee had used a combination of the two methods for monitoring temperatures on the casks. On January 3, 2004, partial ice blockage was noted on three casks. During periods in January 2004 when temperature monitoring was used, the differential temperature values were typically 30°F to 60°F. The licensee was meeting the requirements of Technical Specification 3.1.2, "Heat Removal System," for temperature monitoring of the casks at the ISFSI.

Technical Specification 3.2.3, "Overpack Average Surface Dose Rates," established limits for dose rates from the casks during storage. The limits were 50 mrem/hr on the sides, 10 mrem/hr on the top and 45 mrem/hr at the inlet and outlet vents. These limits applied to the total gamma plus neutron doses. The licensee performed the required surveillances in accordance with Procedure HSP-SFS-C103, "Overpack Average Surface Dose Rates," Revision 2. The data collected by the licensee for Hi-Storm casks #120, #121 and #122 was reviewed. These were the 7^{th}, 8^{th} and 9^{th} casks loaded and all were the hotter casks with heat loads of 17.0 kW and 17.1 kW. Attachment 9.2 of the procedure provided a table for collecting dose rate readings at several locations on the cask and provided for averaging of the dose rates to compare against the technical specification limits. For casks #120 and #121, the dose rates on the side of the casks were less than 1-mrem/hr at all locations. On cask #122, a gamma dose averaging 0.8 mrem/hr was calculated with the highest reading found on the cask side of 1.5 mrem/hr. On the cask lids, all casks measured less than 1-mrem/hr at all points. At the vent locations, the highest reading obtained on the three casks was 2-mrem/hr gamma. No neutron doses were measured. The readings with the highest levels were all on the lower vents. The calculated average dose rates for the vents for the three casks were 0.15 mrem/hr, 0.56 mrem/h and 0.93 mrem/hr. A radiological survey performed by the NRC on the 11^{th} cask immediately after being placed on the ISFSI pad, found gamma radiation levels of 2.8 mR/hr using a microR meter at one of the lower vents. This level was consistent with the licensee's measurements. The licensee was maintaining storage cask dose rates within the requirements of Technical Specification 3.2.3, "Overpack Average Surface Dose Rates".

Technical Specification 3.1.1, Table 3-1, required the following conditions to be met during Vacuum Drying and Helium Backfill: 1) dryness of 3 torr or less for 30 minutes or more; 2) helium backfill 0.1218 +0/-10 percent g-moles/l; 3) helium leak rate less than or equal to 5.0 X 10^{-6} atm cc/sec (He); and 4) helium purity 99.995 percent or higher. The documentation package for MPC-006, Serial No. 091, was selected for review. A dryness of 1.9 torr was held for 30 minutes on February 22, 2004, and documented in Procedure 6.6.7, "MPC Processing", Revision 7, step 7.7.22. The backfill volume of 0.1218 +0/-10 percent g-moles was achieved on February 23, 2004, and documented in Procedure 6.6.7, "MPC Processing", Revision 7, step 7.9.5, and in Attachments 9.6 and 9.7. The helium leak rate testing was performed in accordance with Procedure COGEMA-SVLT-TNS-003.18, Multi-Purpose Canister (MPC) Helium Leak Test," Revision 1. A helium purity of 99.995 percent and "no detectable helium leakage" through the MPC lid-to-shell weld was documented in the MPC Lid-to-Shell Weld Leak Test Report dated February 20, 2004. A Varian 979 detector probe with a sensitivity of 5.1 X 10^{-10} atm cc/sec (He) sensitivity was used. "No detectable helium leakage" through the vent and drain port cover plate welds was documented in the MPC Vent/Drain Cover Port Plate Welds Report dated February 23, 2004. A detector with a sensitivity of 1.2 X 10^{-10} atm cc/sec (He) sensitivity was used. The licensee had met the vacuum drying and helium backfill requirements of Technical Specification 3.1.1, Table 3-1, for MPC-006, Serial No. 091.

f. ISFSI Personnel Training and Certification

10 CFR 72.190 Subpart I, "Training and Certification of Personnel," required the licensee to train and certify all ISFSI personnel operating, or directly supervising the operation of, equipment and controls identified as important to safety. The licensee had organized the ISFSI operation into four ISFSI crews, each one consisting of one supervisor and seven technicians. Additionally, two managers were assigned to the project bringing the total ISFSI staff to 34. Training records for members of the ISFSI crews and management were selected and reviewed. The qualification requirements were specified in the Qualification Directory, Section 2.11, Revision 0. This directory contained 12 technician qualification cards and 1 supervisor qualification card. The technician qualification cards contained classroom instruction and on-the-job (OJT) performance for each phase of the operation. The supervisor qualification card contained classroom training. In addition, all personnel were required to have completed protected area access training, radiation worker training, and a medical examination within the past year. Each crew had two fully qualified supervisors and one technician fully qualified to perform all phases of the operation. The other six technicians on each crew were qualified to perform one or more of the phases of the loading campaign. Both managers were fully qualified as ISFSI supervisors. The personnel on each of the four ISFSI crews had been trained and certified in sufficient numbers and operational phases to meet the requirements of 10 CFR 72 Subpart I, "Training and Certification of Personnel."

g. Problem Reporting and Resolution

The licensee's problem event report (PER) list and condition report (CR) summary list related to the ISFSI were reviewed for the period of September 2002 through March 2004. The following selected reports were reviewed in detail.

- PER #202-2640, "MPC Transfer Operations Interrupted Because no Criteria Provided to Assess Annulus Samples"

  After the first canister was loaded and the annulus seal removed from the gap between the canister and the transfer cask wall, a water sample was taken of the water in the annulus. The annulus had been filled with demineralized water and a seal installed prior to placing the canister in the spent fuel pool to prevent pool water from contaminating the canister walls. The water sample was found to have a Cobolt-60 activity of 1.7 x 10^{-6} microCuries (μCi)/milliliter (ml). However, Procedure PPM 6.6.7, "MPC Processing," did not provide guidance concerning acceptable levels of radioactivity in the water. The licensee performed a calculation that determined that the levels found in the annulus were equivalent to less than 10 percent of the contamination limits in Technical Specification 3.3.2 if it was assumed that these same concentrations had plated out on the canister surface as smearable contamination. Revision 4 to Procedure PPM 6.6.7 was issued to incorporate a contamination limit for the annulus water of 1.0×10^{-6} μCi/ml. If the annulus water exceeded this limit, the annulus was to be flushed with clean water. A review was completed of the licensee's calculations supporting the contamination limit. An annulus water concentration of 1.0×10^{-6} μCi/ml could result in a maximum surface contamination level on the canister wall of 50 dpm/100 cm^2 for a 0.45 cm annulus gap. This provided for a limit in the procedure to ensure contamination levels above the T.S. 3.2.2 limit would not be exceeded on the inaccessible portions of the canister due to contamination being introduced into the water below the annulus seal.

- PER #202-2720, "Contamination Unexpectedly Found Inside ISFSI Hi-Track"

  Contamination was found on a portion of the bottom lid of the Hi-Track transfer cask after the first canister was removed. The contamination level measured on one of the smears was 6,000 dpm/100 cm^2. The contamination was not found on the portion of the bottom lid that was in contact with the canister, nor was any contamination found on the interior walls of the Hi-Track. The technical specification 3.2.2 limits for contamination levels for the canister were 1,000 dpm/100 cm^2 beta-gamma and 20 dpm/100 cm^2 alpha. The low annulus water sample radiation levels (discussed in the PER above) and the lack of any contamination found on the Hi-Track interior walls provided reasonable assurance that the canister walls were not contaminated in excess of the technical specification limit.

- PER #203-3304, "OER 16813-Transfer Cask Contamination Attributed to Leaching," PER #203-3407, "OER 4-03-Damage to Fuel," PER #203-4097, "OER 17218-Weld Failure on Inner Cover," and PER #203-4287, "OER PER-Wrong Canister Loaded."

  These four problem evaluation reports were generated from information received concerning problems at other plants loading storage casks. Columbia Generating Station had implemented an active process of following problems at other utilities that were loading canisters and handling spent fuel. When problems were reported, Columbia would document the problem in their PER process and would analyze the issue to determine if changes to the Columbia program were needed to prevent the problem from occurring at the Columbia site.

- PER #203-3604, "Hi-Track Placed in Protected Area but Not Posted as Radiation Area"

  The Hi-Track transfer cask, while unloaded, was brought into the protected area and left unattended for approximately 30 minutes before it was posted as a radioactive materials area. The Hi-Track was labeled with red and yellow radiation survey tags and the radiation levels were less than 1 mrem/hr on contact. The cause for the transfer cask not being immediately posted was determined to be inadequate turnover. Meetings were held with the craftsmen and radiation protection personnel to emphasize the requirement to post the Hi-Track in a timely manner or provide continuous attendance until posting was completed.

- PER #203-3816, "Lid for MPC #95 Would Not Fit", PER #203-3829, "Lid for MPC #96 Would Not Fit", PER #203-4235, "Lid for MPC #99 Would Not Fit"

  During lid fit-up testing for Canisters #95, #96 and #99, their respective lids would not fit into their shells. Galling between the lid and shell were noted at several points causing the lids to only partially fit into the canisters. The canister fabricator came to the Columbia site and performed additional hydraulic bending of the shells and localized grinding of the lids to get proper fit-up. The canister vendor was also notified.

- PER #203-4070, "Unexpected Contamination Found on Hi-Track Surface"

  Contamination was found on the Hi-Track lifting stud holes and the interior seal ring area. Levels as high as 50,000 dpm/100 cm^2 were measured. The contamination was found November 11, 2003, while the Hi-Track was in storage and not in use. The last radiological survey of the Hi-Track had been performed on September 30, 2003. No detectable contamination had been found. The contamination problem discovered on November 11, 2003, was attributed to leaching of contamination out of the metal surface on the Hi-Track. The contamination had originally come from exposure of the metal to the spent fuel pool water. Leaching of contamination on metal casks is a known phenomenon in the industry when casks are left in storage for periods of time.

The licensee's use of the PER system to track problems related to the ISFSI provided an effective system to document, evaluate and determine corrective actions for the various issues being identified for ISFSI related work. A review of the various issues related to the ISFSI found no trends related to radiological problems or operational aspects of the loading of the casks except for the issue related to the cask lids not fitting properly on several casks. The licensee was addressing this issue with the vendor and cask fabricator.

h. Quality Assurance

Two quality assurance audits and a quality assurance monitoring report related to the ISFSI had been issued by the licensee for the period between September 2002 and

March 2004. Document SR-04-02, "Continuous Monitoring Report February 2004," dated March 19, 2004, discussed quality assurance observations during the loading of the sixth canister. Audit Report AU-TS-03, "Technical Specification and Licensing Conditions," dated September 3, 2003, reviewed compliance with selected site technical specifications including the technical specifications for the ISFSI. Audit Report AU-SE-02, "Security Program/Access Authorization/Personnel Access Date System," dated December 18, 2002, reviewed several areas related to site security including a review of security training related to the ISFSI. No significant issues related to the ISFSI were identified in the QA audits.

1.3 Conclusions

Changes made to the ISFSI operations and procedures had been appropriately screened through the 10 CFR 72.48 process and were bounded by the licensing basis documents.

The personnel exposure data for casks 1-5 was very similar to casks 6-10, even though casks 6-10 were considerably hotter. The loading campaigns reflected a learning curve with decreasing exposures as experience was gained. The ALARA and radiological control practices for the ISFSI operation were effective in minimizing personnel exposure.

The thermo-luminescent dosimeters (TLDs) located on the ISFSI area fence provided adequate representation of the ISFSI radiation levels. The environmental TLD data and licensee surveys were consistent with the survey data obtained independently by the NRC inspectors.

After revising its procedure for conducting contamination surveys of the canister, the licensee was meeting the Technical Specification requirements for removable contamination on the exterior surfaces of the canister.

The fuel selected for loading the 6^{th} and 11^{th} canisters was verified to meet the requirements for approved contents. The documentation package for the loaded spent fuel canister selected for review contained the required information.

Loaded Hi-Storm cask temperatures and/or air inlets and outlets were being monitored as required by Technical Specifications. The loaded cask surface dose rates were documented as being maintained within the limits of the Technical Specifications.

Training and certification records for selected personnel assigned to the ISFSI project were reviewed. The licensee had sufficiently trained and certified their ISFSI personnel to meet the applicable requirements.

The licensee's use of the problem evaluation request (PER) system to track problems related to the ISFSI provided an effective system to document, evaluate and determine corrective actions for the various issues being identified for the ISFSI related work.

2 Review of 10 CFR 72.212 Evaluations (60856)

2.1 Inspection Scope

Changes to the 10 CFR 72.212 evaluation were reviewed to determine if the evaluations required by 10 CFR 72.48 had been properly performed.

2.2 Observations and Findings

No changes had been made to the licensee's "Independent Spent Fuel Storage Installation 10 CFR 72.212 Evaluation", since the last NRC inspection. The licensee was in the final stages of the approval process for an upcoming change to the 72.212 report. The change would incorporate the hydrogen water chemistry project into the 72.212 report, including the results of the analysis of the impact of a hydrogen explosion on the ISFSI or on a cask during transport to the ISFSI. During a walk-down of the ISFSI transport route from the fuel building to the ISFSI pad, conducted prior to the movement of the 11th cask on the morning of March 31, 2004, the location of the hydrogen storage area was observed. The facility was near the transport route that was used for transporting the casks to the ISFSI. The licensee had performed a 72.48 screening (Screen 03-010) of the hydrogen water chemistry project and determined that the storage of the hydrogen onsite and the location of the underground pipeline to transport the hydrogen to the plant was bounded by previously analyzed accidents in the ISFSI Final Safety Analysis Report. Upcoming changes to the 72.212 report for incorporating the hydrogen water chemistry project will be reviewed during a future inspection (IFI 72-35/04-01).

2.3 Conclusions

No changes had been made to the licensee's "Independent Spent Fuel Storage Installation 10 CFR 72.212 Evaluation", since the last NRC inspection.

3 Unresolved Items

<u>(Closed) Unresolved Item (URI) 72-35/0201-01, Adequacy of Holtec 72.48 for Hydrogen Issue</u>

On November 1, 2002, the NRC issued Inspection Report 50-397/02-08;72-35/02-01 which included URI 72-35/0201-01 related to the observation of hydrogen bubbles being generated in a Holtec canister placed in the Columbia spent fuel pool. The unresolved item was identified to allow for further NRC evaluation of the hydrogen issue and further review of the position taken by Holtec the change made to the Final Safety Analysis Report related to hydrogen generation by the canister did not require NRC approval. NRC Region IV submitted a Technical Assistance Request to the NRC's Spent Fuel Project Office on November 22, 2002, requesting review of Holtec's evaluation of the hydrogen issue. On March 11, 2003, the Spent Fuel Project Office provided an initial response to the Technical Assistance Request. On April 22-24, 2003, the Spent Fuel Project Office performed an inspection at the Holtec International offices and issued

Inspection Report 72-1014/2003-201 on June 13, 2003. A follow-up response to the Technical Assistance Request was issued to Region IV on September 8, 2003. As a result of the reviews conducted by the Spent Fuel Project Office and the inspection of the Holtec, Inc. programs, a violation was issued to Holtec, Inc. concerning the failure to provide for adequate design measures to ensure compatibility of the Hi-Storm materials as required by 10 CFR 72.146, "Design Controls." No further follow-up is required related to this issue at Columbia. This closes URI 72-35/0201-01.

4 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the exit meeting on March 31, 2004. The licensee did not identify as proprietary any information provided to, or reviewed by, the inspectors.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

P. Ankrum, Senior Licensing Engineer
A. Carlyle, Technical Specialist V
R. Catlow, Senior Engineer
R. Fuller, Reactor Maintenance Manager
S. Grundhauser, Training Supervisor
D. Larkin, Project Manager
T. McNabb, Scientist II
C. Madden, Scientist IV
R. Madden, Quality Auditor II
S. O'Connor, Engineer
G. Shindehite, Operations Support Specialist IV
N. Zimmerman, Reactor Project Manager

INSPECTION PROCEDURES USED

60855	Operations of an ISFSI
60856	Review of 10 CFR 72.212(b) Evaluations

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

72-35/0401-01	IFI	Review of the 10 CFR 72.212(b) Changes Needed to Support the Hydrogen Water Chemistry Project.

Closed

72-35/0201-01	URI	Adequacy of Holtec 72.48 for hydrogen issue

Discussed

None

LIST OF ACRONYMS

ALARA	As Low as Reasonably Achievable
atm cc/sec (He)	Atmospheric Cubic Centimeters per second (Helium)
CoC	Certificate of Compliance
CFR	Code of Federal Regulations
CR	Condition Report
DPM	Disintegrations Per Minute
FSAR	Final Safety Analysis Report
g-moles/l	Gram Moles Per Liter
ISFSI	Independent Spent Fuel Storage Installation
MPC	Multi-Purpose Canister
MWD/MTU	Megawatt Days per Metric Ton Uranium
NRC	Nuclear Regulatory Commission
PER	Problem Evaluation Request
QA	Quality Assurance
RWP	Radiation Work Permit
TLD	Thermo-luminescent Dosimeter
μCi/ml	MicroCuries Per Milliliter

ATTACHMENT 2

LOADED HI-STORM 100S CASKS AT THE COLUMBIA GENERATING STATION ISFSI

LOADING ORDER	MPC (canister) SERIAL #	DATE ON PAD	HEAT LOAD (Kw)	BURNUP MWd/MTU	MAXIMUM FUEL ENRICHMENT %	PERSON-REM DOSE
1	MPC-68-028	09/22/02	10.81	32,299	2.72	0.385
2	MPC-68-031	10/07/02	11.10	32,416	2.72	0.341
3	MPC-68-022	10/28/02	11.30	32,541	2.72	0.315
4	MPC-68-039	11/18/02	11.42	33,045	2.72	0.298
5	MPC-68-033	12/09/02	11.20	32,804	2.72	0.245
6	MPC-68-091	02/25/04	12.00	32,318	2.72	0.389
7	MPC-68-092	03/03/04	17.10	38,607	2.92	0.299
8	MPC-68-093	03/11/04	17.10	38,738	2.92	0.315
9	MPC-68-094	03/18/04	17.00	38,732	2.92	0.303
10	MPC-68-095	03/24/04	17.00	38,772	2.92	0.276

Notes:
- Heat Load (kw) is the sum of the heat load values for all spent fuel assemblies in the cask
- Burnup is the value for the spent fuel assembly with the highest individual discharge burnup
- Fuel Enrichment is the spent fuel assembly with the highest individual enrichment per cent of U-235