UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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November 1, 2002

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Mr. J. V. Parrish
Chief Executive Officer
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SUBJECT: NRC INSPECTION REPORT 50-397/02-08; 72-35/02-01

Dear Mr. Parrish:

This report provides the results of the Nuclear Regulatory Commission's (NRC) team inspections conducted between June 4 and September 20, 2002, at your Columbia Generating Station nuclear reactor facility to evaluate the dry cask storage activities related to your newly constructed Independent Spent Fuel Storage Installation (ISFSI) and to observe the loading of your first cask. The inspections were conducted to confirm compliance of your program and activities with the requirements specified in the certificate of compliance, technical specifications, and Final Safety Analysis Report (FSAR) for the Holtec HI-STORM 100S cask system (Certificate No. 1014) being used at your ISFSI. The enclosed report presents the scope and results of these inspections.

Preliminary exits were conducted with your staff after each of the five inspection trips associated with the pre-operational testing program. On September 19, 2002, an exit was conducted which summarized the results of the inspection of your first spent fuel cask loading. Throughout the inspection of the pre-operational tests and the loading of the first cask, your staff provided excellent support to the NRC inspection team. Your program was found to be comprehensive and well developed. Your staff was well trained and thoroughly understood the various aspects of dry cask storage operations. The efforts by your staff during the pre-operational tests demonstrated the amount of time and hard work that your staff had committed to this project. As a result, the loading of your first cask with spent fuel went smoothly and was very well controlled.

During the pre-operational testing phase of your ISFSI activities, the NRC determined that a violation of NRC requirements had occurred related to the placement of the cask in the train bay during portions of the pre-operational testing without completing a safety evaluation. This violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A of the Enforcement Policy. The NCV is described in the subject inspection report. If you contest the violation or significance of the NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region IV, the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001 and the NRC Resident Inspector at the Columbia Generating facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure(s), and your response will be made 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/

Dwight D. Chamberlain, Director
Division of Nuclear Materials Safety

Docket Nos.: 50-397; 72-35
License Nos.: NPF-21

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NRC Inspection Report
50-397/2002-08;72-35/2002-01

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket Nos.:	50-397;72-35
License No.:	NPF-21
Report Nos.:	50-397/2002-08; 72-35/2002-01
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Facility:	Columbia Generating Station
Location:	Richland, Washington
Dates:	Fuel Examination: June 4, 2002 Welding/Lid Removal: June 25-27, 2002 Program Inspection/Equipment Demonstrations: July 15-19, 2002 Fuel Pool Operations: August 14-16 & September 11-12, 2002 Heavy Loads: September 3-6, 2002 Fuel Load: September 13-20, 2002
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Attachment:	Supplemental Information
ADAMS Entry :	IR 05000397-2002-08/072000035-2002-01; on 06/4-9/19/2002; Energy Northwest; Columbia Generating Station. Safety Evaluation.

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EXECUTIVE SUMMARY

Columbia Generating Station
NRC Inspection Report 50-397/2002-08; 72-35/2002-01

On January 11, 2002, Energy Northwest notified the NRC of their intent to load spent reactor fuel into an Independent Spent Fuel Storage Installation (ISFSI) to be constructed near the Columbia Generating Station. Energy Northwest selected the Holtec HI-STORM 100S cask system as the storage system for the spent fuel. The Holtec HI-STORM 100S cask is licensed by the Nuclear Regulatory Commission (NRC) as Certificate of Compliance No. 1014.

The NRC conducted an extensive evaluation of the licensee's program for the safe handling and storage of spent fuel at their ISFSI, observed the pre-operational test demonstrations, and observed the loading of the first cask. This inspection effort consisted of an in-depth evaluation of the licensee's programs, procedures, training and staff qualifications against the requirements in 10 CFR Part 72, the HI-STORM 100S Certificate of Compliance and the Final Safety Analysis Report (FSAR). The pre-operational testing program required the licensee to demonstrate, through the use of actual equipment and mock-ups, that preparations had been completed to safely load a cask with spent fuel and move the cask to the ISFSI. These required demonstrations were specified in Condition 10 of the Certificate of Compliance. Condition 10 required pre-operational testing of the loading, closure, handling, unloading, and transfer of the HI-STORM 100S cask system to be conducted by the licensee prior to the first use of the system to load spent fuel assemblies.

Columbia Generating Station developed a pre-operational test plan which consisted of six exercises encompassing the required demonstrations and developed a schedule for conducting the exercises under observation of the NRC. The exercises were conducted between June and September 2002. Five trips to the Columbia Generating Station site were completed by the NRC. Inspectors from the NRC Region IV office, NRC headquarters and the NRC's field office for the Yucca Mountain Project participated in the inspections. The inspections were a comprehensive review of the activities associated with safely loading a cask, placing the cask into the ISFSI and maintaining an ongoing program to ensure the cask will be safely stored. In addition, programs being implemented by the reactor facility under their Part 50 license that would be used to support the cask loading and storage activities were reviewed to ensure adequate integration between the two programs.

As a result of the inspections by the NRC of the ISFSI programs and the pre-operational tests conducted by the licensee, the NRC found that development and implementation of a dry cask storage program which complied with NRC regulations and the provisions of the license for the Holtec cask system had been completed. On September 13, 2002, the NRC issued a letter to Energy Northwest informing the licensee that all pre-operational activities required by the certificate of compliance had been successfully completed and Columbia Generating Station could begin loading spent fuel into their ISFSI. At 1:22 p.m. on September 13, 2002, the licensee placed their first spent fuel assembly into a canister.

This report provides the details of the inspections conducted at the Columbia Generating Station and is divided into 18 topical areas. The following provides the conclusions presented in this inspection report for each of the areas reviewed.

Pre-Operational Test Program

- The licensee was required by the certificate of compliance to conduct a pre-operational test program to demonstrate readiness to load spent fuel. The NRC conducted five inspections over a 4-month period to observe the required demonstrations. All required activities were successfully completed and the licensee demonstrated the capability to implement the various elements of the dry cask storage program to successfully load and store spent fuel at the ISFSI (Section 1).

Evaluation of General License Requirements

- An extensive review of the licensee's dry cask storage program was completed against the requirements in 10 CFR 72.212 for a general license. The licensee had documented the required evaluations and developed an extensive set of procedures to control work activities associated with the ISFSI. Evaluations had been completed to demonstrate that the design features for the HI-STORM cask system were enveloped by the site specific characteristics of the Columbia Generating Station site (Section 2).

- The licensee conducted a heavy loads movement activity in the train bay using a weighted canister, transfer cask and storage cask. Analysis had determined that modifications to the train bay floor were needed to provide additional support during an earthquake. The licensee performed this activity using the risk assessment techniques allowed for in the new maintenance rule in 10 CFR 50.65 without performing a safety evaluation in accordance with 10 CFR 50.59. This has been determined to be a violation of NRC regulations and is being dispositioned as a Non-Cited Violation (Section 2).

Fuel Verification

- The licensee had developed a cask loading plan in accordance with approved procedures. Parameters for the 340 spent fuel assemblies selected for loading into the first five casks had been reviewed to verify compliance with the design parameters in the certificate of compliance (Section 3).

- The licensee had performed a review of operating records and conducted a visual examination of the spent fuel assemblies to verify the physical condition of the assemblies selected for loading in the first five casks. All spent fuel assemblies selected were determined to be intact (Section 3).

Spent Fuel Pool

- The fuel bridge safety limit controls prevented the grapple from moving too close to the wall of the spent fuel pool and hoses on the grapple prevented the grapple from completely lowering spent fuel assemblies into several locations along the canister wall.

The licensee completed modifications to the fuel bridge software and grapple to provide for access to all locations in the canister (Section 4).

- The Part 50 FSAR had included a description of a cask for removing spent fuel from the spent fuel pool. This cask was not approved for storage of spent fuel at an ISFSI and was smaller and lighter than the Holtec design. An amendment to the plant license was required to incorporate the Holtec cask design into the Part 50 FSAR (Section 4).

Procedures and Technical Specification Compliance

- Procedures consisted of a checklist format that provided for good documentation of work activities completed. Procedures included precautions and important reminders of critical parameters. Commitments from the certificate of compliance, technical specifications and the FSAR had been incorporated into procedures. Implementation of the procedures during the pre-operational tests confirmed the adequacy of the procedures for various work tasks observed during the demonstrations (Section 5).

- The licensee had incorporated written guidance into procedures for abnormal events such as unexpected high dose rates, cask drops, tornado or severe weather conditions, high contamination levels encountered in the work areas, stuck fuel assembly during removal from the fuel racks or during insertion into the cask, and dropped fuel bundles (Section 5).

Safety Reviews

- The licensee had implemented a program to perform safety screenings and evaluations in accordance with the requirements in 10 CFR 50.59 and §72.48. Selected screenings and evaluations performed by the licensee were reviewed and found to be adequately dispositioned (Section 6).

Heavy Loads

- The procedures governing the heavy load lift operations appropriately contained the requirements and guidance from the FSAR and national standards for maintenance and testing to ensure the ability of the equipment to support the anticipated loads required during the dry cask storage program activities (Section 7).

- The heavy loads procedures appropriately addressed cask lift limits. The expected component weights were bounded by the weight values established in the FSAR (Section 7).

- The licensee's planned use of an ancillary device placed under the casks on the 441' elevation and 606' elevation to mitigate the effects of a seismic event were reviewed by the NRC and found acceptable (Section 7).

Hydrostatic Testing/Drying/Helium Backfill

- The licensee demonstrated the capability to perform hydrostatic testing, drying, and helium backfill of a canister during the pre-operational tests. Procedures and purchase orders were reviewed to verify that equipment associated with these activities was capable of achieving the required limits specified in the technical specifications (Section 8).

Welding/Non-Destructive Testing

- The procedures governing the operations for canister lid welding, nondestructive testing of the lid welds and removal of the lid, should that be necessary, were detailed, thorough, and appropriately qualified. Welding and nondestructive examination personnel were properly qualified. Observation of the welding, nondestructive examinations and cutting operations on the mockup canisters verified personnel skills, procedure adequacy, and equipment capability. The nondestructive examinations verified the capability of the welders to produce high quality welds (Section 9).

Health Physics

- The licensee was implementing their site radiation protection program for activities associated with the ISFSI. Several additions were incorporated into the program to address cask-specific radiological conditions. These included the development of procedures to address radiological surveys of the loaded cask, development of specific radiation work permits that defined radiological controls required during the different phases of cask loading and movement, the addition of dosimetry around the ISFSI and implementation of training specific to the radiological conditions that would be encountered during cask loading and movement (Section 10).

Emergency Planning

- The licensee incorporated provisions for responding to an emergency at the ISFSI into their existing site wide emergency planning program. The site emergency plan and procedures were revised to incorporate new emergency action levels for the accidents described in the FSAR that would require declaration of an emergency (Section 11).

- Training had been completed for emergency responders. Interviews of personnel confirmed that individuals were adequately trained and knowledgeable of response actions. Existing provisions for response from offsite support organizations during an emergency were found to be adequate for events that could occur at the ISFSI (Section 11).

Fire Protection

- The licensee had developed a program for controlling flammable liquids at the ISFSI to comply with technical specification requirements and to keep flammable liquids below the level analyzed in the FSAR for the worst case fire scenario. The Fire Hazards Analysis evaluated numerous fire scenarios to confirm that site specific fire scenarios were bounded by the design analysis in the FSAR. Administrative controls were established to limit the amount of flammable liquids that could be near a loaded cask and to lock-out rail traffic on the nearby rail line during cask movement. Arrangements had been made for support from an offsite fire department for fires at the ISFSI (Section 12).

Training Program

- The licensee had developed a training and certification program for operator personnel performing work at the ISFSI on equipment and controls that were identified as important to safety. The program incorporated the requirements in 10 CFR 72 Subpart I and Section 12.2.1 of the FSAR and included formal classroom training, on-the-job training and specific task demonstrations (Section 13).

- Training was completed for all operator personnel assigned ISFSI duties. Interviews with selected personnel verified that training had been adequately implemented (Section 13).

Quality Assurance Program

- The licensee conducted quality assurance oversight of ISFSI activities using their NRC approved 10 CFR Part 50 quality assurance program. A review of documents, procedures and audits performed by the quality assurance organization determined that the licensee had appropriately applied their Part 50 quality assurance program to the activities associated with the ISFSI (Section 14).

Security

[redacted]

[redacted]

Records/Documentation

- Adequate provisions were established by the licensee to ensure that required documents and records specified in 10 CFR 72.212 would be retained for the casks loaded at the ISFSI and that required notifications would be completed (Section 16).

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Loading of the First Cask

- The licensee successfully completed the loading of their first cask and placement of the cask on the ISFSI pad on September 20, 2002. Dry cask storage activities were conducted safely and in compliance with procedures. Radiological controls were effectively implemented. The overall dose to complete the project was well below original estimates (Section 17).

Special Topic: Hydrogen Generation by the Holtec Cask

- The FSAR for the HI-STORM 100 cask system provided various statements that hydrogen would not be generated in the canister while in the spent fuel pool. However, during the pre-operational test demonstrations, bubbles were observed coming from the canister. The bubbles were analyzed and found to contain 63 percent to 75 percent hydrogen (Section 18).

- Holtec performed a safety evaluation of the hydrogen generation issue in accordance with 10 CFR 72.48. The evaluation concluded that the FSAR could be revised to incorporate provisions for hydrogen generation by the canister without approval by the NRC. The adequacy of Holtec's evaluation will be reviewed by the NRC and will be tracked as an Unresolved Item (Section 18).

- The licensee implemented a hydrogen mitigation process during the welding of the lid onto the canister that successfully monitored for and prevented any build-up of hydrogen under the lid which could be ignited during the welding (Section 18).

Report Details

Summary of Facility Status

Columbia Generating Station is a General Electric boiling water reactor owned by Energy Northwest. The reactor began commercial operation on December 13, 1984. The facility is located approximately 12 miles northwest of Richland, Washington, on the Department of Energy's Hanford Reservation. On January 11, 2002, Energy Northwest notified the NRC of their intent to load spent reactor fuel into an Independent Spent Fuel Storage Installation (ISFSI) to be constructed near the Columbia Generating Station. The spent fuel would be stored under the general license provisions of 10 CFR Part 72.

Energy Northwest selected the Holtec HI-STORM 100S cask system as the storage system for the spent fuel. The Holtec HI-STORM 100S cask system is licensed by the Nuclear Regulatory Commission (NRC) as Certificate of Compliance No. 1014. This system consists of a multi-purpose storage canister (MPC) which holds 68 fuel assemblies. The canister is placed inside the HI-TRAC transfer cask to provide shielding for protection of the workers during transfer operations and during the drying, helium backfilling and welding of the canister. The canister is loaded with spent fuel, drained of water, filled with helium gas and sealed by welding. The canister is then moved from the refuel floor and lowered to the train bay while in the transfer cask. With the transfer cask and canister placed on top of the HI-STORM storage cask, the canister is lowered from the transfer cask into the storage cask. The loaded storage cask is transported from the plant using a cask transporter and placed on the ISFSI pad for storage.

Columbia Generating Station plans to load five casks before the end of 2002 to make room for the spent fuel that will be removed from the reactor during the May 2003 outage. The spent fuel pool contained 2256 spent fuel assemblies, 230 cells filled with blade guides and other components and 168 open cells. The reactor contained 764 fuel assemblies of which approximately 300 will be removed during the May 2003 outage. The spent fuel pool lost full core offload capability in May 1999. After the May 2003 outage, an additional 10 casks will need to be loaded in order to re-establish full core offload capability.

1 PRE-OPERATIONAL TEST PROGRAM (60854)

1.1 Inspection Scope

The Certificate of Compliance for the HI-STORM 100S cask system and Section 12.2.2 of the FSAR required the licensee to conduct pre-operational testing to demonstrate the loading, closure, handling, unloading, and transfer of the cask system prior to the first loading of spent fuel assemblies. The NRC conducted several onsite inspections to observe the licensee's demonstration of the activities required by the certificate of compliance.

1.2 Observations and Findings

The Certificate of Compliance for the Holtec HI-STORM 100S cask system included a requirement to demonstrate certain specific activities prior to loading the first cask.

These requirements were specified in Condition 10 of the Certificate of Compliance and in Section 12.2.2 of the FSAR as follows:

a) Move a canister and transfer cask into the spent fuel pool.
b) Prepare the Hi-STORM 100S cask system for loading fuel.
c) Select and verify specific fuel assemblies to ensure type conformance.
d) Load specific assemblies and place assemblies into the canister (using a dummy fuel assembly), including appropriate independent verification.
e) Remotely install the canister lid and remove the canister and transfer cask from the spent fuel pool.
f) Demonstrate canister welding, non-destructive examination (NDE), hydrostatic testing, draining, moisture removal (by vacuum drying or forced helium dehydration, as applicable), helium backfilling, and leakage testing.
g) Demonstrate transfer cask upending/downending on the horizontal transfer trailer or other transfer device, as applicable to the site's cask handling arrangement. (Note: This activity is not applicable to the Columbia Generating Station since the cask is moved to the ISFSI using a crawler that maintains the cask in a vertical position.)
h) Transfer the canister from the transfer cask to the overpack.
i) Place the HI-STORM 100S cask system at the ISFSI.
j) Demonstrate HI-STORM 100S cask system unloading, including cooling fuel assemblies, flooding canister cavity, and removing canister lid welds.

Columbia Generating Station developed Instruction No. SFS-06 "ISFSI Dry Run Plan," dated April 4, 2002, to describe the activities that would be conducted to comply with the requirement for the pre-operational test. Six individual exercises were identified in Instruction No. SFS-06. The six exercises and a fuel examination were conducted by the licensee to fulfill the requirements of Condition 10 of the Certificate of Compliance. The six exercises were:

- Exercise No. 1 - Canister Welding Operation
- Exercise No. 2 - Weld Removal Operations
- Exercise No. 3 - Canister Closure Operations
- Exercise No. 4 - Helium Cool Down Skid Operations
- Exercise No. 5 - Cask Handling Operations
- Exercise No. 6 - Fuel Pool Operations and Fuel Loading

Five onsite inspections were performed by the NRC to observe all six exercises and the fuel examination project. The inspections covered the following areas.

- On June 4, 2002, NRC inspectors reviewed the licensee's process for selecting and verifying fuel assemblies for placement in the first five storage casks. This included a review of the licensee's examination process of the fuel assemblies to verify no failed fuel had been selected for storage. The licensee had also videotaped the fuel examinations. Selected portions of the video tape were reviewed. (Fuel exam)

- On June 25-27, 2002, NRC inspectors observed the welding and nondestructive testing demonstrations on a truncated canister. An automatic welding machine was used. Also observed was the automatic cutting machine, which would be used to remove the welds on a canister lid should unloading of a canister be necessary. (Exercise Nos. 1 & 2)

- On July 15-18, 2002, NRC inspectors and representatives from the Washington State Department of Health and Oregon State Office of Energy observed the demonstration of equipment used for the hydro testing, draining, vacuum drying, helium backfill, helium leakage testing and the helium cool-down process. A detailed review of procedures, training and records were completed for the dry cask storage project. This included reviewing reactor programs that were being implemented to support the dry cask storage project. (Exercise Nos. 3 & 4)

- On August 14-16 and September 11-12, 2002, NRC inspectors and a representative from the Washington State Department of Health observed the fuel pool operations associated with loading a canister. This included placing the transfer cask loaded with a canister into the spent fuel pool and demonstrating that the refueling bridge was able to load and remove a dummy fuel assembly into several locations in the canister. The canister lid was positioned in place and the transfer cask and canister removed from the spent fuel pool. (Exercise No. 6)

- On September 3-6, 2002, NRC inspectors and a representative from the Washington State Department of Health observed the cask handling demonstration which included placement of a HI-STORM storage cask in the train bay, positioning the HI-TRAC transfer cask onto the top of the storage cask, removal of a weighted canister from the storage cask, insertion of the canister into the transfer cask, and lifting of the transfer cask and canister to a height above the storage cask. Then the transfer cask and canister were lowered back on top of the storage cask and the canister inserted into the storage cask. The loaded storage cask was then transported to the ISFSI pad. (Exercise No. 5)

A comparison of the exercises planned by the licensee with the requirements in Condition 10 of the Certificate of Compliance is provided in the following table. The licensee successfully completed all the required demonstrations.

Table 1-1
Pre-Operational Test Demonstrations

Condition #10	Requirement	Demonstrated in Exercise #
a	Movement of a canister/transfer cask into pool	#6
b	Preparing a cask for loading	#5 #6

Condition #10	Requirement	Demonstrated in Exercise #
c	Selection and verification of fuel	Fuel Exam
d	Loading assembly into canister and verification	#6
e	Installing the canister lid and removal of canister/transfer cask from pool	#6
f	1) Welding, NDE 2) Hydro, drain, dry, backfill, leak test	#1 #3
g	Upending onto horizontal trailer	Not Applicable
h	Transfer of canister into HI-STORM	#5
i	Moving a HI-STORM cask to the ISFSI	#5
j	1) Unloading a canister from HI-STORM 2) Unloading dummy fuel from canister 3) Cool down and flooding a canister 4) Removal of the canister lid	#5 #6 #4 #2

The licensee performed the various exercises as realistic as practical. Health physics controls were established and personnel performing the work activities were required to dress-out as if an actual contaminated zone existed. Procedures and checklists used for the exercises were modified from the procedures that were planned for use during the actual loading. The procedures were modified to account for the simulations that were necessary when using the truncated casks and performing the demonstrations in the warehouse as opposed to the actual location in the plant. Overall, the workers performed well, took the necessary time to stop and discuss issues, and made sure that all actions being taken were understood.

1.3 Conclusion

The licensee was required by the certificate of compliance to conduct a pre-operational test program to demonstrate readiness to load spent fuel. The NRC conducted five inspections over a 4-month period to observe the required demonstrations. All required activities were successfully completed and the licensee demonstrated the capability to implement the various elements of the dry cask storage program to successfully load and store spent fuel at the ISFSI.

2 EVALUATION OF GENERAL LICENSE REQUIREMENTS (60856)

2.1 Inspection Scope

A nuclear power plant, which operates under a10 CFR Part 50 license, may implement a dry cask storage program under the provisions of a general license in accordance with

10 CFR Part 72. In order to do this, the nuclear power plant must complete certain notifications and written evaluations required by 10 CFR 72.212. The NRC conducted a review of the licensee's documentation to verify compliance with the requirements of §72.212.

2.2 Observations and Findings

Two written evaluations related to the requirements in 10 CFR 72.212 had been completed. One evaluation was performed by Holtec International and issued as Holtec Report HI-2012664 entitled, "10 CFR 72.212 Evaluation of the Columbia Generating Station ISFSI," dated April 11, 2001. The second evaluation was issued in June 2002, by the licensee and entitled, "ISFSI 10 CFR 72.212 Evaluation," Revision 0. To determine if the specific requirements in §72.212 had been met, selected portions of these documents and the associated procedures, program documents and records were reviewed to verify that the licensee had either completed the required activity or had established adequate documentation and program controls to ensure compliance.

The first notification requirement that must be met by a general licensee was the notification to the NRC of the intent to store spent fuel at an ISFSI. The licensee was required by 10 CFR 72.212(b)(1)(i) to notify the NRC at least 90 days prior to the first storage of spent fuel. Energy Northwest notified the NRC on January 11, 2002, of their intent to use the Holtec HI-STORM 100 cask system in accordance with Certificate of Compliance #1014. This letter met the requirement for the 90-day notification.

The licensee was required by 10 CFR 72.212(b)(1)(ii) to register each cask after loading and within 30 days. The licensee had issued procedure SFS-05, "MPC Documentation Tracking Requirements," Revision 0, which required notification to the NRC within 30 days of loading a cask. This procedural requirement met the notification requirement.

A written evaluation was required by §72.212(b)(2)(i)(A) to establish that the conditions set forth in the certificate of compliance have been met. The certificate of compliance for the Holtec HI-STORM-100 cask system has 11 conditions.

Condition 1 discusses the various models of the Holtec HI-STORM-100 cask system that are available for use under Certificate of Compliance #1014. Energy Northwest will use the HI-STORM-100S version with the MPC-68 multipurpose canister. The HI-STORM 100S storage overpack (concrete cask) provided shielding and structural protection for the canister during storage. The HI-STORM 100S was a shortened version of the HI-STORM 100 with a modified lid design incorporating the air outlet ducts into the lid. The MPC-68 can hold 68 boiling water reactor (BWR) fuel assemblies.

Condition 2 of the Certificate of Compliance required written operating procedures to be prepared for cask handling, loading, movement, surveillance and maintenance. The procedures were required to be consistent with the technical basis as described in Chapter 8 of the FSAR. The licensee's §72.212 evaluation report did not contain a list of procedures developed to support the ISFSI activities. However, the licensee had developed a matrix of procedures required for the ISFSI as a separate document. The matrix included a cross reference between the requirements in the certificate of compliance and FSAR with the specific procedure. The matrix also listed the responsible person assigned to the procedure. The NRC inspection team reviewed a significant number of procedures related to the dry cask storage program and observed the implementation of procedures during the dry run demonstrations and cask loading operations. The licensee had developed procedures for all the areas discussed in Chapter 8 of the FSAR. Based on the adequacy of the procedures reviewed and the comprehensive number of procedures that had been developed, the licensee was found to be in compliance with Condition 2 of the Certificate of Compliance.

Condition 3 of the Certificate of Compliance required a written cask acceptance test and maintenance program consistent with the technical basis of Chapter 9. A review of Chapter 9 was completed and issues related to welding and verification of the adequacy of the weld through visual examinations, nondestructive testing and leak testing were selected to verify that the licensee had developed acceptable provisions for testing and maintenance in this area. The results of these reviews are documented in Section 9, "Welding/Nondestructive Testing," of this report. The licensee's programs and procedures for the areas reviewed were found to include the technical basis elements described in Chapter 9 of the FSAR. The licensee was determined to be in compliance with Condition 3 of the Certificate of Compliance.

Condition 4 of the Certificate of Compliance required the licensee to conduct activities that were considered "important to safety" in accordance with an NRC approved quality assurance program which satisfied the requirements in 10 CFR Part 72, Subpart G. 10 CFR 72.140(d) allows the use of a quality assurance program previously approved by the NRC under Appendix B of Part 50 by nuclear power plants without developing a separate quality assurance program for Part 72. By letter dated August 11, 2000, Energy Northwest notified the NRC of their plans to use their Part 50, Appendix B, quality assurance program for activities associated with dry cask storage. The Part 50 quality assurance program was reviewed during this inspection and is documented in Section 14, "Quality Assurance Program," of this report. Implementation of the quality assurance requirements to selected portions of the dry cask storage program were reviewed. The licensee was found to be adequately implementing their Part 50 quality assurance program for the activities related to Part 72 and was in compliance with Condition 4 of the Certificate of Compliance.

Condition 5 of the Certificate of Compliance established requirements for heavy loads. The licensee had developed an extensive evaluation of their crane capabilities and heavy loads program. A significant portion of this information had been developed in response to issues related to NRC Bulletin 96-02, "Movement of Heavy Loads over Spent Fuel Pools, Over Fuel in the Reactor Core, or Over Safety-Related Equipment." On September 13, 2001, Energy Northwest provided to the NRC an independent engineering assessment of the reactor building crane to demonstrate compliance with the requirements of NUREG 0554, "Single Failure Proof Cranes for Nuclear Power Plants," in support for a license amendment. The NRC issued Amendment 174 on October 26, 2001, as supplemented on November 27, 2001, which accepted the reactor building crane as a single failure proof crane.

Condition 5 also specified that heavy load activities outside the plant were governed by Section 5.5 of Appendix A to the Certificate of Compliance and/or Section 3.4.6 and Section 3.5 of Appendix B. Section 5.5 of Appendix A established lift height limits for the loaded transfer cask. Section 5.5.3 stated that a loaded transfer cask could be lifted to any height during transport between the plant and the ISFSI pad when a lifting device was used that was designed in accordance with American National Standards Institute ANSI N14.6 requirements and had redundant drop protection features. The licensee was using a transporter designed to the ANSI standard with redundant drop protection features. Section 3.4.6 of Appendix B had the same applicable lift conditions described in Section 5.5 of Appendix A. Section 3.5 of Appendix B applied to cask transfer facilities, which the licensee does not have. Based on the heavy loads program being implemented at the licensee's facility and the use of a single failure proof crane, the licensee was found to be in compliance with the requirements in Condition 5 of the Certificate of Compliance.

Condition 6 of the Certificate of Compliance required the spent fuel designated for storage in the canister to meet the requirements in Appendix B to the Certificate of Compliance. Section 2.1 and Tables 2.1-1 through 2.1-8 of Appendix B provided the specifications for the fuel. Neither of the §72.212 documents provided information on how the licensee verified that the spent fuel at the Columbia Generating Station complied with the Certificate of Compliance, Appendix B, requirements. However, the licensee had developed Procedure 9.6.1, "Spent Fuel Selection for Cask Storage," Revision 0, which described the process for selecting the spent fuel assemblies to meet the criteria specified in the certificate of compliance. This procedure used calculation NE-02-00-08, "Fuel History Data," and calculation NE-02-02-07, "Fuel Bundle Volumes," to provide the fuel parameter data for comparison with the tables in Appendix B of the Certificate of Compliance. In addition, the licensee performed a visual examination of the spent fuel in accordance with work order #01040506 Task 1 and Procedure 6.3.40, "Determination of Fuel Assembly Condition for ISFSI," Revision 0, to verify that the fuel selected was not damaged. Based on the procedures and calculations, the licensee was found to be in compliance with Condition 6 of the Certificate of Compliance.

Condition 7 of the Certificate of Compliance required the characteristics for the site, cask, and ancillary equipment to be consistent with the requirements in Appendix B of the Certificate of Compliance. Several sections in Appendix B were selected for review to verify that the licensee had completed the required evaluations. Areas reviewed included flooding, temperature extremes, and fire/explosions.

Flooding was discussed in Section 3.4.4 of Appendix B. The flooding limit specified was 15 feet/second velocity and 125 feet height. The licensee's ISFSI 10 CFR 72.212 Evaluation Report, Section 4, included information concerning potential flooding. Failure of the upstream dams on the Columbia River had been evaluated. The ISFSI was located 440 feet above Mean Sea Level. The analyzed worse flooding case conditions, including failure of the upstream dams on the Columbia River, resulted in a flood level 424 feet above Mean Sea Level. Therefore, the ISFSI pad would not be subjected to flooding. In addition, Section 3.4.9 of Appendix B required an analysis to demonstrate adequate heat removal, for those users whose site specific design basis included a flood that results in the blockage of the overpack inlets or outlets for an extended period.

Since flooding was not a credible event, the heat removal capabilities of the storage system were not affected.

Section 3.4.1 of Appendix B established a maximum average yearly temperature of 80°F. The annual average ambient temperature from 1945 to 1999 at the Hanford site was documented to be 53.4° F in the report, "Design Safety Assessment Report Summary/Approval," dated July 11, 2002. This average ambient temperature was well below the maximum allowable average yearly temperature of 80 °F.

Section 3.4.2 of Appendix B restricted allowable temperature extremes, averaged over a 3-day period (i.e., 72 hours per FSAR 2.2.2.2), to not exceed 125° F or be less than -40° F. The licensee had reviewed historical data to confirm that the site was bounded by the limits specified. Maximum daytime temperatures had varied from 100 °F to 115 °F. The hottest period recorded was between July 15 to August 13, 1971. During the 30-day period, 27 days had high temperatures that exceeded 100 °F. Even with these high temperatures, the maximum 72-hour average for the hottest 3 days was only 94 °F. The minimum winter temperatures had varied from -27 °F to 22 °F. All temperature extremes were well within the required limits.

Section 3.4.5 of Appendix B established fire and explosion limits and restricted the onsite transporter fuel tank to a maximum of 50 gallons of diesel. The evaluation of the transporter fuel tank and the potential for a fire or explosion is discussed in Section 12, "Fire Protection," of this report. The licensee had established adequate controls to prevent a serious fire or explosion.

Section 3.4.8 of Appendix B limited loading, transporting and unloading operations to be conducted only when working area ambient temperatures were ≥0 °F. This requirement had been incorporated into Step 4.3 of the precautions and limitations section of Procedure 6.6.4, "HI-Storm System Site Transportation," Revision 3.

Based on the review of selected areas in Appendix B related to flooding, temperature extremes, and fire/explosions, it was determined that the licensee was in compliance with the requirements in Condition 7 of the Certificate of Compliance.

Condition 8 of the Certificate of Compliance required the holder of the certificate of compliance to submit an application for amendment to the NRC for any desired changes to the Certificate or Appendices A and B. This condition applied to Holtec and would not be applicable to Energy Northwest as a general licensee. Energy Northwest cannot make changes or request changes from the NRC for the Certificate of Compliance and Appendices A and B. The licensee had incorporated into Procedure SWP-LIC-02, "Licensing Basis Impact Determinations," Revision 2, a statement in Step 4.9.3 that "Any activity requiring prior NRC approval or requiring a change to plant operating license or technical specifications or ISFSI technical specifications or cask certificate of compliance shall not be implemented until NRC approval has been obtained." The licensee recognized that changes to the certificate of compliance must be requested through Holtec as the certificate holder. The licensee was found to have an adequate understanding of the process for changing the certificate of compliance and associated appendices and was in compliance with the requirements in Condition 8 of the Certificate of Compliance.

Condition 9 of the Certificate of Compliance required that the heat transfer characteristics of the cask system (for each unique MPC basket design: MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-68, MPC-68F, and MPC-68FF) be recorded by temperature measurements for the first HI-STORM cask system placed in service, by any user, with a heat load equal to or greater than 10 kW. An analysis was required to demonstrate that the temperature measurements validate the analytic methods and predicted thermal behavior described in Chapter 4 of the FSAR. Validation tests were required for each subsequent cask system that had a heat load exceeding a previously validated heat load by more than 2 kW. Validation tests were not required for heat loads greater than 16kW. Since the licensee was loading their first cask with a heat load of 10.81 kW, then the validation test would be required if this was the first MPC-68 cask loaded by any user. However, Southern Company had already loaded an MPC-68 cask at their Hatch plant in July 2001 with a heat load of 10.1 kW. A letter had been sent to the NRC certifying that the required heat load test had been completed to validate the analytic methods and predicted thermal behavior described in Chapter 4 of the FSAR. The analysis from Southern Company showed the Holtec models to be conservative. The predicted air temperature for the vent was 133.8 °F. The actual measured value was 115.7 °F. With the ambient temperature at 77.5 °F, the actual temperature rise was 38.2 °F compared to the higher predicted value from the FSAR of 56.3 °F. Based on the documentation provided by Southern Company to the NRC, Condition 9 was not required to be met by Energy Northwest unless they become the first utility to load a cask with a heat load greater than 12 kW.

Condition 10 of the Certificate of Compliance required the licensee to conduct a pre-operational testing and training exercise. The licensee conducted the required pre-operational test in a series of exercises conducted between June 2002 and September 2002. The tests were performed successfully and are documented in Section 1, "Pre-Operational Test Program," of this report.

The licensee was required by 10 CFR 72.212(b)(2)(i)(B) to perform a written evaluation of the cask storage pad and areas to document the adequacy of the design to support the static load of the stored cask. The licensee documented the required evaluation in Section 2 of their ISFSI 10 CFR 72.212 Evaluation Report. The evaluation included a review of the design of the storage pad, the adequacy of the on-site transportation route, and compliance with the drop and tip-over analysis described in Chapter 3 of the FSAR. The licensee completed an extensive effort related to the construction of the ISFSI pad and the placement of the on-site road. The soil under the pad was evaluated by field exploration consisting of drilling and sampling one boring, advancing six piezo-cone probes, excavating eight test pits and performing six plate load tests. Laboratory tests were performed on the soil to index the properties and compaction tests performed. The pad area was excavated down to 4 feet and backfilled with compacted quality Class I structural fill in accordance with written procedures. Two pads were poured which measured 30' by 147' by 2' in depth. Each pad can hold 18 casks. Additional pads will be poured as needed.

In addition to the analysis performed by the licensee, the NRC conducted an inspection of the pad design and pad pouring activities on October 15, 2001. This inspection was documented in Inspection Report 50-397/01-05 dated January 18, 2002. The NRC observed the actual pouring and construction of the first pad including the slump tests and moisture tests for the concrete. The 28-day break test results, which verified the strength of the concrete, were reviewed by the NRC Resident Inspector and found acceptable.

The licensee also performed an extensive evaluation of the roadway used for transporting the cask to the ISFSI pad to verify that no pipes, cables or pits were located under the roadway that would be damaged from the weight of the cask and cask transporter. Based on the extensive evaluations completed by the licensee related to the pad design and construction and the roadway between the plant and the ISFSI, it was determined that the licensee had complied with the requirements related to design of the ISFSI pad.

The licensee was required by 10 CFR 72.212(b)(2)(i)(C) to perform a written evaluation to verify the requirements of 10 CFR 72.104 were met. This regulation established criteria for radioactive material in effluents and direct radiation. Since the casks will be welded and sealed, no effluents will be associated with the cask storage activities. For direct radiation exposures, §72.104 limits the exposure to any real individual located beyond the controlled area to 25 mrem whole body, 75 mrem thyroid, and 25 mrem to any other critical organ. The licensee performed the required evaluation in Section 3 of their ISFSI 10 CFR 72.212 Evaluation Report. The distance from the ISFSI to the exclusion area boundary was approximately 1,450 meters. The calculated dose to a member of the public at the exclusion area boundary from 90 casks loaded in the ISFSI (8,760 hours) from all site operations including the doses due to the reactor operations would be less than 7 mrem/year whole body, 7 mrem/year thyroid and 11 mrem/year to the critical organ. Of these doses, less than 0.02 mrem/year was due to the ISFSI. This was based on the individual remaining at the exclusion area boundary for a full year (8,760 hours).

The licensee was required by 10 CFR 72.212(b)(3) to review the FSAR and the NRC Safety Evaluation Report to verify that reactor site parameters, including analysis of earthquake intensity and tornado missiles were enveloped by the cask design. The licensee documented this review in Section 4 of their ISFSI 10 CFR 72.212 Evaluation Report. Environmental conditions, volcanic eruptions, tornadoes, lightning, floods, fires, and seismic events were included in the evaluations. For tornado generated missiles, the casks were analyzed for 300 mph winds. The maximum tornado wind estimated for the ISFSI was 214 mph. Only 14 tornados had been reported within 100 miles of the Columbia site since 1916, making a tornado generated missile striking a cask stored at the ISFSI a very unlikely event. For earthquakes at the Columbia site, the maximum vibratory acceleration levels (free field) were 0.25g and 0.15g for the site safe shutdown earthquake (SSE) and operating basis earthquake (OBE), respectively. Additional information concerning the earthquake potential at the Columbia site was provided in the Holtec Report No. HI-2012664, "10 CFR 72.212 Evaluation of the Columbia Generating Station ISFSI," dated April 11, 2001. This report provided a detailed evaluation of both sliding and tip over for the cask on the pad and verified that the licensee's ISFSI was bounded by the seismic design parameters in the HI-STORM FSAR.

The licensee was required by 10 CFR 72.212(b)(4) to determine whether activities related to the storage of spent fuel under a general license required any changes to the reactor technical specifications or license, as required by 10 CFR 50.59(c)(2). The licensee had reviewed the various issues related to dry cask storage and the impact on the current Part 50 license and technical specifications. The review was documented in Section 5 of their ISFSI 10 CFR 72.212 Evaluation Report. Only one issue was identified that required an amendment request for a change to the Part 50 FSAR. This involved changing the references in the Part 50 FSAR from the General Electric IF-300 cask to the Holtec cask as the cask identified for removing spent fuel from the spent fuel pool. This issue is discussed further in Section 4, "Spent Fuel Pool," of this report.

During a tour of the plant on July 15, 2002, the NRC observed the transfer cask containing a weighted canister positioned on top of the storage cask in the train bay (elevation 441'). This configuration weighed approximately 500,000 lbs. The licensee had placed the cask and canister in this configuration as part of their training exercise to prepare for the pre-operational test demonstration. The licensee performed an analysis of this configuration in Design Safety Assessment (DSA) 97-0180-4E-750, dated July 11, 2002, which determined that the 441' floor slab was incapable of supporting the 500,000 lbs load during a seismic event. Analysis concluded that floor modifications were needed and that four steel reinforcement columns placed under the 441' floor down to the ground level floor at the 422' elevation would provide adequate support. The licensee determined that the stacked-up configuration was acceptable and the preparations for the pre-operational test could continue based on risk considerations allowed for by Procedure PPM 1.5.14, "Risk Assessment and Management of Maintenance and Surveillance Activities." This procedure implemented the new maintenance rule in 10 CFR 50.65. The licensee documented their analysis of the risk assessment of placing the casks in the train bay in a document entitled, "Risk Management Plan for Testing a HI-STORM 100 Cask Load Handling Simulator Inside Reactor Building Prior to Final Engineering Design Evaluation." The licensee decided to proceed with the dry run practice exercise based on the low probability that an operating basis earthquake would occur during the time the cask system was in the stacked-up configuration in the train bay.

The NRC reviewed the basis for the licensee's decision to proceed and determined that Procedure PPM 1.5.14 did not apply to the dry cask storage activities being conducted. The licensee should have completed a 50.59 safety evaluation and determined that the activity should be delayed until after the floor modifications had been completed. The licensee removed the casks from the train bay and completed the modifications to the floor. The NRC reviewed the calculations for the floor modifications in the licensee's report BDC 97-0180-4E-871, "Analysis of the Reactor Building Railroad Bay 441' Elevation Slab," dated August 7, 2002. This report also included ABS Consulting Calculation 253187.05-C-002, "Analysis of 441' Slab: Hermit Loads," dated August 1, 2002. The addition of the four steel reinforcement columns below the 441' floor were determined to be an adequate solution to support the floor during a seismic event when a loaded canister was in the stacked-up configuration. Subsequent analysis provided by the licensee showed that the floor would not have failed without the steel reinforcement columns in place but that the "design basis margin" for the point load force on the floor specified in the Part 50 FSAR Section 3.8.3.5.1 would have been exceeded.

The NRC has determined that failure to complete a 50.59 safety evaluation for the cask stacked-up configuration inside the train bay was a violation of 10 CFR 72.212(b)(4). This violation is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy (72-35/0201-01). This violation was entered into the licensee's corrective action program as Problem Evaluation Request (PER) No. 2002-2138. Modifications to the floor have been completed.

[redacted]

The licensee was required by 10 CFR 72.212 (b)(6) to review the emergency planning, quality assurance, radiation protection, and site-wide training programs to determine if changes were needed to these programs to incorporate the ISFSI related activities. Section 6.0 of the licensee's ISFSI 10 CFR 72.212 Evaluation Report documented the licensee's review of these programs. Changes were made to each of the programs to incorporate the ISFSI related activities. The details of the changes made to the programs are discussed in this inspection report in the sections for each topical area. No significant changes were necessary to the programs.

The licensee was required by 10 CFR 72.212 (b)(7) to maintain a copy of the certificate of compliance and documents referenced in the certificate for each cask model used. The licensee had a copy of the certificate of compliance readily available. Selected NUREGS, ASME codes, and ANSI standards referenced in the FSAR and technical specifications were easily located by the inspector in the corporate library.

The licensee was required by10 CFR 72.212 (b)(8) to maintain certain records. A review of the licensee's records to confirm compliance with this requirement is included in Section 16, "Records/Documentation," of this report. The licensee was maintaining the required records.

The licensee was required by 10 CFR 72.212 (b)(9) to conduct activities related to storage of spent fuel only in accordance with written procedures. The licensee had developed numerous procedures related to the ISFSI. All areas of the ISFSI operations were found to be controlled by written procedures. An extensive effort during this inspection involved the review and determination of adequacy of procedures. This is further discussed in Section 5, "Procedures and Technical Specification Compliance," of this report. The licensee's procedures were found to be comprehensive and to encompass all required tasks related to the ISFSI.

The licensee was required by 10 CFR 72.212 (b)(10) to make records available to the Commission for inspection. The licensee fully complied with this requirement throughout the inspections conducted by the NRC. Records were readily available on request. Personnel knowledgeable in the content of the documents, procedures or records were available to answer questions and provide additional information.

Based on the reviews conducted of the various documents and through interviews with the licensee's staff, the ISFSI programs were found to be adequately developed and documented. The licensee had established a program that complied with the requirements in 10 CFR 72.212 for a general license.

2.3 Conclusion

An extensive review of the licensee's dry cask storage program was completed against the requirements in 10 CFR 72.212 for a general license. The licensee had documented the required evaluations and developed an extensive set of procedures to control work activities associated with the ISFSI. Evaluations had been completed to demonstrate that the design features for the HI-STORM cask system were enveloped by the site specific characteristics of the Columbia Generating Station site.

The licensee conducted a heavy loads movement activity in the train bay using a weighted canister, transfer cask and storage cask. Analysis had determined that modifications to the train bay floor were needed to provide additional support during an earthquake. The licensee performed this activity using the risk assessment techniques allowed for in the new maintenance rule in 10 CFR 50.65 without performing a safety evaluation in accordance with 10 CFR 50.59. This has been determined to be a violation of NRC regulations and is being dispositioned as a Non-Cited Violation (NCV).

3 FUEL VERIFICATION (60855)

3.1 Inspection Scope

The Certificate of Compliance for the HI-STORM 100S cask system provided detailed parameters for the spent fuel that can be stored in the MPC-68 canister. The licensee's program for verifying the spent fuel to be stored at the ISFSI met the applicable requirements in the certificate of compliance was reviewed.

3.2 Observations and Findings

The Certificate of Compliance, Appendix B, provided a list of the various types of spent fuel that had been analyzed and approved for storage in the HI-STORM 100S cask system. Several types of spent fuel and canister models were listed. The specific canister to be used at Columbia Generating Station was the MPC-68. This canister can hold 68 intact spent fuel assemblies. Tables 2.1-1 and 2.1-3 of Appendix B listed the specific arrays and classes for boiling water reactor fuel allowed for storage in the MPC-68 canister. The licensee planned to store spent fuel with four different array/classes: 8x8C, 9x9A, 9x9B and 10x10C. All four array/classes were listed in Table 2.1-3 as acceptable for storage in the MPC-68 canister.

Technical Specifications 2.1 of Appendix B allowed two types of fuel arrangement configurations for the positioning of the spent fuel in the canister. The two options were "uniform" and "regionalized." Regionalized loading allowed for the storage of higher burnup fuel assemblies in the center of the canister. For the first five canisters loaded at Columbia Generating Station, uniform loading of the spent fuel will be used.

Table 2.1-4 of Appendix B provided the allowable burnup values for the spent fuel. Burnup values ranged from 38,300 megawatt days/metric ton of uranium (MWD/MTU) for spent fuel cooled 5 years to 53,900 MWD/MTU for spent fuel cooled for greater than 15 years. Table 2.1-5 of Appendix B provided the allowable decay heat limits for the spent fuel. The decay heat values ranged from 414 watts/assembly for spent fuel that had cooled for 5 years to 347 watts/assembly for spent fuel older than 15 years. The licensee had verified the burnup and decay heat limits for the spent fuel assemblies to be stored. Total heat load calculations for the five casks ranged from 10.8 kW to11.5 kW.

The licensee had developed a cask loading plan for the first five canisters. The cask loading plan was developed under Procedure 9.6.1 "Spent Fuel Selection for Cask Storage," Revision 0. The licensee selected 340 spent fuel assemblies for loading in the first five casks from reactor cycles 1, 2, 5, 6, 7, and 8. The older fuel had a post irradiation time of 15 years. The newer fuel had been cooling for 9 years.

The licensee conducted an operating history records review and a visual examination of the 340 fuel assemblies. Only fuel assemblies with no known fuel damage were considered for the first five casks. No failed fuel which would require a failed fuel canister (MPC-68FF) will be loaded at this time. The spent fuel examinations were performed in accordance with Procedure 6.3.40 "Determination of Fuel Assembly Condition for ISFSI," Revision 0. This procedure established the acceptance criteria in Section 5.0 for classifying spent fuel as damaged. Damaged fuel was defined in Sections 8.1 and 8.5. The acceptance criteria was consistent with the criteria in FSAR Table 1.0.1 "Terminology and Notation," and in Section 2.1.3 for defining damaged fuel. The condition of the fuel assemblies for the first 14 reactor operating cycles had been documented in Calculation No. NE-02-00-08, "Fuel History Data." This calculation included any known information concerning operating cycles that had indications of damaged spent fuel based on offgas system activity levels The fuel history data records in Calculation No. NE-02-00-08 were reviewed as the primary basis for confirming that the fuel was intact. Visual examination was conducted of each of the 340 fuel assemblies using underwater cameras to verify the overall condition of the fuel assembly. Most fuel assemblies still had channels in place which precluded viewing the individual fuel rods. For these fuel assemblies, the licensee relied on the condition of the channel and the fuel history data records to confirm that the fuel assembly did not contain any damaged fuel rods. A small number of fuel assemblies had the channels removed. No additional channels will be removed prior to loading. If the channels were already removed, they will not be re-installed onto the fuel assembly. No problems were identified during the visual examinations. All required records documenting the review of the fuel assemblies had been completed and properly approved.

Based on offgas indications during operations, only cycle 5 had evidence that minor fuel damage had occurred. Assemblies selected from cycle 5 for loading into one of the first five casks were limited to those that had undergone fuel sipping to confirm the assembly was not damaged. Any assemblies suspected as damaged or that had not been sipped were listed in Appendix D "Failed Fuel Data" to Calculation No. NE-02-00-08 "Fuel History Data."

3.3 Conclusion

The licensee had developed a cask loading plan in accordance with approved procedures. Parameters for the 340 spent fuel assemblies selected for loading into the first five casks had been reviewed to verify compliance with the design parameters in the certificate of compliance.

The licensee had performed a review of operating records and conducted a visual examination of the spent fuel assemblies to verify the physical condition of the assemblies selected for loading in the first five casks. All spent fuel assemblies selected were determined to be intact.

4 SPENT FUEL POOL (60801, 60854)

4.1 Inspection Scope

The licensee had developed a program for moving the spent fuel from the storage racks to the canister using procedures consistent with operational procedures for the spent fuel pool. Modifications were made to the refueling bridge computer and grapple to provide access to all the slots in the canister. These modifications and the fuel movement procedures were reviewed with the licensee.

4.2 Observations and Findings

The licensee planned to perform fuel movement activities for loading the canister using Procedure 6.6.6, "MPC Fuel Loading," Revision 1, Procedure 6.3.23, "Handling Irradiated Fuel in the Spent Fuel Pool," Revision 8, and Procedure 2.14.1, "Refueling Bridge Operations." A dummy weighted fuel assembly, stored in the spent fuel pool, was used by the licensee during the pre-operational test to demonstrate the functionality of the fuel handling equipment.

During preparation for the pre-operational test, the licensee discovered that the software which controlled the fuel bridge movement restricted how close the fuel bridge could move to the edge of the spent fuel pool. This safety feature was intended to prevent damage to a spent fuel assembly during movement. This restriction, however, prevented the operator from being able to reach all the slots in a canister located in the cask loading pit because the canister was too close to the spent fuel pool wall. The position of the canister could not be changed. Eighteen slots in the canister were not accessible due to the safety limit controls on the fuel bridge. The licensee also discovered that the design of the grapple on the fuel handling machine resulted in interference between the hoses on the grapple with the edge of the cask when inserting a fuel assembly into a slot near the cask edge. The interference resulted in the assembly not being able to fully seat by 17".

The licensee implemented corrective actions by reprogramming the fuel bridge software and modifying the grapple hoses. The modifications were successful and all slots in the canister were accessible by the grapple.

In accordance with 10 CFR 72.212(b)(4), the licensee had recognized that information provided in the Part 50 FSAR concerning plans for removing spent fuel from the spent fuel pool were inconsistent with the current plans for dry cask storage. In particular, the Part 50 FSAR described the use of the General Electric IF-300 cask, not the Holtec HI-STORM 100S cask system. The IF-300 was originally envisioned as the cask to be used to remove spent fuel from the pool. The cask weighted 80 tons fully loaded with 18 spent fuel assemblies. The IF-300 cask had not been approved for use at an ISFSI under Part 72. On October 30, 2000, the licensee submitted a request to the NRC for an amendment to address the use of the Holtec cask and clarify a statement in the FSAR concerning the reactor building crane. The clarification was that the crane was precluded from traveling over the spent fuel pool racks, but was allowed to travel over the cask loading pit in the spent fuel pool. On October 26, 2001, as supplemented on November 27, 2001, the NRC approved Amendment 174 to the Part 50 licensee for Columbia Generating Station to incorporate the changes requested.

4.3 Conclusion

The fuel bridge safety limit controls prevented the grapple from moving too close to the wall of the spent fuel pool and hoses on the grapple prevented the grapple from completely lowering spent fuel assemblies into several locations along the canister wall. The licensee completed modifications to the fuel bridge software and grapple to provide for access to all locations in the canister.

The Part 50 FSAR had included a description of a cask for removing spent fuel from the spent fuel pool. This cask was not approved for storage of spent fuel at an ISFSI and was smaller and lighter than the Holtec design. An amendment to the plant license was required to incorporate the Holtec cask design into the Part 50 FSAR.

5 PROCEDURES AND TECHNICAL SPECIFICATION COMPLIANCE (60854)

5.1 Inspection Scope

Selected commitments in the certificate of compliance, technical specifications and the FSAR were compared to statements incorporated into procedures to verify that the commitments had been adequately translated into procedural steps. Through interviews with procedure writers, trainers and technicians, observations during the dry-run, and review of the procedures, the effectiveness of the selected procedures to ensure implementation of these commitments was examined.

5.2 Observations and Findings

Procedures for the dry cask storage program were developed in accordance with the licensee's site-wide procedure (SWP) process used to control the development, format and implementation of all procedures at the Columbia Generating Station. This included Procedure SWP-PRO-01, "Description and Use of Procedures and Instructions," Procedure SWP-PRO-02, "Preparation, Review, Approval, and Distribution of Procedures," and Procedure SWP-PRO-03, "Procedure Writers' Manual." Most ISFSI related procedures were assigned to Volume 6 of the Plant Procedures Manual. For purposes of the required pre-operational tests, a special set of procedures had been developed. These procedures were written for each of the pre-operational test demonstrations that were performed and consisted of sections from the actual ISFSI procedures that applied to the particular demonstration. The pre-operational procedures were modified to account for the simulations that were necessary for the demonstrations.

To verify that commitments specified in the certificate of compliance, technical specification, and FSAR had been incorporated into the appropriate sections of the procedures, a number of commitments were selected and the procedures reviewed. This process included interviews with procedure writers, trainers and technician performing the dry run to verify an adequate understanding of the requirement. Table 5-1 lists the commitments selected from the certificate of compliance and technical specifications and the associated procedures where the commitment had been incorporated. Table 5-2 lists FSAR requirements and the associated procedures.

Table 5-1
Certificate of Compliance Requirements and Associated Procedures

CoC/ T.S.	Requirement	Procedure	Rev	Steps
3.1.1.1	Verify canister cavity vacuum drying pressure is within the limit of Table 3-1, 3 torr for ≥ 30 min.	PPM 6.6.7, MPC Processing	0	7.7.23 - 7.7.25
Condition 9	The heat transfer characteristics of the MPC-68 cask system will be recorded by temperature measurement for the first cask placed in service, by any user, with a heat load equal to or greater than 10 kW	PPM 8.3.419, HI-STORM Cask Cooling Test	0	All
Condition 10	A dry run training exercise of the loading, closure, handling, unloading, and transfer of the HI-STORM 100 Cask system shall be conducted by the licensee prior to the first use of the system to load spent fuel assemblies	SFS-06, ISFSI Dry Run Plan	0	All

3.1.2.1	Verify all overpack inlet and outlet air ducts are free from blockage or for overpacks with installed temperature monitoring equipment, verify that the difference between the average overpack air outlet temperature and ISFSI ambient temperature is less than or equal to 126 degree F	OSP-SFS-D101, Spent Fuel Storage Cask Heat Removal System Daily Checks	0	All
3.1.3	The helium exit temperature shall be less than or equal to 200 degree F during unloading and prior to re-flooding	PPM 6.6.9, MPC Cooldown and Weld Removal Systems	0	Multiple times

Table 5-2
FSAR Requirements and Associated Procedures

Section	Requirement	Procedure	Rev	Steps
2.2.1.2	Handling operations of the loaded HI-TRAC transfer cask or HI-STORM overpack is limited to ambient temperatures above 0 degree F.	PPM 6.6.4, HI-STORM System Site Transport	1	4.3
8	Response to abnormal events that may occur during normal loading operations are provided with the procedural steps	ABN-ISFSI ABN- Fuel-Hand, Damage while Handling Fuel ABN-WIND, Tornado/High Winds ABN-RAD-HIGH, Abnormal High Radiation Levels ABN-RAD-RELEASE, Abnormal Release of Radioactivity PPM 6.6.6, MPC Fuel Handling PPM 13.1.1, Classifying the Emergency PPM 13.5.3, Evacuation of Exclusion Area and/or Nearby Facilities	0 2 4 3 2 0 31 26	All
8	The handling of fuel assemblies in the Damage Fuel Container (DFC) shall be performed in accordance with written site-specific procedures	None. Licensee does not plan on using DFCs for the foreseeable future.		
8.8.3	During unloading, the gas sample may indicate that fuel with damaged cladding is present in the canister	PPM 6.6.9, MPC Cooldown and Weld Removal Systems CI 11.18, Transfer and Analysis of Gas Sample from Spent Fuel MPC	0 0	Att. 9.6
10.1.3	Conduct canister lid fit test and inspection prior to actual loading to ensure smooth operations during loading	PPM 6.6.2, MPC Receipt Inspection	1	7.1.19

Table 8.1.5	Hi-Storm100 bolt torque requirements	PPM 6.6.4, HI-STORM System Site Transport PPM 6.6.5, Movement and Transfer Operations of HI-TRAC and HI-STORM in Reactor Building PPM 10.31.9, Installation & Connection of RTDs for Loaded HI-STORMS	1 1	7.3.47c & d 7.5.9 7.10.13 7.10.38 Att 9.3 7.1.9c 7.1.11
Table 8.1.8	Table 8.1.8 provides a sample receipt inspection check list for HI-STORM overpack. User shall develop site-specific receipt inspection checklist	PPM 6.6.3, HI-STORM Overpack Final Assembly Receipt Inspection	2	All
Table 8.1.9	Table 8.1.9 provides a sample receipt inspection check list for the canister. User shall develop site-specific receipt inspection checklist	PPM 6.6.2, MPC Receipt Inspection	1	All
Table 8.1.10	Table 8.1.10 provides a sample receipt inspection check list for the Hi-TRAC. User shall develop site-specific receipt inspection checklist	PPM 6.6.1, HI-TRAC Receiving Inspection	1	All
Table 9.2.1	HI-STORM maintenance schedule	Passport Computerized maintenance program		

The procedures reviewed were found to include the requirements listed in the certificate of compliance, technical specifications, and FSAR. During the pre-operational tests, the technicians demonstrated familiarity with the procedures and systems and readily identified any problem area encountered where the procedures were not consistent with the physical conditions that were being demonstrated. Most of these anomalies were related to the artificiality of the demonstration, such as a gauge already installed on a connection that had not been removed following a prior training session.

A number of specific procedures were reviewed for general content, ease of use, and completeness. Procedures were thorough and included precautions and important reminders of critical parameters. The checklist format of the procedures provided for good documentation of the work activities completed.

Procedure PPM 1.3.40, "Outage Mode Change, Refueling Activity Readiness, and ISFSI Activity Readiness Evaluation," Revision 14, contained a table and checklist of all procedures and technical specifications related to the ISFSI activities that had to be completed prior to initiating fuel movement. This procedure required approval of the shift manager before loading of the canister could begin.

Plant procedures were reviewed to determine whether the licensee had incorporated guidance in the procedures for response to abnormal events that may occur during loading operations. Chapter 8 of the FSAR provided a list of potential events that should be incorporated into procedures for responding to abnormal events.

The events listed in Table 8.0.1 included:

- Cask drop during handling operation
- Cask tip over prior to welding of the canister lid
- Contamination spread from cask process system exhausts
- Damage to fuel assembly cladding from oxidation/thermal shock
- Damage to vacuum drying system vacuum gauges from positive pressure
- Excess dose from failed fuel assemblies
- Excess dose to operators
- Excess generation of radioactive waste
- Fuel assembly misloading event
- Incomplete moisture removal from canister
- Incomplete canister lid installation
- Load drop
- Over pressurization of canister during loading & unloading
- Over stressing canister lift lugs from side loading
- Overweight cask lift
- Personnel contamination by cutting/grinding activities
- Transfer cask carrying hot particles out of spent fuel pool
- Unplanned or uncontrolled release of radioactive materials
- Weld deficiencies from condensation of water on the weld joint

Selected events listed in Table 8.0.1 were verified as being incorporated into either abnormal response procedures or emergency plan implementing procedures. Abnormal conditions such as dropping a fuel bundle, fuel bundle impact with another object or alarms on the refueling bridge were included in abnormal response procedure "ABN-FUEL-HAND," Revision 2. This procedure identified automatic and immediate operator actions for the emergency condition as well as subsequent operator actions. Abnormal response procedure, "ABN-RAD-HIGH," Revision 3, incorporated provisions for increased radiation levels of a loaded canister and transfer cask being removed from the spent fuel pool. Unexpected or abnormal release of radioactivity were addressed in abnormal response procedure, "ABN-RAD-RELEASE," Revision 2, which initiated an evacuation of personnel from the affected area.

Abnormal events caused by crane failure resulting in a cask dropping were not considered a credible event at Columbia Generating Station because the crane had been evaluated and approved as a "single failure proof" system by the NRC with the issuance of Amendment 174 to the Columbia Generating Station license. Tornados and severe winds were incorporated into abnormal response procedure "ABN-WIND," Revision 4, which provided guidance on actions to take depending on the stage of fuel or cask movement at the time of the severe weather.

Emergency procedures for classifying accidents covered refueling incidents in Emergency Plan Implementing Procedure (EPIP) 13.1.1, "Classifying the Emergency," Attachment 5.1, "Emergency Classification Table," Category 1.3, "Refueling Incidents." Conditions were identified for declaring both an unusual event and alert classification if a problem occurred during fuel handling. ISFSI emergency action levels were incorporated into Procedure 13.1.1, Attachment 5.1 as Category 8. These included unexpected increases in radiation levels, damage to a cask confinement boundary, natural phenomena affecting a loaded cask confinement boundary, cask handling accidents, and security events related to the ISFSI.

The licensee was required by Technical Specification 3.1.2 of the Certificate of Compliance, Appendix A, to monitor the temperature of the casks at the ISFSI. Procedure OSP-SFS-D101, "Spent Fuel Storage Cask Heat Removal System Daily Checks," Revision 0, incorporated guidance concerning verification that all storage cask inlet and outlet air ducts were free from blockage and that the difference between the average cask air outlet temperature and the ambient temperature was ≤126 °F.

Each cask will be connected with temperature monitoring equipment. The monitoring equipment will verify that the difference between the average cask air outlet temperature and the ambient temperature is ≤126 °F. The cask inlet and outlet temperature monitors will be connected to a computer output located on the 471' elevation of the turbine building. If the system fails, an operator will physically inspect the casks for blockage every 24 hours. The temperature records will be retained as a permanent plant record in accordance with the plant administrative procedures.

Chapter 8 of the FSAR required procedures developed by the cask user to be reviewed by the certificate holder, i.e., Holtec, prior to implementation. On July 18, 2002, Holtec issued a letter to Energy Northwest providing specific comments on the operating procedures that had been reviewed as of that date. On August 2, 2002, Holtec provided a second letter confirming that the review of the procedures as required by the FSAR, had been completed.

5.3 Conclusion

Procedures consisted of a checklist format that provided for good documentation of work activities completed. Procedures included precautions and important reminders of critical parameters. Commitments from the certificate of compliance, technical specifications and the FSAR had been incorporated into procedures. Implementation of the procedures during the pre-operational tests confirmed the adequacy of the procedures for various work tasks observed during the demonstrations.

The licensee had incorporated written guidance into procedures for abnormal events such as unexpected high dose rates, cask drops, tornado or severe weather conditions, high contamination levels encountered in the work areas, stuck fuel assembly during removal from the fuel racks or during insertion into the cask, and dropped fuel bundles.

6 SAFETY REVIEWS (60857)

6.1 Inspection Scope

The licensee's process for performing safety screenings and evaluations in accordance with 10 CFR Part 50 and Part 72 was reviewed. Selected screenings and evaluations were examined to verify that the licensee was adequately implementing their program.

6.2 Observations and Findings

The licensee performed 10 CFR 72.48 and §50.59 screenings and evaluations using Procedure SWP-LIC-02, "Licensing Basis Impact Determinations," Revision 3. The index of specific 10 CFR 72.48 screenings and evaluations performed by the licensee for design changes to the spent fuel cask system was reviewed. Very few of the 72.48 evaluations had been performed by the licensee. The majority had been performed by Holtec and were previously examined and found acceptable by the NRC during inspections at Holtec and their manufacturer, U.S. Tool and Die. Inspections of Holtec were performed in December 1997 (Inspection Report 71-0784/97-216), September 2001 (Inspection Report 72-1014/01-201) and May 2002 (Inspection Report 72-1014/02-202). Inspections at US Tool and Die were performed in June 1999 (Inspection Report 72-1008/99-201) and February 2002 (Inspection Report 72-1014/02-201). Of the 72.48 screenings and evaluations performed by Columbia Generating Station, eight were selected for review. The 72.48 screenings and evaluations reviewed were found to be adequately dispositioned.

One safety evaluation selected for review included an unreviewed safety question (USQ). Safety Evaluation SE-00-0026, "FSAR Amendment: Spent Fuel Cask Operations," discussed statements in the Part 50 FSAR concerning the use of the General Electric cask (GE IF-300 model) as the cask for moving spent fuel from the spent fuel pool. The GE IF-300 cask was an NRC approved cask under 10 CFR Part 71 for transporting spent fuel from reactor facilities to a fuel processing facility. This cask weighed 80 tons fully loaded compared to the estimated 120 tons for a loaded canister and HI-TRAC transfer cask being removed from the spent fuel pool with water in the canister. The licensee recognized that changes to the Part 50 FSAR were needed to address this heavier cask and that descriptions provided in the Part 50 FSAR concerning the spent fuel pool design, limitations on movement of the reactor building crane over the spent fuel pool, and movement of a spent fuel cask over safety related equipment needed to be updated. The licensee submitted a request to the NRC on October 30, 2000, for a revision to the Part 50 FSAR. On October 26, 2001, the NRC issued Amendment 174 to incorporate the requested change related to the use of the new cask designs.

Five screenings reviewed contained minor omissions or errors that were identified to the licensee for correction. As an example, the licensee had completed 10 CFR 72.48 Evaluation Control No. ISFI-00-0001, Revision 0, dated November 8, 2002, to evaluate a change to the ISFSI FSAR requirement that the lid for the storage cask not be lifted more than 2 feet above a loaded canister. The licensee's evaluation was determined to be adequate, however, the ratio of the crane hook rating to the weight of the lid was incorrectly stated as 10:1.2 instead of 10:1. The licensee initiated the necessary changes to correct the minor errors to the screenings.

6.3 Conclusion

The licensee had implemented a program to perform safety screenings and evaluations in accordance with the requirements in 10 CFR 50.59 and §72.48. Selected screenings and evaluations performed by the licensee were reviewed and found to be adequately dispositioned.

7 HEAVY LOADS (60854)

7.1 Inspection Scope

The licensee was required to demonstrate the adequacy of their heavy loads program for moving the spent fuel from the spent fuel pool to the ISFSI. In addition, the licensee's plans for using a new ancillary device underneath the casks to mitigate the effects of a seismic event was evaluated.

7.2 Observations and Findings

The reactor building crane was a single trolley seismic Category I overhead crane with a 125-ton capacity main hoist and a span of approximately 126 ft. The reactor building crane was used to move a loaded canister inside the transfer cask from the spent fuel pool to the cask washdown area on the 606' elevation, then down the hatch to the storage cask located on the 441' elevation. The loaded canister was then lowered from the transfer cask into the storage cask and the transfer cask was returned to the 606' elevation. The loaded storage cask was then moved from its position in the train bay to the outside of the reactor building using a tug. The storage cask was moved on six sets of rollers. Each roller had a capacity of 150 tons with a design load limit of 300 tons. The rollers had been tested to 250 tons. A mechanical cask crawler was connected to the storage cask outside the reactor building and was used to move the loaded cask to the ISFSI pad over a roadway that had been specially built for the weight of the cask and crawler. A loaded storage cask weighed 180 tons. The load on the road, consisting of both the loaded storage cask and the crawler was 265 tons.

On April 11, 1996, the NRC issued Bulletin 96-04, "Movement of Heavy Loads over Spent Fuel, over Fuel in the Reactor Core, or over Safety Related Equipment." All operating nuclear power plants were required to respond to the bulletin. This bulletin required licensees to address a number of issues related to heavy loads. Energy Northwest responded to the bulletin on May 10, 1996, and over the period from 1996 through 2001 provided the NRC sufficient information related to the reactor building crane to justify the crane as a "single failure proof" crane in accordance with NUREG 0554, "Single Failure Proof Cranes for Nuclear Power Plants." On October 26, 2001, the NRC issued Amendment 174 to the Columbia Generating Station license which accepted the reactor building crane as a single failure proof crane.

The licensee had analyzed the various loads associated with the dry cask storage project. The loads were consistent with the estimates provided in Section 2.2 and Tables 8.1.1, 8.1.2, 8.1.3, and 8.1.4 of the FSAR.

Applicable portions of the NRC Safety Evaluation Report pertaining to structural design criteria, individual loads, load combinations, lifting devices, maintenance program, and quality assurance program were reviewed to verify incorporation into the heavy loads program. Procedures were found to contain sufficient guidance and incorporated applicable requirements specified in the FSAR, ANSI documents, the certificate of compliance, and the NRC Safety Evaluation Report.

The licensee's program for implementing control of heavy load lifts and maintenance of crane equipment was described in several procedures and documents The following were reviewed to evaluate the adequacy of the licensee's heavy loads program:

- Procedure 10.4.12, "Crane, Hoist, Lifting Device and Rigging Program Control," Revision 15
- Procedure 10.4.5, "Reactor (MT-CRA-2) and Turbine Building (MT-CRA-1) Overhead Traveling Crane Inspection, Maintenance and Testing," Revision 11
- Procedure 10.4.14, "Miscellaneous Load Handling," Revision 7
- Procedure 10.25.11, "Reactor Building Crane MT-CRA-2 Electrical Maintenance Procedure," Revision 12
- Plant Procedures Manual 6.6.7A, "MPC Processing - NRC Dry Run Demonstration," Revision 0
- Safety Evaluation Report, "HOLTEC International, Hi-Storm 100 Cask System"

Also reviewed was selected guidance contained in several American National Standards Institute (ANSI) standards. The information in the licensee's heavy loads program was found to be consistent with the requirements in the ANSI standards.

- ANSI B30.2, Overhead and Gantry Cranes
- ANSI B30.5, Crawler, Locomotive, and Truck Cranes
- ANSI B30.9, Slings
- ANSI B30.10, Hooks
- ANSI B30.16, Overhead Hoists
- ANSI B30.20, Below the Hook Lifting Devices
- ANSI N45.2.15, Hoisting, Rigging and Transporting of Items for Nuclear Power Plants
- ANSI N14.6, for ISFSI Material, Procurement, Testing, Fabrication, and NDE During Manufacture

As part of the pre-operational tests to demonstrate the adequacy of the heavy loads program, the NRC observed several heavy lifts of the transfer cask containing either an empty canister or a weighted canister. These included:

- Transferring a weighted canister from the storage cask to the transfer cask
- Raising the transfer cask loaded with a weighted canister from the train bay to the elevation 606'
- Moving a canister and transfer cask into the spent fuel pool
- Lowering a canister lid into place on the canister in the spent fuel pool
- Removing the canister and transfer cask from the spent fuel pool
- Returning the transfer cask loaded with a weighted canister to the train bay
- Transferring the weighted canister from the transfer cask into the storage cask
- Moving the storage cask with the weighted canister from the reactor building to the ISFSI

All demonstrations were preformed safely with no heavy loads or rigging problems. Discussions were held with the crane operators who were found to be very knowledgeable and safety conscious. The crane operators had many years of experience in dealing with heavy loads.

The licensee had established a safe loads path for movement of the transfer cask and canister. Field Change Request (FCR) 97-0180-1-22 and Figure 9.1-5 of the Part 50 FSAR provided drawings showing the safe load path. The licensee stayed within the designated safe load path for all pre-operational tests and for the loading of the first cask.

When the loaded transfer cask was placed in the cask washdown pit on the 606' level or positioned on top of the storage cask in the train bay, elevation 441', a seismic event could cause the cask to rock. The licensee planned to use a device developed by Holtec, called HERMIT (Holtec Earthquake Response Mitigator) to prevent the cask from rocking by allowing the cask to slide on the floor, thereby ensuring that tip-over of the cask was not a credible event. HERMIT consisted of two steel sheets separated by a low friction material to allow slippage between the two sheets. The loaded transfer cask would be placed on HERMIT while in the washdown pit. A second HERMIT was used on the train bay floor under the storage cask.

Since Columbia Generating Station was the first site to use HERMIT to mitigate the effects of a seismic event, the NRC staff at headquarters performed a special analysis of the design. This analysis involved the review of a number of Holtec documents by the NRC seismic staff to determine if the assumptions and design basis applied to HERMIT were valid for the planned use.

A review was performed of Holtec Report No. HI-2022875, "Material Testing Report for Calibrated Low Friction Material's (CLFM) for Columbia Generating Station HERMITS." This report determined the coefficient of friction for the HERMIT devices during a seismic event based on tests performed by Holtec. The report stated that calibrated low friction material was used between the two steel sheets to allow slippage between the steel sheets during dynamic events. Facilitating the slippage prevented cask tip-over or lift-off from the base surface. In the tests performed on Nylatron as the calibrated low friction material, the lowest coefficient of friction was determined to be 0.125 with the highest coefficient of friction as 0.205. Holtec selected the range 0.125 to 0.25 as the criterion to meet under all postulated dynamic loads.

For the HERMIT used in the train bay, Holtec Report No. HI-2022823, "Hermit Deployment at Energy Northwest Columbia Generating Station Elevation 441'," presented the seismic analyses for the HI-TRAC/HI-STORM/loaded canister stacked configuration on the 441' elevation. The report established a maximum horizontal excursion of the cask assembly top center point relative to the floor slab to be 25" to ensure that there would be no lateral impacts with any part of the hatch opening at the 471' elevation. The lower limit on the coefficient of friction must ensure that the computed horizontal excursions during a seismic event did not exceed a lesser value in order to provide additional margin against uncertainties. HERMIT's calibrated friction material was expected to provide a coefficient of friction within the range of 0.125 to

0.25. For conservatism, in the stacked configuration, the lowest coefficient of friction was assumed to be 0.1 which would result in the maximum sliding.

Calculations concluded that with the assumed minimum coefficient of friction, the lateral movement was less than 10". This was well below the 25" maximum. With the assumed maximum coefficient of friction, a maximum rotation from the vertical was calculated to be less than 1.5 degrees which ensured that there would be no tip-over. The NRC staff determined the use of HERMIT on the 441' elevation was adequate based on the available margin for uncertainties and the assumed coefficient of friction, which was conservative when compared with the search in the literature for the friction coefficients and the test results for the material presented in Holtec Report No. HI-2022875.

For the HERMIT planned for use on the 606' elevation, Holtec Report No. HI-2012795, "Hermit Deployment at Energy Northwest Columbia Generating Station Elevation 606'," was reviewed. This report presented the seismic analyses for use of HERMIT on the 606' elevation. The report calculated that a loaded transfer cask sitting on HERMIT on the 606' elevation could move laterally up to 25" during a seismic event. Calculations showed that using the minimum coefficient of friction, the lateral movement was less than 25". The NRC staff determined that the integrity of the cask and the floor was adequate, such that the cask would slide much less than 25" and would not tip-over on the 606' elevation. This was based on an assumed coefficient of friction, which was conservative when compared to actual test results for the material presented in Holtec Report No. HI-2022875. This conclusion was also based upon independent seismic and drop analyses performed under other ongoing projects in the NRC's Office of Research.

7.3 Conclusion

The procedures governing the heavy load lift operations appropriately contained the requirements and guidance from the FSAR and national standards for maintenance and testing to ensure the ability of the equipment to support the anticipated loads required during the dry cask storage program activities.

The heavy loads procedures appropriately addressed cask lift limits. The expected component weights were bounded by the weight values established in the FSAR.

The licensee's planned use of an ancillary device placed under the casks on the 441' elevation and 606' elevation to mitigate the effects of a seismic event were reviewed by the NRC and found acceptable.

8 HYDROSTATIC TESTING/DRYING/HELIUM BACKFILL (60854)

8.1 Inspection Scope

The licensee was required to perform a hydrostatic test of the canister after the lid is welded, then vacuum dry and backfill the canister with helium. The licensee's equipment and procedures were reviewed. Demonstration of the equipment during the pre-operational testing was observed.

8.2 Observations and Findings

A hydrostatic test of the canister was required to verify the integrity of the canister boundary. Specific criteria was provided in the FSAR, Section 9.1.2.2.2, concerning the testing process and the acceptance criteria. Upon completion of the hydrostatic test, the licensee was required to vacuum dry the canister to less than 3 Torr pressure. This requirement was specified in Technical Specification 3.1.1 and Table 3-1 of Appendix A of the Certificate of Compliance. After the drying criteria was met, the licensee was required to backfill the canister with helium and complete the sealing process. Acceptance criteria for the helium backfill was also specified in Technical Specification 3.1.1 and Table 3-1. The licensee was required by Condition 10 of the Certificate of Compliance to demonstrate the process for hydrostatic testing, drying and helium backfilling as part of their pre-operational testing.

The licensee demonstrated the hydrostatic testing, drying and helium backfilling during the NRC observed pre-operational tests. The licensee demonstrated the equipment using detailed checklist procedures. Personnel were well trained and knowledgeable on the use of the equipment and on the required technical specification limits that applied. Several procedures related to the hydrostatic testing, drying and helium backfill process were reviewed. These included Procedure 6.6.12, "Vacuum Drying System Operation," Revision 1, Procedure 6.6.13, "Helium Backfill System Operation," Revision 1, Procedure 6.6.7, "MPC Processing - NRC Dry Run Demonstration," Revision 0, and Section 7.3.2, "Hydrostatic Test," in Procedure 6.6.7, "MPC Processing," Revision 2. All procedures were found to be comprehensive, easy to understand and implement, and incorporated the acceptance criteria from the technical specifications.

Holtec Purchase Specification, PS-1405, Revision 0, was reviewed to verify that the requirements specified in the FSAR had been incorporated into the purchase specification for the helium backfill system. The review included verifying that the equipment could achieve the required backfill level specified in the technical specification, the proper calibration was specified for the system pressure gauge, helium backfill requirements were properly specified in the controlling documents, and helium backfill pressure limits were within the limits specified in Table 3-1 of the Technical Specifications. Also reviewed was Purchase Order 00311477 to verify the specifications for the purchase of the helium gas. The purchase order specified the correct purity requirements for the helium, required the helium cylinders to be filled at the supplier's facility in accordance with requirements of the purchase order and required that prior to use, the serial numbers were recorded and verified for the helium cylinders.

Procedure PPM 6.6.7, "MPC Processing - NRC Dry Run Demonstration," Revision 0, incorporated the canister helium density requirements specified in Table 3-1 of the technical specifications. The calibration verification for all measuring and test equipment, including the specific pressure gauge to be used to confirm technical specification requirements, was appropriately addressed in Procedure 6.6.7.

Attachment 9.4 of Procedure 6.6.7 properly established the canister backfill helium mass limits and provided for verification of the calculated values.

Based on the review of the Holtec purchase specification and observation of the pre-operational test, it was determined that the helium backfill system was capable of achieving the required backfill level, that appropriate procedural controls had been implemented for the calibration of the associated measuring and test equipment, and that Procedure 6.6.7 properly established the canister helium backfill mass limits.

During the dry run demonstration of the helium backfill process, the technique for the helium leak testing of the final closure welds were demonstrated on the mockup canister. The helium leak testing was performed by experienced contract personnel qualified to the nondestructive testing standard, SNT-TC-1A, "Recommended Practice for Nondestructive Testing Personnel Qualification and Certification." These individuals will be performing the actual helium leak testing on the loaded casks. Performance by both the individuals coordinating the pre-operational test and the individuals performing the leak tests demonstrated a good understanding of the requirements for performing helium leak tests and the acceptance criteria that applied to the testing.

8.3 Conclusion

The licensee demonstrated the capability to perform hydrostatic testing, drying, and helium backfill of a canister during the pre-operational tests. Procedures and purchase orders were reviewed to verify that equipment associated with these activities was capable of achieving the required limits specified in the technical specifications.

9 WELDING/NONDESTRUCTIVE TESTING (60854)

9.1 Inspection Scope

The licensee had developed a program for welding the canister lid, performing nondestructive examinations (NDE) of the welds, and removing the lid from a canister, should opening a canister be necessary. Equipment, procedures, and personnel qualifications were reviewed. Demonstrations of the welding, NDE and lid cutting were observed during the pre-operational testing program.

9.2 Observations and Findings

The licensee had assembled a dedicated contractor welding and nondestructive examination team and had provided materials and facilities for completing the pre-operational testing using canister mockups. The material type, welding equipment, and procedures used on the mockups were the same as those to be used on the actual canisters. Therefore, the welding equipment, nondestructive examination methods and techniques, cutting equipment, and procedures used on the mockups would be directly transferrable to the welding and examination of the actual canisters.

A review was performed of the welding procedure specifications, procedure qualification records, welder and welder operator performance qualifications, nondestructive examination procedures, nondestructive examination personnel qualifications, and the cutting procedure. All tasks were governed by written procedures which provided orderly and sequential steps and contained provisions for signoffs to document completion of the specified task. The following documents were reviewed:

- Welding Procedure Specification WPS-08-08-TS-901, Revision 1, for machine gas tungsten arc welding and manual shielded metal arc welding and gas tungsten arc welding.
- Procedure Qualification Records PQR 08-08-TS-001, January 26, 1999; PQR 08-08-TS-002, August 15, 2000; PQR 08-08-TS-091, March 20, 2002; and PQR 8.8.6-OKG, June 3, 1998.
- Welder and Welding Operator Performance Qualification Records for Stamp Numbers MLB-4156, DDT-1612,GML-6076, BPS-6895, and DGA-2658.
- Certificate of Qualification and Vision Examination Records for the Nondestructive Examination Level II Examiner (liquid penetrant).
- Procedure QAP 9.3, "Workmanship and Visual Inspection Criteria For ASME Welding," Revision 12.
- Procedure QAP 9.6, "Liquid Penetrant Inspection Procedure."
- Procedure QAP 9.16, "High-Temperature Liquid Penetrant Inspection Procedure, Using Color Visible/Solvent Removable Penetrant Technique, Temperature Range: 100°F - 300°F," Revision 1.
- Plant Procedures Manual 6.6.7A, "MPC Processing - NRC Dry Run Demonstration," Revision 0.
- Work Procedure Traveler For MPC Closure Welding," Traveler No. 31099, Revision 0.
- Drawing 1402, Sheet 1, Revision 15, Sheet 2, Revision 15, Sheet 3, Revision 14, Sheet 4, Revision 12, Sheet 5, Revision 10, and sheet 6, Revision 14, "Hi-Star 100 MPC-68 Construction."

The welding and nondestructive examination documents were found to be consistent with the requirements specified in the 1995 Edition through 1997 Addenda of the ASME Boiler and Pressure Vessel Code, Sections V and IX.

The mockup canister used for the welding and nondestructive examination activities was identical to the canisters (i.e., material, diameter, thickness) to be used for the spent fuel, except for height. The mockup canisters were approximately 3 feet in length, which was sufficient for the welding, nondestructive examination, and cutting demonstrations.

The physical characteristics, in terms of work area for the welders and nondestructive examination personnel (e.g., floor height), were arranged to be very similar to what would be expected while working on an actual canister surrounded by scaffolding. The area where the welding and nondestructive examinations were performed was cordoned off to simulate actual conditions expected in the fuel storage building. Radiological controls were instituted and each worker entering the cordoned off area had to "dress-out" in the prescribed protective clothing. Conversely, each person exiting the area had to remove the protective clothing in the prescribed manner.

All welding was performed using the shielded metal arc welding and gas tungsten arc welding processes, with machine gas tungsten arc welding being the predominant process used. Two robotic welding heads were used during machine gas tungsten arc welding. There was excellent 2-way communication between the welding machine operators and welder observers assigned to each head. Welders were appropriately alternated to reduce worker fatigue.

A pre-job briefing, which was well organized and thorough, was conducted prior to initiation of welding, and at the beginning of each work shift. A step-by-step procedure controlled all phases of the work. There was good discussion and interaction within the assigned group during the pre-job briefing and the actual work activities. The welders, welding operators, and nondestructive examination personnel had considerable experience and displayed good knowledge in all aspects of the welding and examination processes.

The inspectors verified that all essential variables for the gas tungsten arc welding process specified in Section IX of the ASME Code were identified in the welding procedure specification, the essential variables were appropriately qualified, and the essential variables were followed (e.g., base metal, filler metal, preheat, postweld heat treatment, gas, and technique). The inspectors also verified that the nonessential variables for that process were identified in the welding procedure specification and that the welders generally adhered to them. The welders were particularly cognizant of gas flows, heat input and interpass temperatures.

The canisters were designed, to the maximum extent practical, in accordance with the ASME Boiler and Pressure Vessel Code, Section III. However, since the design of the canisters did not allow for full compliance with all aspects of Section III of the ASME Code (e.g., partial penetration welds as opposed to full penetration welds of the lid-shell-weld and closure ring welds, and surface examination rather than volumetric examination of the lid-to-shell weld, closure ring welds, and vent and drain cover plate welds), exceptions, with justifications and compensatory measures were established. These were addressed in the FSAR and found to be acceptable in the NRC Safety Evaluation Report for the Hi-Storm 100 Cask System.

The alternate liquid penetrant examinations performed on the root pass, each intermediate layer (which consisted of no more than 3/8-inch of weld depth), and final cap weld were observed. The examinations were performed using both the room temperature and high-temperature liquid penetrant procedures.

If the technical specification requirements for vacuum pressure or helium leak rate could not be met after the lids had been welded on, the licensee could be required to cut open the canister and return the spent fuel to the spent fuel pool. In order to provide for this possibility, the licensee established a procedure and traveler for conducting these operations. A cutting machine was onsite that could be mounted on top of the canister to cut through the welds of the closure ring and shell-to-lid welds. As part of this demonstration, the licensee performed cutting operations on a previously welded mockup canister to verify the adequacy of the cutting procedure, traveler, and cutting equipment. These operations were observed by the inspectors and found to be successful.

9.3 Conclusion

The procedures governing the operations for canister lid welding, nondestructive testing of the lid welds and removal of the lid, should that be necessary, were detailed, thorough, and appropriately qualified. Welding and nondestructive examination personnel were properly qualified. Observation of the welding, nondestructive examinations and cutting operations on the mockup canisters verified personnel skills, procedure adequacy, and equipment capability. The nondestructive examinations verified the capability of the welders to produce high quality welds.

10 HEALTH PHYSICS (60854, 83750)

10.1 Inspection Scope

Provisions for controlling radiological conditions and maintaining radiological exposures to workers as low as reasonable achievable (ALARA) during work associated with the dry cask storage project were reviewed. This included reviews of radiation work permits (RWP), instrumentation, training, procedures and dosimetry.

10.2 Observations and Findings

In compliance with 10 CFR 72.212 (b)(6), the licensee had reviewed their radiation protection program and procedures to ensure the adequacy of the program for the activities associated with the dry cask storage project. The review was documented in the licensee's "ISFSI 10 CFR 72.212 Evaluation Report," Revision 0. Several new radiological procedures were developed. These included procedures for conducting radiation surveys during cask loading operations to ensure compliance with the radiation limits specified in Technical Specifications 3.2.1 through 3.2.3 of Appendix A of the Certificate of Compliance.

To provide radiological controls during work activities, the licensee had developed several RWPs specific to activities associated with the ISFSI project. These RWPs required that workers receive specific pre-job ALARA briefings prior to entering the radiologically controlled area to perform work. The requirement for a pre-job ALARA briefing had been specified in the FSAR, Section 10.1, "Ensuring That Occupational Radiation Exposures Are ALARA." In order to ensure that personnel could not enter the radiologically protected area on an RWP prior to receiving a pre-job ALARA briefing, the health physics staff was required to physically enter the date that each person received the ALARA briefing into the computerized database. Without entry of a date in the computer, the individual could not gain access into the radiologically controlled area. The RWPs also contained specific information concerning expected dose rates. This information was based on analyses performed by Holtec and information obtained from other licensees that had loaded Holtec casks. The licensee had estimated the doses for specific activities during the fuel loading campaign for craft, engineering, labor and HP support personnel.

Training had been developed and provided to the health physics staff concerning the expected radiological conditions that would be encountered during a cask loading campaign. Training on expected dose rates during cask loading, storage and unloading was required by FSAR Section 10.1. The training session included detailed information on the cask system and specifically addressed issues such as the cask loading process, posting and labeling, external and internal dose concerns, and contamination controls that would be needed during cask loading. Additionally, one lead health physics technician had attended a Holtec dose reduction conference held June 12-13, 2002, at Hatch Southern Nuclear Operating Company. Specific information related to dose rates, contamination and cumulative person-rem experienced during actual cask loading activities was provided. This information was incorporated into Columbia Generating Station's health physics training and used to prepare dose estimations for the first fuel loading campaign.

During the dry run, the health physics staff performed numerous demonstrations of health physics controls that would be implemented during the actual loading of the casks. These included establishing health physics boundaries, performing air sampling, conducting contamination and dose rate surveys for both gamma and neutron radiation, preparing a detailed RWP and providing proper alarming and whole body dosimetry. The health physics staff and craft personnel were well versed and knowledgeable of the cask design and the specific areas where the potential existed for radiological problems or high dose rates. The health physics staff demonstrated good radiological practices and followed procedures during the demonstrations.

Monitoring of radiation levels during cask loading activities on the Elevation 606' refuel floor would be accomplished through both fixed and portable radiation instrumentation. Four fixed alarming rate monitors (ARM) were located at strategic locations on the refuel floor to ensure adequate warning during a radiological event. These monitors had alarm set points ranging from 15 mR/hr to 50 R/hr. To ensure that the ARMs were functioning correctly, the licensee was required by procedure CSP-INST-H201, "Chemistry Shift Channel Checks," to perform channel checks twice per day in accordance with Surveillance Requirement (SR) 3.3.6.1.1. Additionally, as required by SR 3.3.6.1.4, the licensee was required to perform a channel calibration at intervals not to exceed 18 months. Interviews with members of the chemistry staff confirmed that the daily channel checks and channel calibrations of the ARMs on the refuel floor were being performed at the required frequencies.

The HP staff had several ion chambers, teletectors and neutron rem balls for use in evaluating area gamma and neutron dose rates. Additional radiological monitoring equipment included portable air samplers, continuous air monitors and several portable ARMs.

In order to evaluate the radiological conditions during the fuel loading campaign and prevent the spread of loose contamination, the licensee established restricted boundaries around the spent fuel pool and cask transfer pathway. A geiger mueller and a zinc sulfide scintillation detector were used to quantify removable beta, gamma and alpha contamination. Additionally, 10 percent of the contamination smears were sent to the licensee's chemistry laboratory for quality assurance checks. A review of the scintillation detector's records was completed to verify that required calibrations and source checks had been completed.

Plant Procedures Manual (PPM) 11.2.24.1, "Radiation Protection Work Routines," Step 3.2.7, required the licensee to perform daily instrument response checks of all portal monitors and hand held survey instruments prior to use. Health Physics Instruction Manual 0.16, "Radiation Protection Portable Instrumentation Use and Calibration Guidelines," required survey instruments to be calibrated at least semi-annually. Review of records and discussions with health physics staff personnel confirmed that the instruments used during cask loading had been source checked daily and were in calibration.

During loading of a cask, the neutron spectrum will be thermalized by the presence of water in the cask. After the lid is welded in place, the cask is drained and filled with helium. When the water is drained, the radiation emitted from the cask will consist of a higher energy neutron spectrum. The licensee had evaluated this condition in Radiation Protection Technical Basis Document 02-05, "Neutron Survey Instruments for ISFSI Surveys," dated July 16, 2002. This document evaluated the suitability of the licensee's instrumentation for performing neutron radiation surveys of the HI-TRAC and HI-STORM casks. The technical basis document concluded that "the Eberline NRD neutron survey instrument, calibrated with reference to a moderated californium-252 standard, was a satisfactory instrument for neutron surveys of the ISFSI shields (HI-TRAC and HI-STORM)." The licensee had designated a rem ball as a transfer standard and had sent the rem ball to a National Voluntary Laboratory Approved Program (NVLAP) accredited calibration laboratory for calibration to a moderated californium-252 source. This transfer standard was then used to calibrate the americium/beryllium source at the licensee's facility for use in calibrating the rem balls used onsite. Calibrations were performed using procedure PPM 11.2.9.26, "Eberline Model ASP-1."

All personnel were issued NVLAP approved thermoluminescent dosimetry capable of monitoring beta, gamma and neutron dose. Prior to entering radiologically controlled areas, personnel were required to log in on the appropriate RWP and obtain a digital alarming dosimeter. The digital alarming dosimeter was provided to track exposure on a real time basis and to alarm in the event personnel were in areas that exceeded the set points for either area dose rates or accrued dose.

The licensee had established an environmental TLD program to monitor radiation levels around the ISFSI. The environmental TLD monitoring program consisted of three sample locations reasonably close to the ISFSI site to establish a baseline background radiation level prior to ISFSI operations. TLDs were exchanged and analyzed on a monthly and annual frequency. The program was operated in conjunction with the Washington State Department of Health. During operation of the ISFSI, the licensee planned to maintain 10 TLDs located around the perimeter of the fence surrounding the ISFSI. The licensee had also performed and documented a baseline soil analysis for future decommissioning purposes and included the results in their 10 CFR 72.212 evaluation.

10.3 Conclusion

The licensee was implementing their site radiation protection program for activities associated with the ISFSI. Several additions were incorporated into the program to address cask-specific radiological conditions. These included the development of procedures to address radiological surveys of the loaded cask, development of specific radiation work permits that defined radiological controls required during the different phases of cask loading and movement, the addition of dosimetry around the ISFSI and implementation of training specific to the radiological conditions that would be encountered during cask loading and movement.

11 EMERGENCY PLANNING (60854)

11.1 Inspection Scope

The licensee's 10 CFR Part 50 emergency preparedness program was reviewed to verify that adequate provisions had been incorporated for response to emergencies at the ISFSI. Emergency action levels were reviewed to verify that credible emergencies involving cask loading and ISFSI operations could be properly detected and classified. Emergency procedures, training, drills/exercises and arrangements with offsite support organizations were reviewed to verify that adequate planning for response actions had been completed.

11.2 Observations and Findings

Columbia Generating Station incorporated their emergency planning program for the ISFSI into their 10 CFR Part 50 site wide emergency planning program, as allowed for in 10 CFR 72.32(c). The site Emergency Plan, Revision 33, included provisions for responding to emergencies at both the reactor facility and the ISFSI. When a licensee incorporates provisions for responding to ISFSI related emergencies into the existing Part 50 site emergency plan, the licensee is required by 10 CFR 72.212(b)(6) to evaluate the site emergency plan to ensure its effectiveness is not decreased and to incorporate the necessary ISFSI related information into the emergency planning program to ensure compliance with the requirements in 10 CFR 50.47. The licensee had completed and documented the required 10 CFR 72.212 evaluation in the "ISFSI 10 CFR 72.212 Evaluation Report," Revision 0. Discussions were held with representatives of the licensee's emergency planning staff to review the extent to which cask loading and ISFSI operations had been evaluated within the context of the site wide emergency planning program and incorporated into the emergency response procedures and training.

The ISFSI was added to the list of principal structures in the site emergency plan. Design basis accidents analyzed for the ISFSI, as described in the FSAR, Sections 11.2 and 12.2.1, were incorporated into the site emergency plan. The various accidents described in the FSAR included transfer cask handling accident, storage cask handling accident, tip over, fire accident, partial blockage of canister vent holes, tornado, flood, earthquake, 100 percent fuel rod rupture, confinement boundary leakage dose calculations, explosion, lightning, 100 percent blockage of air inlets, burial under debris, and extreme environmental temperature. Not all of these events were found to elevate to the significance of requiring classification as an emergency condition. Some of these events were incorporated into the site abnormal event response procedures.

Of the events that were determined to meet the criteria for incorporation into the emergency plan, three new Emergency Action Levels (EALs) were developed and incorporated into Table 4.1 of the site emergency plan and into the Emergency Plan Implementing Procedures (EPIP) 13.1.1, "Emergency Classification," Revision 31 and EPIP 13.1.1A, "Emergency Classification-Technical Basis," Revision 10. These new EALs included unexpected increase in ISFSI radiation, damage to a loaded cask confinement boundary and a confirmed security event with potential loss of level of safety to the ISFSI. Each of the EALs results in declaration of an "unusual event". In addition, EPIP 13.1.1 included refueling incidents in Section 1.3 of Attachment 5.1, "Emergency Classification Table." Conditions for both an unusual event and alert were included in the table for refueling incidents.

The licensee requested NRC approval of the new EALs for incorporation in the site emergency plan by letter dated January 11, 2002. The NRC determined that the proposed changes met the requirements of 10 CFR 72.32(c), 10 CFR 50.47(b) and 10 CFR Part 50, Appendix E, and approved implementation of the proposed EALs in a letter dated April 16, 2002. Notification to offsite agencies would be required for the new EALs, since offsite agencies were notified during the declaration of Unusual Events.

A review of other site emergency procedures, developed for the Part 50 emergency response program, found existing provisions in the emergency procedures were adequate to cope with the different types of emergency conditions that could occur at the ISFSI. The procedure reviewed included response team actions, notifications, technical support, evacuations, transport of a contaminated and injured person to the hospital and authorizing emergency exposure limits during an emergency response. Existing provisions in the emergency procedures for support from offsite organizations were found to be applicable to the ISFSI.

The licensee established a list of emergency response positions and identified associated training for each position. The list of individuals qualified for selected positions, including shift supervisor, control room operator and security manager, were reviewed. Initial training and re-qualification was verified in the emergency plan database. Initial and specific ISFSI training had been conducted during the spring 2001. Control room operators were given training by emergency preparedness staff as part of their refresher re-qualification training. Records reviewed indicated that all training was current. Interviews were conducted with selected operations and health physics personnel to determine their level of knowledge of the various types of accidents that could occur during loading the ISFSI and during ISFSI operations. Personnel interviewed were aware of the types of accidents that could occur and knew the appropriate response actions to take.

The site wide emergency plan required periodic drills and exercises to be conducted in order to test the overall state of emergency preparedness. There was no specific requirement in the emergency plan to conduct a drill at the ISFSI or during handling of spent fuel while loading a cask or moving a cask from the fuel building to the ISFSI pad. At the time of the inspection, no emergency drills or exercises had been conducted using cask loading or ISFSI operations as a scenario.

11.3 Conclusion

The licensee incorporated provisions for responding to an emergency at the ISFSI into their existing site wide emergency planning program. The site emergency plan and procedures were revised to incorporate new emergency action levels for the accidents described in the FSAR that would require declaration of an emergency.

Training had been completed for emergency responders. Interviews of personnel confirmed that individuals were adequately trained and knowledgeable of response actions. Existing provisions for response from offsite support organizations during an emergency were found to be adequate for events that could occur at the ISFSI.

12 FIRE PROTECTION (60854)

12.1 Inspection Scope

The licensee was required to address site specific considerations for a potential fire or explosion that could effect a loaded cask and to have provisions for controlling flammable liquids and for taking action to respond to a fire including arrangements with offsite fire department support.

12.2 Observations and Findings

The licensee was required by Technical Specification 3.4.5 of Appendix B of the Certificate of Compliance to analyze site specific conditions that have the potential for a fire or explosion that could affect a loaded cask and to limit the onsite transporter fuel tank to no more than 50 gallons of diesel while handling a loaded cask.

The licensee had developed three primary fire protection procedures that related to the ISFSI. These were procedure SWP-FFP-01, "Nuclear Fire Protection Program," Revision 2, procedure PPM 1.3.10, "Plant Fire Protection Program Implementation," and procedure PPM 1.3.10C, "Control of Transient Combustibles," Revision 1. Sargent & Lundy had performed a fire hazards analysis for the licensee and issued Report No. SL-5573, "Fire Hazard Analysis," Revision 2. Several fire scenarios had been evaluated in the fire hazards analysis that had the potential to impact a loaded cask. Scenarios included the ISFSI storage pad, the travel route between the plant and the ISFSI pad, the yard area and the train bay located in the reactor building. All scenarios resulted in fire durations and intensity with minimal damage to the storage cask and transfer cask physical integrity. Analysis was also performed for materials having explosive potential at the ISFSI pad, along the ISFSI route and within the buildings. These analysis demonstrated that there was minimal risk associated with the potential for an explosive blast scenario. Both fire scenarios and explosions that could occur at the site were bounded by the design analysis included in the FSAR.

An example of a fire scenario analyzed for the ISFSI was a fire associated with the cask transporter (crawler) or the tow tractor. The licensee planned to use an aircraft tow tractor to move the loaded cask from the train bay to a location just outside the train bay door where the cask would be transferred to the crawler. The tow tractor had a 50-gallon diesel fuel capacity by design to comply with the limit specified in Technical Specification 3.4.5 of Appendix B of the Certificate of Compliance. The crawler had a fuel tank capacity limited by design to hold no more than 49 gallons of diesel fuel. The fire analysis considered a cask engulfed by an open pool fire with complete burning of 50 gallons of diesel fuel from the crawler. In addition, the crawler's hydraulic system contained approximately 300 gallons of hydraulic fluid. The crawler's hydraulic system was designed to use a conventional petroleum based hydraulic fluid according to manufacturer's specification. It was classified as an OSHA/NFPA Class IIIB combustible liquid with the flashpoint of 399° F. Holtec performed an evaluation of a 400-gallon combustible fluid fire in Document No. HI-992284, Revision 1. The results indicate that the fire duration would be less than 29 minutes. Temperatures exceeding 572°F would be limited to less than 2 inches into the concrete shielding of the storage cask with a canister temperature rise of approximately 10°F. The supplemental analysis showed the cask design could withstand the combined effects of a crawler diesel fire and hydraulic fluid fire.

Vehicles other than the crawler and the tug that were used to support cask transport operations were conventional vehicles that typically do not possess any special fire safety design features. The licensee had established administrative controls for vehicles that were required to support cask transport operations. Procedural requirements limited each vehicle operated within 25 feet of a loaded cask to be positively controlled such that the operator must be in continuous attendance and the fuel quantity limited to 50 gallons.

Department of Energy (DOE) railroad tracks were located near the route that was used to move the cask from the plant to the ISFSI pad. Train traffic during cask movement from the plant to the ISFSI could be a hazard to workers and could provide an opportunity for a vehicle with greater than 50 gallons of diesel to transverse near the loaded cask. The licensee had made provisions to disable the rail line whenever a cask is being moved to the ISFSI by having the main rail line locked out at the north and south switches and train derailers placed on the tracks. The railroad office will be notified 5 working days prior to movement of the cask and again 24 hours prior to movement. The requirement to ensure the rail line was locked-out and the railroad office notified was specified in procedure 6.6.4, "HI-STORM System Site Transportation," Revision 3, Step 7.3.4 and Attachment 9.6 "Lock-Out/Restoration of the Railroad Line."

A tour of the entire ISFSI route from the train bay to the ISFSI pad was conducted. The principle fire protection engineer accompanied the NRC Inspector during the tour. The ISFSI route, storage pad and the train bay were observed to have low combustible loading. The area around the ISFSI pad was dirt and gravel with no vegetation to support a fire hazard. The licensee plans to implement monthly inspections of the ISFSI pad to ensure control of combustible material at or near the pad. Within the auxiliary building, combustible and transient material control requirements already existed and were appropriate.

The licensee had made arrangements with the Hanford Fire Department for offsite fire support to provide response and support during a fire at the ISFSI. The site emergency plan also included provisions for classifying emergencies involving fires and explosions.

Interviews were conducted with personnel during the dry run regarding fire protection. All individuals questioned, whether from the fuel handling crew, health physics or security were aware that only vehicles or equipment on the authorized list were permitted in the ISFSI protected area and that this was required to limit the amount of flammable liquids that could be involved in a fire.

12.3 Conclusion

The licensee had developed a program for controlling flammable liquids at the ISFSI to comply with technical specification requirements and to keep flammable liquids below the level analyzed in the FSAR for the worst case fire scenario. The Fire Hazards Analysis evaluated numerous fire scenarios to confirm that site specific fire scenarios were bounded by the design analysis in the FSAR. Administrative controls were established to limit the amount of flammable liquids that could be near a loaded cask and to lock-out rail traffic on the nearby rail line during cask movement. Arrangements had been made for support from an offsite fire department for fires at the ISFSI.

13 TRAINING PROGRAMS (60854)

13.1 Inspection Scope

The licensee is required to develop and implement a training and certification program for personnel who will operate equipment or controls that are identified as important to safety. The licensee's training program was reviewed to verify that the required elements described in 10 CFR 72 Subpart I and in Section 12.2.1 of the Final Safety Analysis Report were incorporated into the ISFSI training program.

13.2 Observations and Findings

The licensee was required to develop a training and certification program that meets the requirements of 10 CFR 72 Subpart I. This training program was required to address provisions for operator training and certification as well as physical condition and health of operations personnel. Personnel who operated equipment or controls that were identified as important to safety were required to be trained and certified under this program.

In addition to the requirements in Subpart I, the licensee was required to develop training modules that covered the topical areas described in Section 12.2.1 of the Final Safety Analysis Report. Topical areas identified in Section 12.2.1 included cask design, ISFSI facility design, certificate of compliance requirements, regulatory requirements, operating experience reviews and an overview of the Final Safety Analysis Report and NRC Safety Evaluation Report.

The training program for the ISFSI project was reviewed and interviews conducted with workers to determine the effectiveness of the training. The training program was being implemented under Quality Assurance Procedure QAP-7, "Personnel Indoctrination, Training, Qualification and Certification", Revision 11 and training policy, TRG-TQS-01 "Training Administration," Revision 6. The ISFSI training program was consistent with the site-wide training program in areas such as passing scores for tests, approval process and instructor qualifications. In accordance with the site-wide policy, ISFSI exams and lesson plans required a passing score of 80 percent.

The licensee had developed a total of 13 training modules. These modules were organized primarily by topic and included: dry cask overview, receipt inspections, transporter (crawler), tug, preparing the canister, canister closure, operating the alternate cooling water systems, vacuum drying system, helium backfill system, canister transfer, unloading a canister, helium cooldown system, and aerial lift. The elements listed in Section 12.2.1 of the Final Safety Analysis Report were incorporated into the "dry cask overview" module entitled "Columbia Generating Station Dry Cask Storage Project HI-STORM/HI-TRAC Overview, Spent Nuclear Fuel Dry Storage and Transport System," with operating experience reviews incorporated into each pertinent module. The modules incorporated expected dose rates and radiological conditions for the various work activities.

Each module underwent a technical review, instructional review, and line management review as part of the approval process. All training modules were approved by the training department supervisor.

Training also included on-the-job training (OJT) and on-the-job evaluations (OJE). A total of 25 areas were identified for OJT and OJE. Not all personnel had passed the OJE training on the first attempt. In those cases, remedial OJT and/or classroom training was provided and the workers had to re-test and pass.

To meet the certification requirements in 10 CFR 72.190, the licensee had established the following 14 qualification groups that required demonstration of qualifications.

- Dry Cask Overview
- HI-STORM & Canister Receipt Inspection
- Operate Spent Fuel Cask Transporter
- Operate Tug and Cart
- Prep Canister for Fuel Load
- Canister Closure
- Operate Alternate Cooling Water System
- Operate the Vacuum Drying System
- Operate the Helium Backfill System
- Canister Transfer
- Cask Unloading
- Operate the Helium Cooling System
- Reactor Building Equipment Access Area
- ISFSI Supervision

Individuals were considered certified when they had completed all in-class training, all OJT/OJE training and had completed the requirements in the qualification groups listed above that applied to their work assignments. Selected training records were reviewed for the classroom training, the OJT/OJE training and the completion of the qualification group training. Training for all personnel assigned to the ISFSI had been completed. A total of 18 personnel were trained, consisting of 12 craft workers, 2 radiation protection technicians, and 4 members of management.

To confirm that the licensee had adequately implemented an effective training program, four ISFSI technicians were interviewed to determine their familiarity with ISFSI operations. The technicians were able to describe the main components of the Holtec cask system and associated equipment. They understood and were familiar with radiological conditions of ISFSI operations as well as other hazards that could be encountered during cask loading and moving operations such as falls, potential high temperatures of the cask, and heavy load concerns. The technicians knew which steps required quality assurance sign-off and what types of data that must be recorded. All technicians interviewed felt comfortable reporting safety concerns and felt empowered to stop work if conditions were not safe. Technicians felt that the ISFSI procedures were adequate and that suggestions for improvements were welcome.

ISFSI training information had been incorporated into the licensee's computer tracking system "Portal/J". This program allowed supervisors to check on the qualification status of personnel under their supervision. By utilizing this system, supervisors could confirm the training and qualification status of individuals before assigning them work tasks.

A retraining schedule had not been developed by the licensee. The licensee recognized the need for the eventual retraining/requalification of ISFSI personnel and had created an action item in the plant tracking log. Several issues were being considered by the ISFSI Training Advisory Group related to the retraining program including continuing training (refresher), recommended interval or frequency of training (ie., annual, 18 month, or prior to cask loading campaign), and determining the need for additional employees to be trained.

The licensee ISFSI certification program was also required to consider the general health and physical condition of personnel assigned to operate equipment and controls that were considered important to safety. These individuals must be certified as physically capable of performing the required functions as specified in 10 CFR 72.194. The licensee had developed Occupational Health Instructions OHI-14 "ISFSI Technician Examination," dated December 19, 2001. Instruction OHI-14 provided guidance for determining the medical suitability for ISFSI technicians. This document included a description of the frequency of examination, the examination content and potentially disqualifying conditions.

The licensee identified one ISFSI technician that failed the peripheral vision test. This individual was not disqualified from being an ISFSI technician, as allowed for in 10 CFR 72.194, but was restricted from working on certain equipment important to safety. The inspector interviewed the individual and found the individual to be aware of tasks that he would not be allowed to perform. The individual stated that he felt comfortable informing his supervisor when he encountered tasks he felt he should not perform. The inspector interviewed the craft supervisor, who was also aware of the restrictions on the employee concerning performing tasks important to safety.

The licensee implemented several provisions to strengthen their training program for the staff assigned to the ISFSI. This included hiring personnel from another utility that had recently loaded Holtec casks, as subject matter experts. These individuals developed the training program and provided the instruction. By using these individuals, first hand knowledge was available to the workers concerning the Holtec cask and the process for safely loading a cask and placing it into the ISFSI. This added to the reassurance of the workers by allowing them to interface with personnel that were experienced in handling the casks. The licensee had also incorporated extensive information concerning problems encountered at other sites during loading of casks into the training. The NRC inspection team felt the training program was one of the strengths of the Columbia Generating Station program due to the use of the experienced subject matter experts in developing and implementing the ISFSI training program.

13.3 Conclusion

The licensee had developed a training and certification program for operator personnel performing work at the ISFSI on equipment and controls that were identified as important to safety. The program incorporated the requirements in 10 CFR 72 Subpart I and Section 12.2.1 of the FSAR and included formal classroom training, on-the-job training and specific task demonstrations.

Training was completed for all operator personnel assigned ISFSI duties. Interviews with selected personnel verified that training had been adequately implemented.

14 QUALITY ASSURANCE PROGRAMS (60854)

14.1 Inspection Scope

The licensee had applied their NRC approved 10 CFR Part 50, Appendix B quality assurance (QA) program to the dry cask storage project. This inspection reviewed the implementation of the Part 50 QA program to the various aspects of the dry cask storage project. Selected audits and surveillances were reviewed to verify that an active QA oversight program was being implemented for the ISFSI related activities.

14.2 Observations and Findings

The licensee's 10 CFR Part 50 QA program, as described in the Operational Quality Assurance Program Description (OQAPD) document, was developed in accordance with the requirements of 10 CFR 50 Appendix B and had been approved by the NRC. The current version of the OQAPD was Revision 36. On August 11, 2000, Energy Northwest had notified the NRC of their intent to use their Part 50 QA program for the activities at the ISFSI, as allowed for in 10 CFR 72.140(d). The licensee identified quality assurance controls in their OQAPD to be applied to items classified as important to safety in the Final Safety Analysis Report for the HI-STORM 100 cask design. This included provisions for a graded approach to quality consistent with the importance to safety of a structure, system, component or ISFSI activity. Specifically, Revision 34 of the OQAPD had incorporated the ISFSI into the applicable sections of the document, classified which structures, systems, and components related to the ISFSI were "important to safety," identified the level of QA controls to be applied, and revised the QA program to reflect the record keeping requirements of 10 CFR 72.174.

To evaluate the implementation of the QA program, the results of several recent audits, continuous monitoring processes, third party assessments of vendors and fabricators, and self-assessments associated with the design, construction and pre-operational phase activities of the ISFSI were reviewed. The audits had been performed in accordance with the following procedures: SWP-ASU-01, "Audits, Surveillances, and Assessments," Revision 10, QSI-5, "Audit Preparation," "Entrance Meetings and Exit Meetings," Revision 1, QSI-6, "Activity Reports," Revision 1, and QSI-7, "Quality Reports," Revision 2. Independent oversight was performed in accordance with procedure QSI-4, "Continuous Monitoring Activities," Revision 2. A representative sampling of deficiency documents related to the ISFSI were selected and evaluated to verify the adequacy of the licensee's program for addressing conditions adverse to quality in accordance with procedure SWP-CAP-01, "Problem Evaluation Requests (PERs)," Revision 4. Also evaluated and found acceptable were procedural controls related to the ISFSI in the areas of materials management, control of measuring and test equipment, inspection and testing, and operating status controls.

Based on the results of these reviews, it was determined that the licensee was performing effective QA oversight of activities. The licensee's program for identifying and documenting deficiencies in the PERS system was adequately implemented and the identified conditions were being appropriately addressed. As a result of the review of selected procedures pertaining to the ISFSI, it was determined that appropriate administrative controls had been established and items classified as important to safety were being properly controlled.

During the inspection of the quality assurance program, several procedures were reviewed and found to adequately incorporate QA controls into the procedures. These included:

- Procedure SWP-MMP-02,Warehousing, Revision 2
- Procedure 1.5.4, Control of Measuring and Test Equipment/Transfer Standards/Portable Tools, Revision 25
- Procedure 6.6.3, Hi-Storm Overpack Final Assembly Receipt Inspection, Revision 2
- Procedure SWP-MAI-01, Work Management-Planning, Scheduling and Work Activities, Revision 10

14.3 Conclusion

The licensee conducted quality assurance oversight of ISFSI activities using their NRC approved 10 CFR Part 50 quality assurance program. A review of documents, procedures and audits performed by the quality assurance organization determined that the licensee had appropriately applied their Part 50 quality assurance program to the activities associated with the ISFSI.

15 SECURITY (60854, 81001)

15.1 Inspection Scope

[redacted]

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REDACTED SECTION INCLUDES OFFICIAL USE ONLY - SECURITY RELATED INFORMATION

[redacted]

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[redacted] [redacted]

[redacted]

[redacted].

16 RECORDS/DOCUMENTATION (60854)

16.1 Inspection Scope

The licensee was required to establish and maintain certain records related to the dry cask storage project including specific information related to each cask loaded. The licensee's program for collecting and maintaining these records was reviewed.

16.2 Observations and Findings

The licensee was required by 10 CFR 72.212(b)(1)(i) to notify the NRC at least 90 days prior to the first storage of spent fuel at the ISFSI. Energy Northwest issued a letter on January 11, 2002, informing the NRC of the intent to store spent fuel at their ISFSI. With the first cask loading completed on September 20, 2002, this letter meets the requirement for the 90 day prior notification.

The licensee was required to maintain a copy of the certificate of compliance and documents referenced in the certificate of compliance, FSAR and technical specifications. This requirement was specified in 10 CFR 72.212 (b)(7). The licensee had a current copy of the Certificate of Compliance and Technical Specifications, Amendment 1 dated July 15, 2002, and the HI-STORM FSAR Report HI-2002444, Revision 1. Other pertinent documents were maintained in the corporate library. Selected NUREGS, ASME codes, and ANSI standards referenced in the FSAR and technical specifications were easily located by the inspector in the corporate library.

The licensee was required, pursuant to 10 CFR 72.212 (b)(2)(i)(C), to retain a copy of the written evaluations performed under 10 CFR 72.212 (b)(2) until spent fuel was no longer stored at the site. The licensee had established a list of required documents for retention in a requirements tracking list and in procedure SWP-REC-01 "Records Management," Revision 4. The requirement for retention of the §72.212 (b)(2) documentation had been identified in the requirement tracking list and was scheduled for inclusion in the next revision of procedure SWP-REC-01.

The licensee was required to maintain certain records in accordance with 10 CFR 72.212 (b)(8)(i) thru (iii) for each cask loaded. The licensee had issued procedure SFS-05 "MPC Documentation Tracking Requirements," Revision 0, which established the required documentation to be collected for each cask and specified that this documentation was to be forwarded to the permanent plant files for storage until the cask was shipped offsite. Documentation required by procedure SFS-05 for each cask included the fabricator's certificate of compliance for the cask, which included the name and address of the cask vendor, a copy of completed procedure PPM 9.6.1 "Spent Fuel Selection for Cask Loading," which included the cask loading plan and the cask loading map, and a copy of completed procedure PPM 6.6.15 "Spent Fuel Cask Loading Verification," which included records of the post-loading verification of the fuel assembly locations to confirm that the serial numbers match the approved cask loading map.

Maintenance records were also required by §72.212(b)(8) to be maintained for each loaded cask. Procedure SFS-05 required retention of all work orders relating to the cask. The licensee tracked each cask with a specific equipment number. Work orders related to each cask specific equipment number could be tracked and retrieved using the licensee's software package referred to as "Passport 6". Passport 6 was operated under the licensee's NRC approved QA program.

Once a cask was loaded with spent fuel and placed in the ISFSI, the licensee was required within 30 days to notify the NRC that the cask was now in use. Notification was required by 10 CFR 72.21 (b)(ii). This notification, referred to as the "Registration for

Use" letter by the licensee was issued by the Regulatory Affairs department and was included in the list of required documents that must be collected within 30 days as specified in the checklist in procedure SFS-05.

The licensee was required by 10 CFR 72.72(d) to maintain a duplicate set of records related to their special nuclear material at a separate location. The licensee had submitted a request to the NRC on December 12, 2000, to maintain only a single set of records using the provisions in ANSI N45.2-1974. This ANSI standard provided requirements for protection of records against degradation mechanisms such as fire, humidity, and condensation. The requirements in the ANSI standard have been endorsed by the NRC in Regulatory Guide 1.88 "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records," as adequate for satisfying the record keeping requirements of 10 CFR Part 50, Appendix B. The ANSI standard also satisfied the record keeping requirement in 10 CFR 72.72 by providing for adequate maintenance of records regarding the identity and history of the spent fuel in storage. On March 15 2001, the NRC responded to the exemption request by the licensee and approved storage of only one set of records.

16.3 <u>Conclusion</u>

Adequate provisions were established by the licensee to ensure that required documents and records specified in 10 CFR 72.212 would be retained for the casks loaded at the ISFSI and that required notifications would be completed.

17 Loading of the First Cask (60855)

17.1 <u>Inspection Scope</u>

On September 13, 2002, Columbia Generating Station began the loading of their first canister. The NRC provided full time coverage of the activities until the canister was downloaded into the storage cask in the train bay. Observation of activities was performed to verify compliance with the certificate of compliance and technical specifications and to ensure work activities were performed safely and in accordance with radiation protection requirements.

17.2 <u>Observations and Findings</u>

On September 13, 2002, Columbia Generating Station was informed by the NRC that all pre-operational inspection activities of the dry cask storage project had been completed. At 1:00 p.m., the licensee initiated the movement of the first spent fuel assembly from the storage racks and by 1:22 p.m., had placed the spent fuel assembly into the canister. By 8:50 p.m., the 68th spent fuel assembly had been placed in the canister and loading was complete.

Independent verification of the spent fuel assemblies was performed to confirm that the assemblies selected for storage in the first canister, as documented on the approved cask loading map, were the ones actually placed in the canister. Procedure 6.6.15, "Spent Fuel Cask loading Verification," Revision 1, was used for the verification process. This procedure also required verification that the assemblies were fully lowered into the canister to ensure the top of the assembly would not interfere with the seating of the lid.

The lid was placed on the canister and the canister removed from the spent fuel pool on September 14, 2002 at 2:25 a.m. The licensee determined the time-to-boil limit for the canister based on a spent fuel pool water temperature of 95 °F. Using Attachment 9.7, "Maximum Allowable Time Duration for Wet Transfer Operations," Revision 1, of Procedure 6.6.6, the licensee determined that the time-to-boil limit was 86 hours. This is the time limit that water can remain in the canister with no provisions for cooling. The cask was welded and drained of water on September 16, 2002, at 2:15 a.m. This was a total of 48 hours.

As the transfer cask was removed from the spent fuel pool, the crane lifting yoke, cabling, and transfer cask were washed down with demineralized water to reduce the potential for the spread of contamination and hot particles. As the loaded transfer cask was removed from the spent fuel pool, the crane's digital load cell measured 118.1 tons. This represented the weight of the spent fuel assemblies, transfer cask, canister, canister lid, canister filled with water, and the lifting yoke.

The transfer cask was placed in an area near the spent fuel pool and an initial decontamination of the transfer cask was performed. The cask was then moved to the wash down pit on the 606' elevation and decontamination continued. Radiation protection personnel decontaminated the transfer cask sufficiently that protective clothing was no longer required for personnel working around the canister.

The radiation protection staff monitored craft personnel doses and area dose rates via wireless remote monitoring. The remote monitoring system consisted of a series of cameras, video monitors, receivers and transmitters that provided the radiation protection staff with visual, audio and real time dose monitoring capabilities. The radiation protection staff transmitted information to video monitors throughout the plant to allow management and plant personnel the ability to remotely observe fuel loading operations without being on the 606' elevation.

During the work activities inside the reactor building, the licensee implemented effective ALARA practices that were observed by the NRC inspectors. This included:

- establishing boundaries for potentially contaminated zones
- designating areas as restricted areas and low dose areas
- establishing video monitors for remote observation of activities
- assigning dose estimations by activity and personnel group
- using computerized RWPs which did not allow personnel to sign in without completing a pre-job ALARA briefing
- providing continuous health physics coverage for all jobs
- providing real time area and personnel dose rate monitoring
- conducting frequent dose rate surveys to evaluate any changes in radiological conditions

- maintaining excellent decontamination practices to mitigate the potential for spread of loose contamination
- implementing neutron dosimetry and survey requirements
- completing procedural requirements related to health physics as specified in the procedures developed specific to the ISFSI operations

The health physics staff was very knowledgeable of health physics practices, federal regulations and station practices for implementing the health physics program and had a good understanding of radiological concerns related to ISFSI operations. Pre-job ALARA briefings provided good instructions to workers, described the potential radiological hazards and conditions that may occur, identified the required protective clothing for specific jobs, and identified specific activities that required craft personnel to obtain clearance from the health physics personnel before the job could continue.

The licensee had placed special emphasis on controlling the spread of contamination and was committed to keeping work areas as free of contamination as possible. After the transfer cask was removed from the spent fuel pool, considerable time and effort was directed toward removing smearable contamination from the transfer cask. Final contamination survey results of the cask were typically below detectable limits. Results of the surveys had been recorded in Procedure HSP-SFS-C102, "Transfer Cask Surface Contamination."

The licensee drained approximately 60 gallons of water from the cask prior to welding to ensure that the metal lid would not be in contact with the water and thereby provide a heat sink during welding. After the 60 gallons were removed, dose rate measurements were taken. The contact dose rate on the lid was approximately 10 mrem/hr. The dose rate above the gap between the canister and the transfer cask was 45 mrem/hr. Dose rates on the side of the transfer cask were 1 mrem/hr. Neutron dose rates were less than 1 mrem/hr.

The welding of the canister was performed using an automatic welding system. This system is described in Section 9 of this report. Tack welding of the lid started on September 14, 2002, at 9:40 p.m. and was completed by 11:00 p.m. Tack welding was used to keep the lid properly positioned during the full welding. Monitoring for hydrogen gas was performed prior to and during the welding. Section 18 of this report discusses the hydrogen monitoring effort implemented by the licensee. Hydrogen levels as high as 80 percent of the lower explosive limit were detected prior to welding. At 12:40 a.m. on September 15, 2002, root pass welding was initiated. The root pass weld required more than one pass for one side of the lid because the gap between the canister wall and the lid was too large for one pass to completely fill in. Welding of the lid took 16 ½ hours. Visual examinations and nondestructive testing of the welds using dye penetrant found only minor indications in the welds that required repair. The helium leak tests on the root pass weld and the final weld confirmed the adequacy of the welds on the lid. The helium calibration standard used for the weld examinations on the lid had a certified leak rate of 6.6×10^{-8} atmospheres-cubic centimeters/second (atm cc/sec). The helium standard had been calibrated on September 5, 2002. The helium calibration standard used for the vent covers had a leak rate of 3.0×10^{-6} atm cc/sec and had also been calibrated on September 5, 2002.

Hydrostatic testing of the canister was completed in 50 minutes. The hydrostatic test pressure was 127 lbs/in^2 (psig). No water leakage was observed from the lid welds. Purging of the water from the canister required 2 hours and 20 minutes. Vacuum drying took 48 hours to achieve the 3 torr technical specification limit. The actual limit achieved was less than 2.5 torr for 30 minutes. The canister was backfilled with helium and the vent and drain ports and closure ring were welded in place.

Temperature readings were taken on several occasions while the transfer cask was in the wash down pit. Readings were taken on the lid. When the water was in the cask, the temperature on the lid was approximately 122°F. The original spent fuel pool water temperature had been 95°F. During vacuum drying, the temperature was 114°F soon after the process started and reached 98°F when the vacuum drying was completed. Five hours after the helium was in the canister, the temperature reached 145°F.

The licensee performed numerous radiation surveys of the transfer cask. Dose rates changed when the water was drained from the canister. Dose rates on the lid increased from 3-4 mrem/hr to 10 mrem/hr. Above the gap between the canister and the transfer cask, dose rates increased from 10 mrem/hr up to 60 mrem/hr. Gamma dose rates on the side of the transfer cask went from less than 1 mrem/hr up to 4 mrem/hr. Neutron dose rates did not change significantly and stayed about 1 mrem/hr.

The licensee was required by Technical Specification 3.2.1 of the Certificate of Compliance, Appendix A, to perform a radiological survey of the transfer cask to ensure radiation levels were below 220 mrem/hr (gamma plus neutron) on the side and 60 mrem/hr (gamma plus neutron) on the top. These limits were average values from several readings and would demonstrate that the cask had not been loaded with the incorrect fuel. The licensee performed this survey on the transfer cask after the water was removed and prior to lowering the transfer cask to the train bay. The highest readings obtained on the side of the transfer cask were 4 mrem/hr gamma and 1 mrem/hr neutron. The average of the dose rates taken on the side, both gamma and neutron, was calculated to be 4.25 mrem. The average of the dose rates taken on the top was calculated to be 2 mrem/hr.

In preparation for downloading the canister into the storage cask, the health physics staff established boundaries to prevent personnel from accessing areas where the dose rates were expected to be significantly high. Health physics personnel positioned wireless remote monitors on the hatchway railing located on the 471' level and at locations surrounding the transfer mating device. After the canister was transferred into the storage cask, neutron and gamma dose rates were measured. The highest reading obtained was 2400 mrem/hr on contact at the gap between the storage cask and the canister prior to installation of the storage cask lid.

The loaded storage cask was successfully moved from the reactor building and placed on the ISFSI pad on September 20, 2002. Radiation surveys were performed to verify radiation levels were in compliance with Technical Specification 3.2.3 of the Certificate of Compliance, Appendix A. The limits in the technical specification were 50 mrem/hr on the sides, 10 mrem/hr on the top and 45 mrem/hr at the inlet and outlet vents. The surveys were conducted in accordance with Procedure HSP-SFS-C103, "Overpack

Average Surface Dose Rates," Revision 1. Radiation levels on the sides, top and vents were less than 1 mrem/hr respectively.

A review of records and an analysis performed by the Columbia Generating Station ALARA planner determined that the accrued dose for all activities associated with the loading of the first cask was 0.358 manrem. This was well below the original estimated dose of 0.640 manrem.

17.3 Conclusion

The licensee successfully completed the loading of their first cask and placement of the cask on the ISFSI pad on September 20, 2002. Dry cask storage activities were conducted safely and in compliance with procedures. Radiological controls were effectively implemented. The overall dose to complete the project was well below original estimates.

18 Special Topic: Hydrogen Generation by the Holtec Cask (60855)

18.1 Inspection Scope

The FSAR for the Holtec cask had not identified any chemical or galvanic reactions that would generate hydrogen while the canister was filled with water. However, during the observation of pre-operational test activities involving a canister in the spent fuel pool, bubbles were observed being generated by the canister. These bubbles were later analyzed and found to contain hydrogen. The licensee's program for dealing with the hydrogen during welding of the canister lid was evaluated.

18.2 Observations and Findings

During the pre-operational test demonstrations conducted on August 15, 2002, a canister was placed in the spent fuel pool to demonstrate loading a dummy fuel assembly into the canister. The canister had been placed in the pool the previous day after sitting on the 606' elevation for almost a week filled with condensate water. During the demonstration, the NRC inspector and the representative from the State of Washington Department of Health observed bubbles being generated by the canister. The licensee was questioned about the bubbles and had assumed the bubbles to be trapped air from inside the canister. The licensee was requested to obtain a sample of the bubbles and analyze the sample for hydrogen. The sampling and analysis was completed that same day. The bubbles collected were found to contain 63 percent hydrogen, 8 percent oxygen, and 29 percent nitrogen. A second sample was collected the following day and found to contain 75 percent hydrogen. The licensee estimated that the rate of generation of the bubbles was approximately 4 liters/hr.

FSAR, Revision 0, was in effect at the time of the observation of the bubbles. The FSAR was reviewed for information concerning the potential for hydrogen generation by the Holtec canister. Section 3.4.1 of the FSAR stated, "In this section, it is shown that there is no credible mechanism for chemical or galvanic reactions in the HI-STORM 100 system." Section 3.4.1 also stated, "In accordance with NRC Bulletin 96-04, a review of the potential for chemical, galvanic, or other reactions among the materials of the HI-STORM 100 system, its contents and the operating environments which may produce adverse reactions has been performed. Table 3.4.2 provided a listing of the materials of fabrication for the HI-STORM 100 system and evaluated the performance of the material in the expected operating environments during short term loading/unloading operations and long term storage operations. As a result of this review, no operations were identified which could produce adverse reactions beyond those conditions already analyzed in this FSAR." Based on these conclusions by Holtec, no monitoring for explosive or combustible gases was required or recommended by Holtec during welding operations. As a result of Holtec's analysis, Columbia Generating Station had not included any provisions in their procedures to analyze for explosive gases prior to welding operations or to vent during welding.

The statements in the FSAR were in response to issues identified in NRC Bulletin 96-04, "Chemical, Galvanic, or other Reactions in Spent Fuel Storage and Transportation Casks," issued July 5, 1996. This bulletin was issued because hydrogen gas had been produced in a canister at another reactor site resulting in ignition of the gas during welding of the lid. The canister that experienced the hydrogen ignition was a different design than the Holtec design. Loading of a canister that produced hydrogen was discussed in Bulletin 96-04 and determined acceptable by the NRC as long as controls were established to minimize hazardous conditions that may be created by any reactions that could generate hydrogen. For canisters known to generate hydrogen, this could be accomplished by monitoring for hydrogen, venting the cask in a way to preclude ignition of the hydrogen during welding, and having provisions for stopping the welding process if the hydrogen levels reached explosive limits. Licensees were also required to address hydrogen problems related to unloading a canister.

The Holtec FSAR did not address hydrogen generation during loading or unloading and did not require licensee procedures to provide for hydrogen monitoring or venting during welding. The Holtec FSAR became effective May 31, 2000, after Bulletin 96-04 was issued. Holtec addressed the issues in Bulletin 96-04 in Section 3.4.1 of the FSAR by taking the position that no flammable or explosive gases were generated by their canisters.

Holtec was a supplier of boral fuel racks for spent fuel pools and has experience with boral since 1987. As such, Holtec was aware that boral generated hydrogen when placed in the spent fuel pool until the aluminum in the boral had undergone sufficient reaction with the water to form an aluminum oxide coating on the boral. As the aluminum oxide was formed, hydrogen was given off in the reaction. The process of creating the aluminum oxide coating was called passivation. Section 3.4.1 of the FSAR discussed passivation of the boral used in the Holtec design and stated that all aluminum surfaces would be pre-passivated or anodized to eliminate the incidence of aluminum water reaction inside the canister. FSAR Table 3.4.2, which listed the compatibility of the HI-STORM 100 components with the fuel pool, also stated, "The boral will be passivated before installation in the fuel basket." The table stated that extensive in-pool experience on spent fuel pool racks has shown no adverse reactions."

The licensee reviewed the records for the canister in the spent fuel pool and communicated with Holtec to verify that passivation had been completed for the canister. Records were located that documented that the canister had been pre-passivated for the minimum 72 hours that Holtec had established as the required time for the passivation process.

Additional review of the FSAR found several other sections that discussed boral but did not indicate that hydrogen would be generated. The FSAR sections reviewed included: Sections 1.2.1.3.1, 3.4.1 and 4.4.1; Tables 1.0.1 and 3.4.2; and Figures 4.4.6 and 4.4.7. All information reviewed indicated that the generation of explosive gases would not occur. In fact, the NRC had accepted the pre-passivation process used by Holtec as adequate in Federal Register Notice 65FR11458, dated March 3, 2000, which addressed a comment submitted to the NRC concerning the adequacy of the pre-passivation process. Based on Holtec's FSAR and pre-passivation analysis, the NRC had accepted Holtec's 72-hour pre-passivation of the boral as an adequate process and stated that during the fabrication process, the absence of any gas bubbles emanating from the water after 72 hours was an adequate indication that the aluminum surfaces had been covered with aluminum oxide.

The NRC approved the use of the Holtec HI-STORM 100 cask system and added the cask to the list of approved casks in 10 CFR Part 72 with the issuance of Federal Register Notice 65FR25241 on May 1, 2000. The NRC issued a safety evaluation report which documented the NRC's review of the Holtec information submitted for cask approval. A discussion of the potential for a chemical and galvanic reaction was provided in Section 3.3.3 of the safety evaluation report which stated that the NRC staff agreed with Holtec's conclusion that the HI-STORM 100 cask system is constructed of materials which will not produce a significant chemical or galvanic reaction and the attendant corrosion or combustible gas generation.

Further review of the hydrogen issue involved examining other licensee and Holtec documents supporting the conclusion that hydrogen gas generation in the canister would not occur. A Holtec document entitled DS-248, "Chemical Stability of the Holtec MPC Internals During Fuel Loading and Dry Storage," dated August 13, 2001, was reviewed. This document provided an analysis of the chemical stability of the Holtec canister internals during fuel load. DS-248 stated on page 1 that "The HI-STAR and HI-STORM license applications were the first submittals that came under Spent Fuel Project Office review subsequent to the Point Beach hydrogen ignition event. They accordingly received a close regulatory scrutiny with respect to the potential of operational surprises. The information contained in the HI-STORM and HI-STAR FSARs provides a summary of the material presented to the NRC to demonstrate the absence of an actuating mechanism for the adverse chemical/galvanic reaction in a Holtec MPC." The conclusion in DS-248 stated, "The technical information presented in the foregoing leads to the firm conclusion that there is no mechanistic means for the generation of hydrogen or any other combustible gas in meaningful quantities in a Holtec MPC during fuel loading evolutions. Therefore, the occurrence of a disruptive chemical event such as hydrogen ignition in a Holtec MPC can be ruled out categorically." The conclusion goes on to discuss that 10 MPCs had been loaded at that time without any problem, thus the field experience confirmed the conclusion that there was not a hydrogen problem.

The NRC had inspected the use of boral in the Holtec canister. This inspection focused on compliance with the design and fabrication requirements related to criticality control and observed that passivation was being performed by the fabricator as required by the FSAR. The minimum boral concentration required in the neutron absorber material inside the canister was specified in the Certificate of Compliance, Appendix B, Section 3.2. Confirmation of compliance with these limits had been performed as part of an NRC inspection at US Tool and Die on February 4-8, 2002, and documented in NRC Inspection Report 72-1014/02-201 dated February 28, 2002.

In response to the discovery of the hydrogen bubbles at the Columbia Generating Station, Holtec issued Information Bulletin #8 on August 16, 2002, to all of their cask users. The bulletin described the observation of the gas bubbles and stated that the bubbles were the result of hydrogen off-gassing from the boral. The bulletin also stated, "Generation of hydrogen in BWR fuel pools sufficient to be visible in the form of rising bubbles from fuel racks when they are installed was a well known phenomenon to Holtec's site engineers who have installed hundreds of boral equipped fuel racks."

The Holtec bulletin also stated that over 20 canisters have been welded without actions taken to purge the space below the canister lid during welding, that ISFSI personnel have seen varying levels of subdued gas generation from water filled canisters in previous loadings, and that the Hatch plant has monitored for hydrogen since the first cask loading. Despite these statements, there was no documentation that the other Holtec cask users had been previously alerted to the potential for hydrogen problems and in fact, the staff at Columbia Generating Station and the Holtec representative onsite at the time were adamant when the bubbles were first observed that they were not hydrogen.

On September 6, 2002, Holtec issued Revision 1 of their FSAR. This revision incorporated the potential for hydrogen generation by the boral in the canister. Section 3.4.1 was revised to incorporate wording that limited quantities of hydrogen could be generated in the canister. Table 3.4.2 was revised to state that the boral was passivated before installation in the canister to minimize the amount of hydrogen generated. FSAR Sections 8.1.5 and 8.3.3 relating to lid welding and cutting were revised to caution the user that oxidation of the boral panels may create hydrogen gas while the canister is filled with water. The user was required to perform monitoring for combustible gases prior to and during welding or lid cutting. The change to the FSAR also recommended that the space below the lid be exhausted or purged with an inert gas prior to and during lid welding and cutting.

The changes to the FSAR were reviewed by Holtec against the criteria in 10 CFR 72.48 and documented in Holtec 72.48 Screening/Evaluation Form # 621. This evaluation concluded that the changes to the FSAR did not require approval by the NRC. A review of the basis for this conclusion and a determination of the adequacy of Holtec's 72.48 evaluation will be performed by the NRC. This issue is being considered an Unresolved Item (72-35/0201-05). An Unresolved Item is defined as an issue about which more review is needed in order to determine if the issue is acceptable or is a violation or deviation of NRC regulations.

In response to the hydrogen generation, the Columbia Generating Station implemented changes to their procedures for canister lid welding to include the requirement to monitor for hydrogen and if present, to purge under the canister lid during welding. Procedure 6.6.7, "MPC Processing," Revision 2, had incorporated requirements into Section 7.2.3 to monitor the weld area prior to welding using a combustible gas monitor. If combustible gas levels exceed 50 percent of the lower explosive limit (LEL), welding was to be stopped.

During the welding of the first canister loaded with spent fuel on September 14, 2002, the licensee performed hydrogen monitoring prior to welding. Monitoring for combustible gases was performed using an Industrial Scientific Model ATX612 multi-gas monitor. This monitor measured both oxygen levels and combustible gas levels. The user manual warned that inaccurate readings for combustible gas levels would occur if oxygen levels were not normal.

Prior to welding, 60 gallons of water had been drained from the canister. The first reading using the combustible gas monitor indicated 80 percent of the LEL around the vent opening on the canister lid. A purge of the air under the lid was started by connecting a hose to the vent port on the lid and pulling the air through a HEPA filter. The purge was continued for approximately an hour. The purge line was disconnected and a second measurement taken around the vent opening which registered 8 percent LEL. The purge was again continued. An hour later the purge was stopped and additional measurements taken. The initial measurement indicated 27 percent LEL. Within 15 minutes, the combustible gas monitor alarmed at 50 percent LEL. After discussions between the NRC and the licensee, the combustible gas monitor was returned to the vent opening 20 minutes later and registered 67 percent LEL. This level remained stable with readings ranging between 66 percent to 68 percent LEL over a 20-minute period. One anomaly did occur with a reading of 78 percent, but the reading only lasted for 3 seconds, then returned to 68 percent. This was suspected to be a bubble of gas. After the 20-minute monitoring period, the combustible gas monitor reading suddenly began to drop finally reaching 56 percent, at which time a gas monitor failure alarm occurred. High humidity was suspected as the problem. Excessive moisture accumulating on the internal dust filter/water stop was identified in the user manual as a failure mechanism for the monitor.

Continuous purging of the vent on the lid was established and a second hydrogen monitor was obtained. No hydrogen was detected around the weld area and tack welding of the lid was initiated. The combustible gas monitor was connected into the purge line to provide continuous monitoring of the air being removed from under the canister lid. Readings were 1 percent to 2 percent LEL on the purge line. Oxygen levels were 20.9 percent. A third combustible gas monitor was obtained and used to verify the readings on the purge line. During the tack welding process, the combustible gas monitor alarmed on a low oxygen level of 19.5 percent, then returned to 20.8 percent. Discussions with the welders confirmed that they had performed a tack weld using argon gas. The argon gas had apparently been pulled into the opening between the lid and the canister wall where the welding was being performed and then into the purge line causing the monitor to alarm.

Welding was successfully performed for the root pass weld with hydrogen levels on the purge line remaining around 1 percent to 2 percent. By the time the welding on the root pass had been completed over a 16 hour period, four combustible gas monitors had failed, again probably due to moisture.

18.3 Conclusion

The FSAR for the HI-STORM 100 cask system provided various statements that hydrogen would not be generated in the canister while in the spent fuel pool. However, during the pre-operational test demonstrations, bubbles were observed coming from the canister. The bubbles were analyzed and found to contain 63 percent to 75 percent hydrogen.

Holtec performed a safety evaluation of the hydrogen generation issue in accordance with 10 CFR 72.48. The evaluation concluded that the FSAR could be revised to incorporate provisions for hydrogen generation by the canister without approval by the NRC. The adequacy of Holtec's evaluation will be reviewed by the NRC and will be tracked as an Unresolved Item.

The licensee implemented a hydrogen mitigation process during the welding of the lid onto the canister that successfully monitored for and prevented any build-up of hydrogen under the lid which could be ignited during the welding.

19 Exit Meeting

The inspectors presented the inspection results to members of licensee management at exit meetings on July 19, 2002, and September 19, 2002. The licensee acknowledged the findings presented. The licensee did not identify as proprietary any information provided to, or reviewed by, the inspectors.

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

Energy Northwest

J. Allen, Radiation Support Supervisor
S. Beatty, Lead Health Physics Technician
I. Borland, Radiation Protection Manager
W. Braun, Mechanic
D. Butlek, ISFSI Health Physics Technician
A. Carlyle, Technical Specialist, Regulatory Services
D. Dinger, Health Physics Training Specialist
T. Erwin, Materials Processes and Qualification Supervisor
L. Ferek, Sr. Engineer, Fuel Cycle Management
D. Gertz, ISFSI Health Physics Technician
S. Grundhauser, Maintenance Training Supervisor
W. Harper, P.E., Fire Protection Engineering Supervisor
S. Hatfield, Technician
P. Henckel, Mechanic
B. Hodges, Corporate Librarian
D. Holmes, Emergency Planner
B. Hopkin, Leak Test Examiner
W Kropp, Acting Quality Services Supervisor of Operations and Engineering
B. LaFramboise, Project Engineering Supervisor
D. Larkin, ISFSI Project Manager, Project Support
L. Linik, Fuels Design Supervisor
R. Madden, Sr. Quality Program Auditor
C. McDonald, Supervisor Health Physics/Chem/General Employee Training
J. McDonald, Environmental Scientist
T. McNabb, ALARA Planner
S. Metzger, Lead ISFSI Health Physics Technician
D. Montgomery, Project Manager, Operations & Management Services
C. Moore, ISFSI Project Manager, Engineering
R. Nash, Sr. Quality Program Auditor
L. Noble, ISFSI Project Engineer, ISFSI
A. Owens, Mechanic
T. Powell, Licensing Specialist
A. Ramos, Craft Supervisor
G. Rowell, ISFSI Project Engineer, Plant Modifications
K. Saenz, Information Management Specialist
S. Scammon, ISFSI Program Manager
J. Sisk, Welding Engineer
J. Wells, ISFSI Planner
N. Zimmerman, ISFSI Project Manager, Operations

Holtec Contacts

B. Gilligan, Program Manager

Cogema Incorporated

J. Keve, Level III Helium Leak test Examiner
B. Hopkin, Level II Helium Leak test Examiner

Welding Services, Inc.

R. Acree, Quality Assurance/Quality Control Supervisor, NDE Level II Examiner
D. Antoine, Welding Group Lead

Bechtel Inc

D. Maley, Project Engineer

Ferg and Associates

L. Ruff, Procedure Writer/Training Contractor
J.. Rupp, Procedure Writer/Training Contractor
B. Zank, Procedure Writer/Training Contractor

INSPECTION PROCEDURES USED

60801	Spent Fuel Pool Safety
60851	Design Control of ISFSI Components
60854	Preoperational Testing for an ISFSI
60855	Operations of an ISFSI
60856	Review of 10 CFR 72.212(b) Evaluations
60857	Review of 10 CFR 72.48 Evaluations
81001	ISFSI Security
83750	Occupational Radiation Exposure

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

72-35/0201-01	NCV	Failure to complete a 50.59 safety evaluation for the train bay floor
[redacted]	[redacted]	[redacted]
72-35/0201-05	URI	Adequacy of Holtec 72.48 for hydrogen issue

REDACTED SECTION INCLUDES OFFICIAL USE ONLY - SECURITY RELATED INFORMATION

Closed

72-35/0201-01	NCV	Failure to complete a 50.59 safety evaluation for the train bay floor

Discussed

None

LIST OF ACRONYMS

ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
ARM	Alarming Rate Monitors
ASME	American Society of Mechanical Engineers
atm cc/sec	atmospheres-cubic centimeters/second
CLFM	Calibrated Low Friction Material
CCTV	Closed Circuit Television
CFR	Code of Federal Regulations
dpm	Disintegrations per minute
EAL	Emergency Action Levels
EP	Emergency Plan
EPIP	Emergency Plan Implementing Procedure
FSAR	Final Safety Analysis Report
HI-STORM	Holtec International Storage Module
HI-TRAC	Holtec International Transfer Cask
ISFSI	Independent Spent Fuel Storage Installation
kW	killowatt
LEL	Lower Explosive Limit
MPC	Multi-Purpose Canister
mR/hr	milliRoentgen/hr
MWD/MTU	megawatt days/metric ton of uranium
NDE	Nondestructive Examination
NRC	Nuclear Regulatory Commission
OJT	On-The-Job Training
OJE	On-The-Job Evaluations
OQAPD	Operational Quality Assurance Program Description
PERS	Problem Evaluation Requests (computerized tracking system)
ppm	Parts per Million
PPM	Plant Procedures Manual
psig	lbs/in^2
R/hr	Roentgen/hr
SFS	Spent Fuel Storage Instructions
SNM	Special Nuclear Material
SWP	Site-Wide Procedures
std-cc/sec	Standard Cubic Centimeters/Second
USQ	Unreviewed Safety Question

ATTACHMENT 2

LOADED HI-STORM 100 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-HOURS TO LOAD	PERSON-REM DOSE
1	MPC-68 028	9/02	10.81	32,299	2.89 %	Not available	0.385

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