UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
612 EAST LAMAR BLVD, SUITE 400
ARLINGTON, TEXAS 76011-4125

May 20, 2011

Mr. Peter Dietrich
Senior Vice President and
Chief Nuclear Officer
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, CA 92674-0128

SUBJECT: SAN ONOFRE NUCLEAR GENERATING STATION – INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) INSPECTION REPORT 050-206/2011-011; 050-361/2011-011; 050-362/2011-011; 072-041/2011-001

Dear Mr. Dietrich:

A routine inspection of spent fuel storage activities at the San Onofre Nuclear Generating Station Independent Spent Fuel Storage Installation (ISFSI) was conducted on April 19-21, 2011. The inspection also covered the decommissioning performance and status review of your dismantled Unit 1 reactor facility. At the conclusion of the inspection on April 21, 2011, an exit briefing was conducted with members of your staff. The enclosed report presents the scope and results of this inspection.

The inspections examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection included a review of the current physical condition of your ISFSI facility, radiological conditions, quality assurance program, safety review program, and compliance with selected Technical Specifications. Two special areas were included in this inspection. These were the June 2010 pressurization incident with Dry Shielded Canister (DSC) 41 and your earthquake/tsunami preparations as related to dry cask storage on the ISFSI pad and fuel handling activities to load canisters inside the fuel building.

Based on the results of this inspection, the NRC has determined that one Severity Level IV violation of NRC requirements occurred related to an inadequate safety review of a procedure change, as required by 10 CFR 72.48, to ensure the procedure change did not create the possibility for an accident of a different type than previously evaluated in the safety analysis report. The NRC is treating this violation as a noncited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy because the issue was entered into your corrective action program and you took effective corrective actions. If you contest the violation or the significance of the noncited violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 612 E. Lamar Blvd, Suite 400, Arlington, Texas, 76011-4125; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the San Onofre Nuclear Generating Station facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response if you choose to provide one, will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy or proprietary, information so that it can be made available to the Public without redaction.

Should you have any questions concerning this inspection, please contact the undersigned at 817-860-8191 or Mr. Vincent Everett at 817-860-8198.

Sincerely,

/RA/

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

Dockets: 05000206; 05000361; 05000362; 07200041
Licenses: DPR-13; NPF-10; NPF-15

Enclosure:
NRC Inspection Report 50-206/2011-011;
50-361/2011-011; 50-362/2011-011;
72-41/2011-001

Attachments:
1. Supplemental Information
2. Loaded Casks at the SONGS ISFSI

cc w/enclosure and attachments:

Chairman, Board of Supervisors
County of San Diego
1600 Pacific Highway, Room 335
San Diego, CA 92101

James D. Boyd, Commissioner
California Energy Commission
1516 Ninth Street (MS 34)
Sacramento, CA 95814

Gary L. Nolff
Assistant Director-Resources
City of Riverside
3900 Main Street
Riverside, CA 92522

Douglas K. Porter, Esquire
Southern California Edison Company
2244 Walnut Grove Avenue
Rosemead, CA 91770

Mark L. Parsons
Deputy City Attorney
City of Riverside
3900 Main Street
Riverside, CA 92522

Gary H. Yamamoto, P.E., Chief
Division of Drinking Water and
Environmental Management
1616 Capitol Avenue, MS 7400
P. O. Box 997377
Sacramento, CA 95899-7377

Michael L. DeMarco
San Onofre Liaison
San Diego Gas & Electric Company
8315 Century Park Ct. CP21C
San Diego, CA 92123-1548

Chief, Technology Hazards Branch
Region IX
Federal Emergency Management Agency
Department of Homeland Security
1111 Broadway, Suite 1200
Oakland, CA 94607-4052

Institute of Nuclear Power Operations (INPO)
Records Center
700 Galleria Parkway SE, Suite 100
Atlanta, GA 30339

The Mayor of the City of San Clemente
100 Avenida Presidio
San Clemente, CA 92672

Douglas R. Bauder
Southern California Edison Company
San Onofre Nuclear Generating Station
P. O. Box 128
San Clemente, CA 92674-0128

Steve Hsu
Department of Health Services
Radiologic Health Branch
MS 7610, P. O. Box 997414
Sacramento, CA 95899-7414

Richard J. St. Onge
Southern California Edison Company
San Onofre Nuclear Generating Station
P. O. Box 128
San Clemente, CA 92674-0128

Thomas P. McCool
Southern California Edison Company
San Onofre Nuclear Generating Station
P. O. Box 128
San Clemente, CA 92674-0128

Director, Radiological Health Branch
State Department of Health Services
P. O. Box 997414 (MS 7610)
Sacramento, CA 95899-7414

Electronic distribution by RIV:
Regional Administrator (Elmo.Collins@nrc.gov)
Deputy Regional Administrator (Art.Howell@nrc.gov)
DRP Director (Kriss.Kennedy@nrc.gov)
DRP Deputy Director (Troy.Pruett@nrc.gov)
DRS Director (Anton.Vegel@nrc.gov)
DRS Deputy Director (vacant)
DNMS Director (Roy.Caniano@nrc.gov)
DNMS Deputy Director (Charles.Cain@nrc.gov)
Branch Chief (Blair.Spitzberg@nrc.gov)
Senior Resident Inspector (Greg.Warnick@nrc.gov)
Resident Inspector (John.Reynoso@nrc.gov)
Resident Site Admin (Heather.Hutchinson@nrc.gov)
Branch Chief, DRP/D (Ryan.Lantz@nrc.gov)
Senior Project Engineer, DRP/D (Don.Allen@nrc.gov)
SONGS Administrative Assistant (Heather.Hutchinson@nrc.gov)
Project Engineer, DRP/D (Peter.Jayroe@nrc.gov)
Project Engineer, DRP/D (Zachary.Hollcraft@nrc.gov)
Public Affairs Officer (Victor.Dricks@nrc.gov)
Public Affairs Officer (Lara.Uselding@nrc.gov)
Project Manager (Randy.Hall@nrc.gov)
Branch Chief, DRS/TSB (Michael.Hay@nrc.gov)
Senior Inspector SFST (Rob.Temps@nrc.gov)
Inspector Vincent Everett (Vincent.Everett@nrc.gov)
Inspector Lee Brookhart (Lee.Brookhart@nrc.gov)
SFST Region IV Project Manager (Chris.Staab@nrc.gov)
Fee Coordinator (Marisa.Herrera@nrc.gov)
Regional Counsel (Karla.Fuller@nrc.gov)
Congressional Affairs Officer (Jenny.Weil@nrc.gov)
OEMail.Resource@nrc.gov
OEDO RIV Executive Technical Assistant (Stephanie.Bush-Goddard@nrc.gov)
DRS/TSB STA (Dale.Powers@nrc.gov)

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RIV:DNMS:RSFS	C:RSFS		
JVEverett	DBSpitzberg		
/RA/	***/RA/***		
5/20/11	5/20/11		

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	05000206; 05000361; 05000362; 07200041
License No.	DPR-13; NPF-10; NPF-15
Report No.	50-206/2011-011; 50-361/2011-011; 50-362/2011-011; 72-41/2011-001
Licensee:	Southern California Edison Co. (SCE)
Facility:	San Onofre Nuclear Generating Station (SONGS) Independent Spent Fuel Storage Installation (ISFSI)
Location:	5000 S. Pacific Coast Hwy San Clemente, California
Dates:	April 19-21, 2011
Inspectors:	Vincent Everett, Senior Health Physicist Repository and Spent Fuel Storage, Region IV
Accompanied By:	D. Blair Spitzberg, Ph. D., Chief Repository and Spent Fuel Storage Branch, Region IV Rob Temps, Senior Inspector Spent Fuel Storage and Transportation Office of Nuclear Material Safety and Safeguards Juan Montesinos, Visiting Inspector Consejo de Seguridad Nuclear, Spain
Approved By:	D. Blair Spitzberg, Ph. D., Chief Repository and Spent Fuel Storage Branch
Attachments:	1. Supplemental Inspection Information 2. Loaded Casks at the SONGS ISFSI

EXECUTIVE SUMMARY

San Onofre Nuclear Generating Station
NRC Inspection Report 50-206/2011-011; 50-361/2011-011
50-362/2011-011; 72-41/2011-001

Southern California Edison Company (SCE) has placed 42 canisters in their Independent Spent Fuel Storage Installation (ISFSI) under a 10 CFR Part 72 general license. The Advanced NUHOMS Horizontal Modular Storage System, approved by the NRC as Certificate of Compliance 72-1029, was used at the San Onofre Nuclear Generating Station (SONGS) site to store spent fuel in dry cask storage at a storage pad area adjacent to the dismantled Unit 1 facility. Areva Transnuclear, Inc. is the cask vendor for this system. Spent fuel from all three reactor facilities had been placed in 41 of the canisters, with the first canister loaded into the ISFSI on October 3, 2003. Greater-than-Class-C (GTCC) waste removed from the internals of the Unit 1 reactor was stored at the ISFSI in one canister. The licensee was currently loading canisters and maintaining the ISFSI under Amendment 1 of Certificate of Compliance 72-1029 and Revision 3 of the Final Safety Analysis Report (FSAR).

The inspection conducted at the ISFSI evaluated the current condition of the advanced horizontal storage modules (AHSM) containing the canisters loaded with spent fuel and GTCC. The inspection found the ISFSI to be in good physical condition and in compliance with Technical Specifications and license conditions for the ongoing storage of spent fuel.

The inspection reviewed two special areas. The first related to an incident in June 2010 concerning Dry Shielded Canister (DSC) 41 when both the vent and drain lines were closed briefly after the lid had been welded in place and the canister completely filled with water. When the canister vent port was opened, pressure was released into the helium fill lines and caused a pressure release valve to open, releasing a small amount of water. Review of this incident determined that your staff had approved the loading procedure with steps that resulted in isolating the canister, contrary to the requirements specified in the NUHOMS FSAR. As such, a noncited violation (NCV) was identified. The second special area reviewed included the status of tsunami and earthquake preparedness as it related to the ISFSI operations and the current storage of your canisters. These areas were originally reviewed prior to your first canister loading. However, due to the recent events in Sendai, Japan, the NRC has initiated additional reviews of your preparedness and facility design to determine if additional modifications to your programs should be made. This review effort will be ongoing as additional information related to the conditions that initially occurred in Japan become available to ensure an adequate margin of safety has been applied to your dry cask storage operations and facility.

Operation of an ISFSI at an Operating Plant

- The ISFSI concrete storage modules and pad were being maintained in good physical condition, with no deterioration or cracking observed. There were 55 storage modules located on the pad, with 41 storing spent fuel and one storing GTCC. The next loading campaign this summer will load up to six more storage modules (Section 1.2.a).

- Spent fuel from all three reactor facilities was loaded in the ISFSI. All Unit 1 spent fuel stored at the SONGS site has been loaded into 17 canisters. Twelve canisters have been loaded with Unit 2 spent fuel and 12 have been loaded with Unit 3 spent fuel. Technical Specification requirements related to maximum burn-up, enrichment, heat load, empty slots, and failed fuel were reviewed against records for selected canisters. All Technical Specification requirements had been met (Section 1.2.b).

- Records for the environmental radiological monitoring program were reviewed for the period of 2006 through 2010. Thermoluminescent dosimeters (TLDs) placed around the ISFSI pad typically measured exposure levels at background or slightly higher, unless nearby radioactive material influenced the readings. The most significant influence was the storage of the Unit 2 steam generators near the northern side of the ISFSI (Section 1.2.c).

- Radiological surveys and contamination surveys of the ISFSI pad continued to confirm there was no contamination at the ISFSI and radiation levels were low. Radiation levels from the canisters during handling and welding were well within the Technical Specification limits. Accumulated doses to workers to load each canister were continuing to improve as good ALARA (as low as reasonably achievable) practices were implemented (Section 1.2.d).

- The training program met the requirements of Technical Specification 5.2.2. Training included the requirements in the license, ISFSI operations, and procedures used for ISFSI activities. Job-specific training, consisting mostly of hands-on training under the direction of a qualified individual, was required prior to an individual working independently (Section 1.2.e).

- Temperature monitoring of the AHSMs was being performed in accordance with Technical Specification 5.2.5. Temperature limits had been established for the temperature monitoring system being used by the licensee consistent with the FSAR requirements. No blockage of air vents had been found during tours of the ISFSI. Periodic alarms due to malfunctions, ambient temperature changes, or the insertion of a new canister in an adjacent AHSM were effectively evaluated, resolved, and entered into the corrective action program (Section 1.2.f).

- The licensee was implementing their Part 50 quality assurance program for the ISFSI. The quality assurance program had been adequately incorporated into the ISFSI procedures with appropriate control points and sign-offs. However, the quality assurance program had implemented an auditing program that only reviewed the ISFSI programs in a generic fashion (Section 1.2.g).

- The licensee was actively documenting problems related to the ISFSI in the site corrective action program. Selected condition reports were reviewed and found to be properly dispositioned. An in-depth review was performed on one selected condition related to a rag found in the siphon line of a canister. The issue was properly dispositioned and changes made in the foreign material control program at the fabrication shop (Section 1.2.h)

- One canister of GTCC waste was stored at the ISFSI. The canister contained irradiated and activated metal from the Unit 1 reactor internals. GTCC is allowed for storage at an ISFSI if the licensee verifies that the GTCC meets the storage requirements of the Part 72 license. Calculations were performed and a revision to the 10 CFR 72.212 evaluation was completed to verify compliance (Section 1.2.i).

- The ISFSI had been evaluated for postulated earthquakes and tsunamis that could affect the site. A recent California Coastal Commission report completed a preliminary review of the reactor facilities on the California coast. The report concluded that the San Andreas fault would not produce the same conditions as occurred in Sendai, Japan, in March 2011. However, the licensee was re-evaluating their site preparedness for a major event. The ISFSI pad and storage modules were designed for the postulated earthquake that could occur from a nearby fault. For a tsunami, the estimated water height that could be experienced on the ISFSI pad was 7 feet. This was less than the 50-foot flood level that the AHSMs could withstand based on possible flooding effects analyzed in the FSAR (Section 1.2.j).

- The FSAR does not include an accident analysis for a situation where the canister is filled with water after the lid is welded in place, with the vent and siphon port valves closed, resulting in a pressure build-up in the canister. The FSAR does not evaluate this accident because it states in Section A.4.7.3.2 that the canister is always vented during the heat-up transient. Contrary to this, while loading Canister DSC 41 in June 2010, the licensee briefly closed both valves, allowing the canister to build up pressure. This action was performed in accordance with a revision of the procedure that directed the workers to close both valves. The 10 CFR 72.48 evaluation of the procedure revision did not recognize that the change was inconsistent with the FSAR and should not have been approved. Consequently, a NCV was identified for the failure to perform an adequate 72.48 evaluation of the procedure change (Section 1.2.k).

- The licensee had not implemented a voluntary aging management program at the site since the ISFSI had become operational only 8 years ago. However, the features of the advanced horizontal storage module system and the thicker canister design used at SONGS provided additional protection of the spent fuel from the marine environment. A study conducted in 2002 found that corrosion of the stainless steel canisters would not create storage problems for a considerable time into the future (Section 1.2.l).

- Records required by 10 CFR 72.234(d) related to each canister placed in service were verified for Canister DSC 42 to verify the required records could be retrieved and provided for inspection. The required records were found to include the required information as specified by 10 CFR 72.234(d) (Section 1.2.m).

<u>Review of 10 CFR 72.48 Evaluations</u>

- The licensee and the cask vendor were performing the safety screenings and safety reviews required by 10 CFR 72.48. A sampling of these evaluations was reviewed to determine if an adequate evaluation was being performed. For those reviewed, no issues or concerns were identified related to the conclusions reached (Section 2).

Decommissioning Performance and Status Review at Permanently Shutdown Reactors

- The inspectors toured the footprint of the former Unit 1 facility and reviewed records associated with the licensee's NRC required radiological environmental monitoring program. The inspectors also discussed and reviewed the licensee's voluntary efforts undertaken pursuant to the Nuclear Energy Institute (NEI) Groundwater Protection Initiative. Ambient area radiation levels were influenced by the operating unit's steam generator removal work and the Unit 1 reactor pressure vessel transportation/storage package (Section 3).

REPORT DETAILS

Summary of Facility Status

The ISFSI consisted of 42 concrete advanced horizontal storage modules (AHSMs) which house steel canisters loaded with spent fuel. Spent fuel from all three reactors was stored at the ISFSI in 41 of the canisters. Greater-than-Class-C (GTCC) waste from the Unit 1 reactor decommissioning project was stored in one canister. The advanced aspect of the AHSM was developed for use at high seismic sites and provided for additional dose reduction over the standard horizontal storage modules. The concrete AHSMs provided shielding for the radioactive material and protected the spent fuel from adverse natural phenomena. There were a total of 55 AHSMs on the pad, with 13 empty. The next loading campaign was scheduled for this summer and will load up to six additional canisters.

The radiation levels around the perimeter of the ISFSI were slightly above natural background levels with no unusual radiation levels. The facility was in good condition with no degradation observed on the AHSMs or pad. All AHSMs were properly located on the pad in compliance with requirements for separation, distance from the edge of the pad, and interconnection between adjacent modules. The thermal monitoring program continued to indicate that canister temperatures had remained within Technical Specification limits.

The SONGS site was unique in that a canister fabrication shop was located on the Mesa across the highway from the ISFSI. The fabrication shop constructed the canisters for use at the site under a license issued by the NRC to Areva Transnuclear. During this inspection, a tour of the fabrication shop was completed. The shop was in the process of fabricating new canisters. The tour included the fabrication activities, the quality control program work area, and a review of the foreign material exclusion program.

1 Operations of an ISFSI at Operating Plants (60855.1)

1.1 Inspection Scope

The ISFSI inspection included a site tour of the facility and a review of various programs related to compliance with the Part 72 license requirements. Areas reviewed included radiological environmental monitoring, radiological conditions at the ISFSI, safety evaluations, Technical Specification compliance, implementation of the quality assurance program, and effective use of the corrective action program to identify and resolve issues.

1.2 Observation and Findings

a. Site Tour of ISFSI and Current Status

A tour of the ISFSI pad found the concrete pad and the AHSM in good physical condition. No cracks or deterioration were observed. The ISFSI pad consisted of two adjacent pad areas designed to hold the AHSMs. The pad was 293 feet in length. The first pad area was 43 feet 6 inches wide and held 31 canisters. The second pad area was 60 feet 6 inches wide and was designed to hold a double row of canisters. There were 55 AHSMs on the pad. Radiation levels on and near the ISFSI pad were measured by the NRC inspector during the tour. Radiation levels ranged from 9 to 14 μR/hr at the ISFSI fence entrance. The loaded AHSMs were labeled as "Radioactive Material Area." A radiation rope approximately 10 feet out from the AHSMs established a boundary that was also posted "Radioactive Material Area." Radiation levels along the rope ranged from background to slightly over 100 μR/hr (0.1 mR/hr). Canister DSC 18, which contained the GTCC, was reading 30 μR/hr. For comparison, radiation levels measured inside the NRC Resident Inspector's office ranged from 8 to 13 μR/hr. Radiation levels at the hotel in Oceanside were 9 to 11 μR/hr. During the drive between the site and the hotel, radiation levels varied from 7 to 16 μR/hr.

There were 31 AHSMs in a single row along the west side of the ISFSI pad. A shield wall was on each end of the row. A second row on the west side of the ISFSI was set up as a double array with AHSMs back to back. There were 24 AHSMs in this configuration for a total of 55 AHSMs on the pad, of which 42 were loaded. The 13 empty AHSMs were on the south end of the double row to provide shielding. A shield wall was on the other end.

The AHSMs were required to be in a certain configuration on the pad. Technical Specification 4.4.1 required the AHSMs to be tied together in a single row or in a back-to-back array with no less than 3 modules tied together, side by side. The first row of 31 AHSMs was tied together side by side. The second row consisted of 24 AHSMs were back to back (i.e.,12 AHSMs on a side). These 24 AHSMs were connected together in a row of 12 and interconnected between the two rows. There was approximately 40 feet between the row of 31 and the double-sided row of 24. Both rows were at least 10 feet from the edge of the pad as required by the Technical Specification.

The 55 AHSMs currently on the pad were designed for the 24 PT1-DSC and 24PT4-DSC canisters, which hold a maximum of 24 spent fuel assemblies. The licensee was considering use of the 32-assembly version of the NUHOMS system. With 24 assembly canisters, 38 more AHSMs can be added to the current pad for a total of 93 AHSMs. The AHSMs for storing the 32 assembly canisters were larger. If the 32 assembly canisters are used in the future, the ISFSI will hold a total of 89 AHSMs. The next loading campaign was planned to start in June. Up to six of the currently empty 24PT4-DSC canisters will be loaded with Unit 2 fuel during this campaign.

b. <u>Fuel Loaded at the ISFSI</u>

Forty-two canisters were stored at the SONGS ISFSI. Forty-one of these contained spent fuel from the three reactors. One canister contained GTCC waste from Unit 1. The first canister loaded, containing Unit 1 spent fuel, was placed on the ISFSI pad on October 2003. On September 3, 2004, SCE notified the NRC by letter (ML042530258) that all spent fuel had been permanently removed from the Unit 1 spent fuel pool and stored in the ISFSI. By June 2005, all Unit 1 spent fuel stored in the Units 2 and Unit 3 spent fuel pools had been placed in the ISFSI. The Unit 1 spent fuel was placed in 24PT1-DSC canisters, which are approved for shipment in the MP-187 transportation cask under Certificate of Compliance 71-9255, Revision 10 (ML083300432). There were 395 fuel assemblies loaded into the 17 canisters at the ISFSI. This leaves 270 Unit 1 spent fuel assemblies stored in wet storage at the GE Morris facility in Illinois. These assemblies had been moved to GE Morris when the facility was accepting spent fuel for storage and prior to SONGS constructing an onsite ISFSI.

The Unit 1 canisters were loaded under Amendment 0 of Certificate of Compliance 1029. The Unit 1 fuel was Westinghouse 14 x 14 pressurized water reactor spent fuel assemblies. The final canister loading patterns were documented in an internal memo dated July 15, 2010, entitled "The Special Nuclear Material (SNM) Inventory San Onofre Nuclear Generating Station, Units 1, 2 and 3." The memo contained Attachment 2, "DSC Fuel Loading Pattern" from Procedure SO23-X-9 "Dry Cask Loading," which included the specific location for each fuel assembly, by assembly number, in each of the canisters loaded. The Attachment 2 documents were the final record of the canister content and were signed-off by the Fuels Services Engineer and the Supervisor, Fuel Services, certifying that the canister was loaded in accordance with the assigned calculation.

The final loading patterns and the associated data for enrichment, burn-up, and heat load for Canisters DSC 1 through DSC 17 were compared to Technical Specification 2.1 "Fuel to be Stored in the 24PT1-DSC," associated Table 2-1 "Fuel Specifications," and Table 2-4 "Fuel Qualification Table." The heat load for the 17 Unit 1 canisters, based on the sum of the heat load of the individual assemblies in each canister, ranged from 7.23 Kw to 9.36 Kw. The maximum Technical Specification 2.1.(c) limit was 14 Kw. Burn-up for the hottest assembly in each canister ranged from 34.9 to 43.2 GWd/MTU. The maximum limit from Table 2-4 was 45 GWd/MTU based on specific cooling times and initial enrichment. Maximum enrichment of U-235 for the Unit-1 assemblies loaded in the canisters was 4.02%. Table 2-1 allowed enrichments up to 4.05%.

Technical Specification 2.1.(d) limited the 24PT1-DSC canister to no more than two empty fuel assembly slots in each canister. The empty slots must be located at symmetrical locations about the 0-180º and 90-270º axes. The licensee's Calculation N-1020-164, "Dry Cask Storage 24PT1-DSC Canister Loading Patterns for Unit 1 Fuel Assemblies Stored in the Unit 3 Spent Fuel Storage Pool," Revision 0, was reviewed to verify that the Technical Specification requirement had been included. Section 1.2.6 of the calculation included the requirement. Only Unit 1 spent fuel assembly had been placed in the 24PT1-DSC canister. When fully loaded, the 24PT1-DSC holds 24 spent fuel assemblies. A review of the DSC loading patterns in Attachment 2 of the July 15, 2010, internal memo provided the number and location of fuel assemblies loaded into the 17 canisters containing Unit 1 spent fuel. Of these, ten were fully loaded with 24 assemblies, one was loaded with 23 assemblies, and six were loaded with 22 assemblies. The loading pattern for the six 24PT1-DSC canisters loaded with only 22 fuel assemblies showed that all six canisters had been loaded with a pattern that met the orientation requirement in Technical Specification 2.1.(d). The one canister (DSC 17) that had been loaded with 23 fuel assemblies, leaving one empty slot, met the requirement in Technical Specification 2.1.(d) by using a dummy fuel assembly in the empty slot that approximated the weight and center of gravity of a fuel assembly. As such, all slots were filled in Canister DSC 17. All 17 canisters where in compliance with the Technical Specification limit for no more than two empty slots.

The Unit 2 canisters were loaded under Amendment 1 of Certificate of Compliance 1029. The Unit 2 Combustion Engineering 16 x 16 pressurized water reactor spent fuel assemblies were loaded into 24PT4-DSC canisters. Unit 2 spent fuel was initially placed in the ISFSI in March 2007. There were 12 canisters, containing 288 assemblies, loaded with Unit 2 spent fuel. This left 1221 spent fuel assemblies in the Unit 2 spent fuel pool. The pool had a capacity of 1542 spent fuel assemblies, though not all slots were available due to storage of highly radioactive material in several of the fuel slots. Canisters DSC 19 through 25 and DSC 32 through 36 were loaded with Unit 2 spent fuel. A review of the DSC loading patterns in Attachment 2 of the July 15, 2010, internal memo provided the number and location of the fuel assemblies loaded into the 12 canisters. Requirements for the fuel loaded into the 24PT4-DSC canister were listed in Technical Specification 2.2, “Fuel to be Stored in the 24PT4-DSC,” and associated Tables 2-9 through 2-16.

The heat load for the 12 canisters, based on the sum of the heat load of the individual assemblies in each canister, ranged from 9.66 Kw to 14.71 Kw. The maximum heat load allowed by Technical Specification 2.2.(k) was 24 Kw. Burn-up for the hottest assembly in each canister ranged from 38.3 to 48.3 GWd/MTU. The maximum limit from Technical Specification 2.2.(i) was 60 GWd/MTU, based on enrichment and cooling times in Tables 2-9 through 2-16. Maximum enrichment of U-235 in any of the canisters was 4.49 percent. The maximum enrichment allowed by Technical Specification 2.2.(i) was 4.85 percent. This limit was further defined in Technical Specification 2.2.(l), which identified a Type A standard basket and a Type B high boral basket. The Type A basket was limited to an initial U-235 enrichment of 4.1 percent. The Type B basket was limited to 4.85 percent. Starting with Canister DSC 32, the U-235 enrichment exceeded the 4.1 percent. Selected records for several of these canisters were reviewed to confirm Type B canisters only were used.

The 24PT4-DSC did not have the same limitation as the 24PT1-DSC concerning empty slots. Technical Specification 2.2.(j) allowed the 24PT4-DSC canister to contain empty slots or dummy fuel assemblies. For the 12 Unit 2 canisters, all slots were occupied with no empty slots in any of the canisters. The 24PT4-DSC canister had been included in the application for Amendment 3 to the MP-197 transportation cask Certificate of Compliance 71-9302, which was currently under review by the NRC.

The Unit 3 canisters were loaded under Amendment 1 of Certificate of Compliance 1029. The Unit 3 Combustion Engineering 16 x 16 pressurized water reactor spent fuel assemblies were loaded into 24PT4-DSC canisters. Unit 3 spent fuel was initially placed in the ISFSI on March 2008. There have been 12 canisters, containing 288 assemblies, loaded with Unit 3 spent fuel. This leaves 1229 spent fuel assemblies in the spent fuel pool, which has a capacity of 1542 assemblies. The Unit 3 spent fuel pool also had slots occupied with highly radioactive material in storage. Canisters DSC 26 through 31 and DSC 37 through 42 were loaded with Unit 3 spent fuel. A review of the DSC loading patterns in Attachment 2 of the July 15, 2010, internal memo provided the number and location of the fuel assemblies loaded into the 12 canisters. Requirements for the fuel loaded into the 24PT4-DSC canister were listed in Technical Specification 2.2, “Fuel to be Stored in the 24PT4-DSC,” and associated Tables 2.9 through 2-16.

The heat loads for the 12 canisters containing Unit 3 fuel ranged from 7.78 to 15.39 Kw. The maximum heat load allowed by Technical Specification 2.2.(k) was 24 Kw. Burn-up for the hottest assembly in each canister ranged from 29.5 to 50.1 GWd/MTU. The maximum limit from Technical Specification 2.2.(i) was 60 GWd/MTU, based on enrichment and cooling times in Tables 2-9 through 2-16. Maximum enrichment of U-235 in any of the canisters was 4.6 percent. The maximum enrichment allowed by Technical Specification 2.2.(l) was 4.1 percent for a Type A basket and 4.85 percent for a Type B basket. Fuel stored in Canisters DSC 26 through 31 was all less than the 4.1 percent enrichment and were stored in Type A canisters. Selected records for Canisters DSC 37 through 42, which were above the 4.1 percent limit, were reviewed and confirmed as Type B canisters. All Unit 3 canisters were filled with 24 spent fuel assemblies with no empty slots.

The location of the spent fuel assemblies in Canisters DSC 36 through 42 were reviewed against Technical Specification 2.2.(k) related to the three allowed loading patterns based on decay heat of the assembly as shown in Figures 2-1 through 2-3. The computer software code, CaskWorks, was used to generate the loading patterns. Calculation N-1020-176, “Dry Cask Storage 24PT4-DSC Canister Loading Pattern for Unit 3 Fuel Assemblies: 2008 Loading Campaign,” dated December 6, 2007, Calculation N-1020-006, “Dry Cask Storage 24PT4-DSC Canister Loading Pattern for Unit 2 Fuel Assemblies: 2009 Loading Campaign,” dated January 30, 2009, and Calculation N-1020-177, “Dry Cask Storage 24PT4-DSC Canister Loading Pattern for Unit 3 Fuel Assemblies: 2010 Loading Campaign,” dated January 14, 2010, provided the approved loading patterns for the canisters reviewed. The latest loading calculations had incorporated the concept of placing the hotter fuel in the inner slots to reduce the dose to workers. Attachment 2 of the July 15, 2010, internal memo provided the final location for each of the assemblies. All seven canisters reviewed complied with the required loading pattern.

Placement of failed fuel in the NUHOMS DSC canisters was required to comply with Technical Specification 2.1 for the 24PT1-DSC canister and Technical Specification 2.2 for the 24PT4-DSC. The 24PT1-DSC was allowed 4 damaged fuel assemblies per canister, placed in failed fuel cans. The 24PT4-DSC was allowed up to 12 damaged fuel assemblies per canister, placed in failed fuel cans. The licensee determined whether fuel was damaged by a combination of visual inspections using a camera, review of reactor operating records, wet fuel sipping, ultrasonic testing (UT), and eddy current testing. If reactor operating records indicated the potential for fuel damage, then fuel sipping was performed as the fuel was off-loaded from the core. If sipping identified a potential leaker, then UT and/or eddy current testing was used to identify the individual rod that had failed. The problem creating the leaking fuel rods related to the current design of the assemblies not having enough grid-to-rod fretting (GTRF) margin for the conditions at the core periphery. The standard design fuel used at SONGS was designed in the early 1980's when annual cycles and relatively low fuel duty cycles were normal. As cycle lengths and fuel duty increased, GTRF wear margin increased. The ISFSI, as currently loaded with 41 canisters containing spent fuel, has 9 canisters containing 27 Unit 1 failed fuel assemblies (Westinghouse 14 x 14 fuel), 4 canisters containing 46 Unit 2 failed fuel assemblies (CE 16 x 16 fuel), and 2 canisters containing 22 Unit 3 failed fuel assemblies (CE 16 x 16 fuel). The limit of 4 failed fuel cans in the 24PT1-DSC and 12 in the 24PT4-DSC canister were reviewed against the final loading configurations. The failed fuel was properly loaded in accordance with the Technical Specification limits.

Decay heat calculations for the canisters stored at the SONGS ISFSI were calculated using SAS2H/ORIGEN-S. Power for each assembly was calculated by multiplying burn-up by the metric tons of uranium divided by days of operation and a 0.95 factor for initial boron concentration of the first cycle. The hardware activation gamma source was also accounted for, with appropriate flux correction for bottom plenum. A +7 percent uncertainty was assumed and added to the burn-up calculation.

c. Radiological Environmental Program

The SONGS Radiological Environmental Reports for calendar year 2006, dated April 2007 (ML071380337), calendar year 2007, dated April 2008 (ML081350336), calendar year 2008, dated April 2009 (ML091320637), calendar year 2009, dated April 2010 (ML101320085), and calendar year 2010, dated May 9, 2011 (ML111310156) were reviewed. The ISFSI environmental TLD results were provided in Appendix J of the annual reports. There were nine TLDs placed along the perimeter of the ISFSI fence. These were TLDs 309 thru 312 and TLDs 322 thru 326. The TLDs along the northeast side of the ISFSI fence (TLDs 309 – 312) ranged from 13 to 21 mrem/quarter for 2010, prior to background subtraction. Background levels were estimated by the licensee to average 15 mrem/quarter (approximately 7 μR/hr). The TLDs located on the side of the ISFSI toward the nuclear facilities and the north side near the stored Unit 1 reactor vessel (TLDs 324 - 326) measured higher dose rates. These ranged from 17.9 to 112 mrem/quarter. TLD 325 had shown unusually high radiation levels beginning in the 4^{th} quarter 2009. This coincided with the storage of the Unit 2 steam generators, which were moved into the north industrial area, near TLD 325, in the 4^{th} quarter. The two adjacent TLDs near TLD 325 also showed the effects of the nearby steam generators. TLD 326, located near TLD 325 on the north side fence, also read as high as 44.1 mrem/quarter in 2010. TLD 324, located south of TLD 325 measured a high of 35.6 mrem/quarter in 2010. During the development of the annual radiological environmental report for 2011, the high results for TLDs 325 and 326 were documented in the licensee's corrective action reporting system in Nuclear Notification 201407401 dated April 6, 2011. The report identified that the higher dose rates were due to the nearby steam generators. The closest publically accessible location to the ISFSI is the San Onofre beach access road. TLDs 55 and 56 provided background corrected readings of 13 and 14 mR for 2010 near the publically accessible location. Assuming a maximum occupancy factor of 300 hours/year, the dose to the general members of the public would be < 1 mrem/yr.

Neutron dosimeters were placed in locations 311, 324, 325, and 326 starting the 4th quarter of 2009. No neutron doses have been recorded on the TLDs.

The licensee was required by 10 CFR 72.44(d)(3) to provide an annual radioactive effluent release report to the NRC. This requirement was also included in Technical Specification 5.2.3(c). The reports for 2007 thru 2010 were reviewed. No releases of any effluents were reported from the SONGS ISFSI.

d. Radiological Conditions Associated with the ISFSI and Cask Loading

Requirements for the radiation protection program associated with the ISFSI were incorporated into Technical Specification 5.2.4, “Radiation Protection Program.” Technical Specification 5.2.4.(b) required environmental monitoring to ensure annual dose equivalent to any real member of the public located outside the ISFSI controlled area did not exceed regulatory limits. The radiological environmental monitoring program included more TLDs than just those discussed in the previous section above. The licensee reported results for 24 locations associated with TLDs on and near the ISFSI in the 2010 radiological environmental report. When evaluating the TLD results versus their location in relation to the Unit 2 steam generators, the Unit 1 reactor vessel, and the Units 2 and 3 operating units, the influence from these other activities can be associated with the higher TLD results. When selecting TLDs at a location with minimal influence from other activities, the TLD values were at background or only slightly higher. As such, the ISFSI contribution was far below the 10 CFR 72.104(a) limit of 25 mrem/year to the nearest real member of the public at the controlled area boundary. Since there were no radiological releases from the ISFSI with the spent fuel sealed in canisters, the thyroid and critical organ limits in 72.104(d) were not applicable.

The requirement to perform the contamination survey of the canister prior to placement in the AHSMs was specified in Technical Specification 5.2.4.(c). The licensee had incorporated this requirement into Procedure SO23-I-30.9, “24PT4 DSC Dry Cask Storage Loading,” Revision 13, Step 6.2.3, and performed the required survey in Step 6.7.13. Records for Canisters DSC 37 though 42 were reviewed. All six canisters were documented in the procedure as meeting the contamination limit. The contamination surveys were conducted according to Health Physics Instruction HP-I-17, “Health Physics Division Performance Instruction,” Revision 0.

A radiological survey was required of the loaded transfer cask by Technical Specification 5.2.4.(d) to confirm dose rates were consistent with the offsite dose analysis. Two limits were established in the Technical Specifications. The first limit was 260 mrem/hr gamma at 3 feet from the centerline of the top of the welder neutron shield prior to wet welding operations, with the shield plug in place, approximately 4 inches of water drained, and the welder with its neutron shield in place. The second limit was 95 mrem/hr at 3 feet from the surface of the transfer cask neutron shield at the centerline of the transfer cask prior to wet welding operations. The licensee had established these requirements in Procedure SO23-I-30.9, Step 6.2.4, and performed the required survey in step 6.7.25. To remove enough water to reduce the water level 4 inches required approximately 50 gallons to be removed. This was performed in Steps 6.7.4, 6.7.14, and 6.7.18.2 of the procedure prior to taking the radiological survey in Step 6.7.25. Records for Canisters DSC 29, 30, 31, 36, 41, and 42 were reviewed. Concerning the first limit of 260 mrem/hr, the readings obtained from the six canisters reviewed ranged from 0.3 mrem/hr (DSC 31) to 3.0 mrem/hr (DSC 42). These six canisters ranged in heat load from 7.78 Kw (DSC 31) to 15.39 Kw (DSC 42), with the lowest heat load producing the lowest reading and the highest heat load producing the highest reading, as expected. These values were well within the Technical Specification limit of 260 mrem/hr. For the second limit of 95 mrem/hr, the six canisters ranged from 2 mrem/hr (DSC 31) to 11 mrem/hr (DSC 42). The radiological surveys were performed in accordance with Procedure SO123-VII-20.9, “Radiological Surveys.”

The SONGS radiological protection program was implemented for the ISFSI pad to perform periodic radiological surveys and contamination checks. The program was implemented under Procedure SO123-VII-20.9. Monthly survey records for the period June 2010 through March 2011 were reviewed. During that period, no contamination was found on the ISFSI pad. For the radiological surveys conducted on the pad, dose rates ranged from 8 to 150 μR/hr (0.15 mR/hr). Radiation readings taken in contact with the AHSMs doors measured levels up to 1.8 mR/hr, with most AHSMs reading less than 1 mR/hr. A limited amount of neutron surveying was performed on the pad with maximum neutron levels measured at 0.4 mrem/hr.

The licensee was implementing an ongoing effort to maintain exposures during cask loading to low levels and to continually improve upon their overall accumulated personnel exposures during each new canister loading. The first canister loaded in 2003 resulted in 1.366 person-rem. This has been the highest exposure received of any canister loaded to date at SONGS. From 2003 through mid-2005, all of the Unit 1 fuel was loaded into 17 canisters. The average exposure for the Unit 1 fuel loading was 0.699 person-rem/canister. In 2007 and again in 2009, Unit 2 spent fuel was loaded into canisters. Twelve canisters were loaded with an average of 0.351 person-rem/canister. The Unit 2 spent fuel had a higher average heat load of 12.2 Kw/canister compared to the Unit 1 spent fuel of 8.2 Kw. Typically, the higher the Kw value, the higher the dose rates around the canister. For the Unit 3 spent fuel, 12 canisters were loaded in 2008 and 2010. The average heat load was similar to the Unit 2 spent fuel at 11.8 Kw. The exposure of personnel continued to decline with an average of 0.253 person-rem. The more recent fuel loadings have included hotter fuel, with Canisters DSC 40, 41, and 42 over 15 Kw. These three canisters averaged 0.313 person-rem. The licensee has made good progress over the past several years to implement sound ALARA practices.

The ISFSI program was included in the overall site ALARA program. As such, the ALARA committee approved the dose goals for each loading campaign. After each loading campaign, workers were provided an opportunity to discuss ways to improve the loading campaign and reduce exposures. For Canister DSC 31, a video of loading from beginning to end was made and a critique provided to the station ALARA committee with recommended improvements. Improvements to reduce exposures included considering new equipment such as an automatic vacuum drying system, establishing loading patterns for the fuel with the hotter fuel assemblies located in the center of the canister with the lower dose assemblies on the outside to provide a shielding effect, adding additional shielding, performing time-motion studies to evaluate why some canisters are taking more man-hours to load than others, and evaluating why canisters with similar heat loads varied in the overall exposure to complete loading. Numerous nuclear notifications had been entered into the corrective action system discussing ALARA and ALARA improvement ideas, indicating a very active process was in place to continue to reduce employee exposures.

e. Training Program

Technical Specification 5.2.2 required a comprehensive training program for the operation and maintenance of the dry storage system and listed required elements that shall be included in the licensee's training program. Selected elements from the Technical Specification were verified as being incorporated into the licensee's training modules and personnel qualification standards. These elements included operating experience reviews, off-normal and accident conditions, and responses and corrective actions.

ISFSI training was described in Procedure SO123-XXI-1.11.27, "ISFSI Training Program Description," Revision 3. Initial basic training required to be completed by all personnel consisted of three modules: Module 3ISFS01, "NUHOMS 72-1029 Certificate of Compliance Overview," Module 3ISFS02, "NUHOMS Cask/DSC Preparations Operations," and Module 3ISFS15, "ISFSI General Worker Required Procedure Review." An exam score of 80 percent was required to pass. In addition, specific training for each of the following worker groups was required:

- ISFSI General Worker
- ISFSI Crane Operator
- ISFSI Prime Mover Operator
- ISFSI Transfer Trailer Operator
- ISFSI Alignment Operator
- ISFSI Draining and Drying Operator
- ISFSI Hydraulic Ram Operator
- ISFSI Fuel Movement Supervisor

Each group was required to qualify on their specific modules. The required set of qualifications needed for different positions was established in the "San Onofre Unit 2 Dry Fuel Storage Campaign 3 Training Matrix," dated April 19, 2010. Training was provided to individuals while working with their supervisor using the cognizant operating procedure. This on-the-job training was fail or pass according to the supervisor's judgment. The workers were trained to learn the operating procedures under supervision and, if successfully qualified by the supervisor, to work independently. A personnel qualification standard was completed and signed by the training materials developer, technical reviewer, line supervisor, and cognizant training manager. The initial training was a one-time training.

Re-training was described in Procedure SO123-XXI-1.11.27, Section 6.6, “Continuing Training Program.” The re-training program was required by the FSAR, Section A.9.3.2, which referenced FSAR Section 9.3.2. FSAR Section 9.3.2 simply stated “Retraining is generally consistent with the retraining requirements in effect at the plant for personnel involved in fuel handling operations.” Step 6.6.1.1 of Procedure SO123-XXI-1.11.27 stated that the ISFSI continuing training curriculum shall be determined by the maintenance training manager and line management based on new industry events and the time since the last training was performed. Sources of information for the lesson learned included not only internally learned lessons by the licensee, but also information from NRC’s operating experience, from the Institute of Nuclear Power Operations (INPO) data base, and information collected by project personnel attending industry seminars and workshops that included ISFSI lessons learned information. Prior to a new campaign, workers were required to attend a refresher seminar. The licensee was working on the refresher seminar to support the June 2011 loading campaign with lessons learned from the 2009 loading campaign and from post-job critique notes and meeting minutes from the 2010 loading campaign. Refresher training also included any procedure changes since the last campaign. For the refresher training, there was no sufficiency test, just a check that personnel assigned to the loading campaign had attended the seminar. An attendance list was provided for attendees to sign.

Procedure SO123-XXI-1.11.27, Section 7.0, “Records,” identified the requirements for establishing and maintaining training records. Records were maintained in the training records information management system (T2000) and archived by corporate document management. A computerized training matrix that provided the current status of training for each individual by name was maintained. The licensee provided a list of qualified personnel from the last loading campaign. Qualifications for selected individuals on the computerized training matrix from different work groups were verified against the records from the records files.

f. <u>AHSM Temperature Monitoring Program</u>

Temperature monitoring and visual inspections of AHSM vents was required by Technical Specification 5.2.5, “AHSM Thermal Monitoring Program.” Technical Specification 5.2.5.(c) required daily visual inspection of the AHSM air vents loaded with 24PT1-DSC canisters. This daily vent inspection requirement did not apply to the 24PT4-DSC canisters. Operating Procedure SO23-3-2.37, “AHSM System,” Revision 4, provided the instructions for the visual inspection of the vents and the thermal monitoring program. Step 4.1.2 required daily visual inspection of Canisters DSC 1 through 18. These were the canisters that contained the Unit 1 spent fuel and the GTCC. This visual inspection requirement was not included in Section 4.2 of the procedure related to the remaining canisters, which were 24PT4-DSC canisters. If any blockage of an AHSM containing a 24PT1-DSC canister was found during the daily surveillance, the blockage was required to be removed such that the 40-hour limit specified in Technical Specification 5.2.5.(a) was not exceeded. No blockage had occurred at any of the canister vents in the ISFSI.

Temperature monitoring of all AHSMs was required by Technical Specification 5.2.5.(a). For the AHSMs containing 24PT1-DSC canisters, daily temperature monitoring was required with a 40-hour blocked vent time limit. An 80ºF temperature rise limit and a 225ºF overall temperature limit were established for the AHSMs containing the 24PT1-DSC canisters. The Technical Specification limits were established to alert the operator to possible blockage of the air vents. These limits were based on a single thermocouple monitoring system. The licensee used dual thermocouples. FSAR Section 4.4.2.5, “Monitoring of AHSM Temperatures,” and Table 4.4-12, “Technical Specification 5.2.5.(a), Temperature Monitoring Limits for the 24PT1-DSC,” established different temperature limits for dual thermocouple systems. The temperature rise was limited to 8ºF when monitoring the canister temperatures on a 24-hour basis. The maximum temperature limit was 175ºF. The licensee provided a review of temperature records dating back to January 2007. The 8ºF and 175ºF limits had not been exceeded during that time frame, except for a single incident in which the temperature monitoring system for multiple AHSMs alarmed simultaneously, then reset 2 - 3 minutes later. The vents were verified as clear and the incident has not recurred. This problem was documented in the licensee’s corrective action program as NN 201164625.

For the AHSMs containing the 24PT4-DSC canisters, the Technical Specification 5.2.5.(a) temperature rise limit was 30ºF with an overall upper limit of 225ºF. Temperatures were required to be measured twice a day. If these limits were exceeded, the licensee had a 25-hour blocked vent time limit. Since the licensee was using a dual thermocouple monitoring system, these limits were modified to comply with FSAR Section A.4.4.2.4, “Monitoring of AHSM Temperatures,” and Table A.4.4-11, “Technical Specification 5.2.5.(a) Temperature Monitoring Limits for the 24PT4-DSC.” The temperature rise limit was 5ºF for a 12-hour period with a maximum temperature limit of 200ºF. The 200ºF limit had not been exceeded during the period reviewed back to January 2007; however, the 5ºF limit in 12 hours had been exceeded several times. Each incident had been entered into the corrective action program and the vents found to be clear. Most alarms were due to a recent insertion of a new canister into the AHSM. Some of the alarms could be related to ambient temperature rises from the morning to the afternoon. For Canister DSC 40, the temperature rise happened after the adjacent AHSM was loaded with the hotter Canister DSC 42.

g. Quality Assurance Program

The licensee was implementing their Part 50 quality assurance program for the ISFSI activities. The licensee had incorporated various quality control hold points and sign-off steps in procedures. Quality controls were being applied to the important-to-safety work. The licensee had effectively incorporated the quality assurance program into the ISFSI procedures that were being implemented by the workers. Numerous surveillances were being performed for a broad range of work tasks involving dry fuel storage.

The quality assurance program, however, was not providing a strong oversight function of the ISFSI activities from the perspective of auditing. Auditing is described in 10 CFR 72.176 as a comprehensive system of planned and periodic audits to verify compliance with all aspects of the quality assurance program to determine the effectiveness of the program. This auditing function is a key independent confirmation that the workers are properly implementing the quality assurance program requirements. The licensee's quality assurance program was performing the required audits specified by Part 50 for the power plant operations, but was viewing the ISFSI as a subset of the overall site auditing program. As such, since the ISFSI was only a small part of site activities, auditing of the various site programs rarely included specific evaluations of activities being performed to load canisters and store spent fuel at the ISFSI. An exception to this was the ISFSI security program, which had received several ISFSI specific audits.

As an example, if an audit of the instrument calibration program was conducted, randomly selected instruments were chosen for verification that the quality assurance program was being properly applied. The activities associated with the calibration of the specific instrument and the operations of the instrument calibration lab were reviewed. This fulfilled the site quality assurance requirement to conduct an audit per 10 CFR 72.164, "Control of Test and Measuring Equipment," but it only generically encompassed the ISFSI calibration activities. Because the instrumentation used at the ISFSI represented such a small portion of the overall instrumentation used at the site, the likelihood that the audit team would chose an instrument used for ISFSI activities was very low and, in fact, there was no evidence that ISFSI related instrumentation had been audited in the past several years. During the interviews conducted by the NRC of the quality assurance staff, it was clear that the ISFSI had been considered a minor area of the overall audit program. The licensee's staff was unable to provide detailed examples of comprehensive audits performed of ISFSI activities and programs or where emphasis had been placed on implementing the Part 72 quality assurance requirements related to the ISFSI work activities. Since Part 72 is specific to ISFSIs, and Part 72 specifies a quality assurance program equivalent to Part 50, audits would be expected to be conducted specific to ISFSI activities. In a broader picture, the NRC inspectors found few examples where implementation of the site quality assurance program was being applied to the ISFSI program at the level expected by Part 72. Despite this, the inspectors found through interviews with ISFSI workers and review of the ISFSI programs and procedures that workers were complying with Part 72 quality assurance requirements.

h. Corrective Action Program

The licensee was actively documenting problems related to the ISFSI in the site corrective action program. Issues were effectively being captured and resolved. A list of condition reports issued over the past several years related to the ISFSI and the cranes was reviewed. From this list, the following condition reports were reviewed.

Report	Title	Issue
NX 201412324 and NX 201395210	ISFSI/DSC Design Change	This was a quality surveillance of design documents to verify they were being prepared, reviewed, approved, and processed in accordance with engineering procedures. Calculations, drawings, supplier documents, and supplier deviation requests were reviewed and found in compliance with the procedures.
NX 201410203	ISFSI/DSC Design Changes	This was a quality review of the design documents related to the temperature monitoring systems for AHSMs 48 through 63. No problems with the documents were identified.
NN 200982175	Piece of Cloth Found in DSC Fitting	While attempting to drain water from DSC 41 prior to welding, no water came out. A rag was found clogging the siphon drain line. The rag had been placed in the siphon line during canister fabrication to prevent grinding dust from getting into the siphon line. After removal of the rag, the water was successfully drained.
NN 200974685	DSC 40 Helium Backfill	This was a quality surveillance to verify that the correct pressure of helium was added to the canister after vacuum drying. The procedures were correctly followed
NN 200913221	Revise HP-I-17 Procedure	Alpha samples collected during a required survey of the cask were not counted in a timely manner. Health physics coverage for cask activities was not covered by procedures, but used a health physics instruction. The issue concerned whether sufficient detail was included in the procedure. The health physics staff considered the problem to be human performance and not a lack of detail in the procedure.
NX 200892142	DSC 37 Work Stoppage	During preparations to place DSC 37 into the transfer cask, the paperwork verifying cleanliness of the canister was missing a signature. Work was stopped until the proper signature was obtained.
NX 200890644	Transfer Cask Contamination	While the transfer cask was loaned to another facility, it was exposed to contamination levels higher than normal. After cleaning, contamination reappeared due to leaching from the metal. Cleaning by the radiation protection staff continued and workers were made aware of the issue.

Report	Title	Issue
NN 200448877	Unanticipated Volume of Helium for DSC 36	During the blow-down and vacuum drying of DSC 36, nuclear fuels services decided to blow-down the canister twice prior to vacuum drying. This departure from normal requirements required obtaining additional helium bottles. Difficulties were encountered with the delivery of the additional bottles on short notice and moving them to the Unit 2 fuel handling bay.
NN 200444654	DSC 35 & 36 Comparison of Drying Times	Drying times for the two canisters were very similar until the very low pressures. DSC 35 had a heat load of 1.7 kW lower than DSC 36. It was not clear why the difference in drying times.
NN 200423041 and NN 200421277	DSC 35 Vacuum Drying Time	Typical expected time to dry the canister to 25 torr, then hold the pressure to 50 torr for 5 minutes with the canister isolated was 3 hours. For DSC 35, the time was 30 hours. Vacuum pump problems were suspected. Water was observed in the oil reservoir.
NN 200402578	DSC 34 VDS Drying Time	Typical expected time to dry the canister to 25 torr, then hold the pressure to 50 torr for 5 minutes with the canister isolated was 3 hours. For DSC 34, the time exceeded 28 hours. No clear reason for the delay was determined.
NN 200400869	Request ALARA Review of Loading Pattern	An ALARA review was requested of the loading patterns being used for the canisters. It was felt that priority was not being given to placing the hottest fuel assemblies in the most shielded slots or placing the hottest assemblies where they would be on the bottom during transport to the ISFSI pad. Additional attention was applied to future loadings to emphasis ALARA when planning the loading patterns.
NN 200377091 and NN 200378726	DSC 33 VDS Unexpected Drying Time	Typical expected time to dry the canister to 25 torr, then hold the pressure to 50 torr for 5 minutes with the canister isolated was 3 hours. For DSC 33, the time was 27 hours. No clear reason for the delay was determined.
NN 200322678	AHSM Grout Patch Degradation	AHSM 6 showed degradation and partial detachment of some of the grout patches applied during construction. AHSMs 1 and 2 showed similar problems, but less severe. The licensee repaired the patched areas.

Report	Title	Issue
NN 200322247	AHSM/Door Fastener	This was a quality surveillance that observed the refueling group perform a period door fastener torque verification to confirm the AHSM door nuts were torqued to 45 pounds. The torque verification was successfully performed.
NN 200305172	ISFSI Campaign Heat Load/ Dose Correlation	An observation was made in Feb. 2009 that a possible relationship between the canister heat load and the job dose could be developed. The correlation was obtained by fitting a linear relationship, which suggested a relationship of dose in mrem equals (0.061 times heat load in watts) minus 375.
NX 200212109	AHSM/Door Stud Engagement	This was a quality surveillance that verified the AHSM door studs had a minimum 3-pound engagement. All four studs on AHSM 37 were installed properly.
NX 200144352	AHSM/Concrete Inspection Qualifications	This was a quality surveillance that verified the individuals performing the grout inspections were qualified to Standard 3TPI06. All individuals were found to be qualified.
AR 070301362	White Cloud Observed While Loading DSC 21	While loading DSC 21, a white cloud of particles was observed in the canister. There was no white material observed in the fuel racks where the assemblies had been originally stored and no cloud followed the assemblies over to the canister. The canister was unloaded and a sample of the clouded water was filtered and analyzed. The canister was thoroughly flushed and reloaded with the spent fuel. The problem did not recur. Analysis determined the probable cause was due to inadequate flushing of the canister after being cleaned at the fabrication shop. The chemical analysis showed the most likely source was one of the cleaning agents. The sample was found to have a pH of 7.

A special review of the issue related to NN 200982175 was performed. On June 24, 2010, the licensee found a piece of rag blocking the siphon line of Canister DSC 19 during the process of removing water from the canister prior to welding of the lid. The removal of approximately 53 gallons of water prior to welding was being performed in accordance with Steps 6.7.4 and 6.7.18 of Procedure SO23-I-30.9, Revision 12, and was necessary to remove enough water from under the lid to prevent a heat-sink during welding. At the time of the blockage due to the rag, the vacuum drying system liquid pump was connected to the siphon line with a drain line going to the spent fuel pool. The vent line was open to the atmosphere. The pump, after removing approximately

5 gallons in Step 6.7.4, encountered blockage and failed to remove the remaining 48 gallons of water per Step 6.7.18. The siphon line swagelock fitting was removed and a piece of rag found. The rag was removed, the swagelock fitting replaced, and the pump-down continued. The approximately 48 gallons of water was successfully removed. The configuration that existed at the time the rag problem surfaced could not have resulted in a pressure build-up in the canister, since a suction was being applied to the canister at that time. After the welding was completed, the water was replaced in the canister to fill the area under the lid and remove any air inside the canister. This was performed by injecting service water through the siphon port with the vent port open to atmosphere per Step 6.9.1 of Procedure SO23-I-30.9. The canister was then drained by injecting helium into the canister through the vent port and allowing the water to drain into the spent fuel pool through a hose connected to the siphon port. This was performed per Steps 6.9.6 through 6.9.18 of Procedure SO23-I-30.9. The approximately 1,760 gallons of water in the canister after the welding was completed was successfully drained. Upon completion of the draining, both the siphon port and the vent port were connected to the vacuum drying skid, providing for two paths to vacuum dry the canister. The canister was successfully vacuum dried to 3 torr per Sections 6.10 (initial) and 6.12 (final) of Procedure SO23-I-30.9 and backfilled with helium per Sections 6.11 (initial) and 6.13 (final). The vacuum drying and helium backfill was performed using calibrated important-to-safety gauges.

During this inspection, a tour of the SONGS Mesa fabrication shop was conducted and a discussion held with the shop manager concerning the rag. When the siphon line was being fabricated, the line was measured for the correct length and cut. To ensure that no metal filings would get into the line during grinding, a rag was typically placed in the line. After grinding was completed, the rag would be removed. The use of the rag was not proceduralized. During the construction of Canister DSC 41, the worker placed the rag in the tubing, but then completed his shift without performing the grinding. When the individual on the next shift began the grinding operation, he was not aware that a rag was already in the tubing and he placed a second rag into the tube. Upon completing the work, he removed his rag. As a result, the original rag was not removed.

As a corrective action from this incident, the licensee has strengthened the controls applied to the fabrication shop's foreign material exclusion program. Rags are no longer used when grinding on the siphon line. Instead a foam plug with an attached string and sign stating that the plug had been installed is now used. The procedure was revised to include a step to install the foam plug and to remove it upon completion of the work on the siphon line.

i. <u>Storage of Greater-Than-Class-C (GTCC) Waste</u>

Canister DSC 15 contained GTCC from Unit 1. The NRC allows GTCC to be stored in the ISFSI. Interim Staff Guidance Document 17, "Interim Storage of Greater-Than-Class-C Waste," provides the NRC position related to GTCC storage at an ISFSI. GTCC stored at an ISFSI must be in solid form and stored in a separate container. The waste can consist of irradiated and activated metal components from reactors, as well as filters and resins from reactor operations and decommissioning. An analysis under

10 CFR 72.212 must be performed to verify the storage of the GTCC complies with the Part 72 license requirements. Federal Register Notice 66FRN51823 dated October 11, 2001, provides the NRC's final rule on GTCC.

On September 14, 2004, SCE submitted a letter to the NRC (ML042600206) describing the GTCC planned for storage at the ISFSI. In order to prepare the Unit 1 reactor vessel and its internals for disposal at a licensed radioactive waste disposal facility, it was necessary to remove portions of the baffle and core barrel stainless steel material from the beltline region of the vessel in order to meet waste classification criteria. This material was placed in a canister at the ISFSI. The canister containing the GTCC was referred to as a GTCC-DSC. The GTCC-DSC was of the same design as a 24PT1-DSC canister, except that the internal basket was replaced with a 5/8-inch thick radial liner and basket to hold GTCC waste containers. There were 14 GTCC waste containers. Seven were 85.5" long and 7 were 75.5" long. The GTCC waste containers were approximately 16" square, constructed of a ¼-inch stainless steel plate with lifting bars in the top lid for handling. The shorter GTCC waste containers weighed approximately 475 pounds. The longer weighed approximately 535 pounds. The loading pattern considered the center of gravity related to the GTCC waste container locations. The longer containers were placed on the bottom of the GTCC-DSC and the shorter and lighter containers were placed on the top. The total weight of the loaded GTCC-DSC was 76,526 pounds. This was less than the maximum weight for a loaded 24PT1-DSC of 82,000 pounds.

Engineering Change Package 031001485-1, "Movement of Unit 1 Spent Fuel and GTCC from Unit 1 to ISFSI," performed an evaluation of placing the GTCC in the canister and storing it at the ISFSI. The evaluation included a review of the shielding, confinement, structural, thermal, pressure, and seismic loads that bound the 24PT1-DSC design. Calculations for normal, off-normal, and accident conditions were performed. By being of the same design as the 24PT1-DSC, the canister was compatible for use with the AHSMs, the OS-197 transfer cask, and the MP-187 transportation cask. The analysis found the GTCC would not have any adverse effects on the spent fuel stored in the adjacent AHSMs at the ISFSI or increase the dose to the public. A 10 CFR 72.48 and 10 CFR 50.59 evaluation was performed to include the GTCC in the ISFSI, and Revision 2 of the licensee's 10 CFR 72.212 evaluation was issued incorporating the GTCC canister information.

The GTCC canister was loaded in accordance with Procedure SO1-X-9, "Dry Cask Storage Loading," Revision 0. The canister was dried, backfilled with helium, and welded closed using the same equipment, processes, and procedures used for the 24PT1-DSC. Radiological surveys performed during the loading campaign on September 10, 2004, measured 200 mR/hr gamma contact dose on the side of the transfer cask and 100 mR/hr at 30 cm. After insertion into the AHSM and installation of the door, the dose rate was 80 μR/hr contact dose with the door at approximately 7 feet high. No neutron dose was measured or expected from the activated and irradiated metal. Post storage operations applied to the AHSM containing the GTCC-DSC used the same procedures for storage inspections and temperature monitoring as used for the other AHSMs containing 24PT1-DSC canisters.

j. <u>Earthquake/Tsunami Potential for the SONGS ISFSI Operations</u>

The recent earthquake and tsunami in Sendai, Japan, highlighted this unique risk to the nuclear facilities along the California coastline. The SONGS site is located near the San Andreas fault and is at risk of an earthquake. The seismic design of the ISFSI was reviewed as part of the NRC inspection of the original ISFSI construction activities and was included in NRC Inspection Report 72-41/2002-01 (ML021410532), dated May 21, 2002. Section 5 of that report discussed the design features of the first ISFSI pad in relationship to the safe shutdown earthquake postulated for the SONGS site. The seismic design of the Advanced NUHOMS storage system exceeded the postulated earthquake conditions that could occur at the SONGS site. Calculations demonstrated that the canisters and storage modules were designed to withstand the postulated safe shutdown earthquake that had been used for the design of the reactor facilities. The seismic information included in the 2002 inspection report was still current relative to a postulated earthquake. Since the Japan earthquake, numerous evaluations were underway to reconfirm the seismic parameters that are currently applied to the SONGS site. Since the first NRC inspection, an additional ISFSI pad had been constructed. This pad was adjacent to the first pad and supported a double row of AHSMs. The pad was designed using the same seismic criteria as the original pad. A new soil structure interaction analysis was performed to evaluate the seismic response of the larger pad area and the back-to-back AHSMs. Construction of the pad was completed as Quality Class IV under the quality assurance plan. The first pad had been constructed as Quality Class II.

The tsunami potential for the ISFSI was evaluated and documented in NRC Inspection Report 72-41/03-01 (ML040060764, ML040060771, and ML040070255), dated January 6, 2004. Tsunami information was included in Attachment 2, "Inspector Notes," under the category General License and the topic Environmental Extremes (page 15-17 of 70). The affects of a tsunami reported in the 2004 NRC inspection reports was still considered current. The tsunami was considered bounded by the flood analysis for the ISFSI. The ISFSI was located 19.75 feet above sea level. A flooding condition was assumed to reach elevation 29 feet, resulting in 9 feet of water on the pad. This was less than the 50 feet of water evaluated in Section 2.2.2 of the FSAR for the design basis flood. The maximum tsunami, which included the storm height of the waves, was 27 feet. As such, the ISFSI pad could be 7 feet underwater. The height of an AHSM was approximately 18 1/2 feet above the pad level (i.e., 38 feet above sea level). The outlet vents were on the top of the AHSM. The inlet vents were approximately 2 feet above the ISFSI pad. The SONGS site had a sea wall between the ocean and the ISFSI which was 28 feet in height. The licensee did not take credit for the sea wall and assumed it could fail during a tsunami.

On March 24, 2011, the California Coastal Commission issued a report entitled "The Tohoku Earthquake of March 11, 2011: A Preliminary Report on Implications for Coastal California." The report evaluated the earthquake and tsunami potential effects for the nuclear plants on the California coast. The evaluation relative to the SONGS site determined that strong ground motion and a massive tsunami similar to that which occurred near Japan could not be generated by the faults near the San Onofre Nuclear

Generating Station. The San Andreas fault could not produce a magnitude 9 earthquake, as compared to the Cascadia subduction zone in northern California, Oregon, and Washington, which was susceptible to an earthquake and tsunami similar to that experienced in Japan. The 2011 report noted that, since the original California Coastal Commission review of the SONGS ISFSI in 2001, additional information related to the potential for a tsunami, including one generated by an underwater landslide, indicated that the ISFSI was in the tsunami inundation zone. The tsunami inundation line calculated for this area was at an elevation of approximately 20 feet mean sea level. The ISFSI is located at 19.75 feet.

As a result of the renewed interest in the affects of an earthquake and tsunami at the SONGS site, several discussions were held with the licensee concerning their earthquake and tsunami preparedness. For a tsunami, the licensee believed the design of the ISFSI provided adequate protection from the impact of a large wave. The AHSMs were designed to withstand a flood surge and had been evaluated in the FSAR for submersion under 50 feet of water. The submersion actually contributed to providing additional cooling of the canisters. Because of the large weight of the concrete AHSMs containing the canisters, which weighed several tons, and due to the inter-connecting of the AHSMs with adjacent AHSMs, the effects of a large wave breaching the seawall and flooding the ISFSI was not considered a threat to moving the AHSMs from their location.

Concerning an earthquake, the ISFSI pad design met the current seismic criteria for the site and had used conservative assumptions. The AHSMs were designed by the vendor to provide additional safety margin for sites prone to earthquakes. As such, the potential for damage to the spent fuel stored at the ISFSI had adequately considered the potential for an earthquake at SONGS.

When the canister was in the spent fuel pool for loading fuel, it was physically separated from the fuel racks by a reinforced concrete wall. When the canister loaded with spent fuel was placed in the cask washdown area inside the plant, it was seismically restrained to prevent tipover or movement during an earthquake when not connected to the crane hook. The seismic restraints were required by Step 8 of Technical Specification 4.4.3, “Site Specific Parameters and Analysis.” These seismic restraints were designed to account for the 1.2 g horizontal acceleration that could be experienced at the cask washdown pit based on the SONGS design basis earthquake peak horizontal ground acceleration of 0.67 g. Calculation C-296-03.04, “Unit 2 & 3 Seismic Restraint for 110-Ton Crane,” documented the calculations. The separation feature in the spent fuel pool, the use of a single failure proof crane, and the use of seismic restraints on the cask while in the washdown area reduced the potential for an earthquake causing a significant problem during a loading campaign, since these provisions encompassed the majority of the time frames when the canister was in the plant being loaded and sealed.

The licensee had incorporated instructions for conducting post accident inspections to determine if any damage occurred after various emergency situations. Procedure SO123-X-9.11, “Post-Accident Inspection for Dry Fuel Storage,” Revision 3, provided actions and methods for recovery following various postulated accidents. Events discussed in the procedure included earthquakes, tornado, flooding, fire/explosion, lightning, burial of the AHSMs due to an earthquake, extreme ambient temperatures, blockage of the AHSM air inlet and outlet openings, a jammed canister during insertion into an AHSM, and dropping a cask. For blockage of the vent openings, the required time frames of 40 hours for the 24PT1-DSC and 25 hours for the 24PT4-DSC from Technical Specification 5.2.5 were included in Step 4.0, "Precautions." For an earthquake, Section 6.1, "Earthquake," of the procedure required an inspection of the AHSMs for structural damage, inspection of the AHSM vents for blockage, and if damage or movement of the AHSM had occurred, initiation of a radiation survey and taking of corrective actions if radiation levels had changed, such as new postings or use of temporary shielding. For flooding, Section 6.4 "Flood," required all AHSMs to be inspected for damage or vent blockage. Blockage was to be removed. For an event that caused earthen material to bury the AHSMs, Section 6.8, "Burial," required clearance of the blockage to restore ventilation flow to the AHSMs and inspection for damage.

For events that could affect the crane, Procedure SO2-I-3.32, "Unit 2 Cask Handling Crane Checkout and Operations," Revision 8, and Procedure SO3-I-3.32, "Unit 3 Cask Handling Crane Checkout and Operations," Revision 8, included Section 6.8, "Post-Seismic Inspection," which directed maintenance engineering to perform a visual inspection of the crane after a seismic event. The visual inspection included the hydraulics, lubrication systems, welds on the bridge, and main trolley and limit switches. If the crane was not operational, Procedures SO2-I-3.32 and SO3-I-3.32 included Attachment 4, "Emergency Load Lowering/Bridge and Trolley Emergency Instructions," which provided direction on how to manually move the bridge and trolley and lower a load suspended on the crane.

During cask loading operations, if an emergency condition occurred, personnel were trained to stop work and place the canister in a safe configuration. During discussions with various personnel, the definition of what constituted a safe configuration if a large earthquake occurred had different answers. For the most part, personnel had considered an earthquake as movement felt onsite after which work was re-initiated once everything was verified as not damaged. When considering the Japan earthquake and the damage that resulted, consideration that things may not get back to normal immediately after the earthquake needed to be considered. These considerations included an extended period of no onsite power leaving the cask suspended on the hook, the need to divert ISFSI personnel to more important response efforts occurring onsite, and the length of time the canister could be left unattended while the time-to-boil clock continued. If the canister was in the washdown area, should it be returned to the spent fuel pool because higher priorities would prevent resumption of cask activities in the near future? Would power be available to operate the crane to return the canister to the pool? These types of discussions generated enough open questions among the licensee's staff that a condition report (NN 201428976, "Evaluate Response to Seismic Event") was generated to systematically address the various situations and make a determination as to the most prudent actions to take. Plans were to strengthen the ISFSI loading procedures to include more contingency plans for a serious earthquake situation. In addition to this effort, the site was conducting an evaluation in response to the Institute of Nuclear Power Operations (INPO) Event Report "Fukushima Daiichi

Nuclear Station Damage Caused by Earthquake and Tsunami," Issued March 15, 2011. The INPO report contained numerous recommendations for individual plant owners to verify the capability to mitigate conditions that could result from beyond-design-basis events using severe management guidelines. This included verifying that required equipment needed to mitigate a severe accident would be operational if called upon and was properly staged and protected. The NRC resident inspector's office was monitoring the licensee's effort to evaluate the site in response to the INPO recommendations.

k. Over-Pressurization of Canister DSC 41

On June 24, 2010, the licensee had completed the welding of the inner top lid on Canister DSC 41 (15.3 kW) and had connected the hoses to begin the blow-down of the canister. The blow-down process removes the water from the canister by injecting helium gas through the vent port to force the water out of the canister through the siphon (drain) port. The water is directed into the spent fuel pool through a drain line. Then the canister is vacuum dried, after which the canister is backfilled with helium, an outer lid welded on, and the canister placed in the ISFSI.

During welding of the inner top lid, a male Swagelok stem fitting was installed into the female vent port Swagelok fitting. This opened the internal shutoff valve inside the female Swagelok fitting and provided for venting of the canister. The siphon port was closed by having no connections made to its Swagelok fitting and was left closed during welding. The siphon port was maintained closed during welding with no hoses connected to prevent any potential for unexpected draining of the water from the canister due to a siphoning action. When welding was completed, the nuclear service water line was connected to the siphon port Swagelok fitting and the canister was refilled with water to replace the approximately 53 gallons removed prior to welding. The 53 gallons was removed prior to welding to ensure the inner lid was not in contact with the water which would cause a heat sink on the lid during welding. The canister was refilled after welding to remove all the air out of the canister before the start of blow-down with the helium to ensure no air could come into contact with the fuel during blow-down. For Canisters DSC 01 through 31, the 53 gallons had not been refilled into the canister after welding and the bubble remained in the top of the canister. But starting with Canister DSC 32, the license began refilling the canister prior to blow-down to make sure no air could be in the canister during the blow-down process. This was to address concerns expressed by the NRC (generically to the industry) that licensees should make sure the fuel does not come into contact with air during the loading and sealing process of the canisters.

After the canister was filled with water, the service water line was disconnected from the siphon port Swagelok fitting. This resulted in the inner valve in the siphon port Swagelok fitting to close. The male Swagelok stem fitting was then removed from the vent port. This allowed the vent port's Swagelok fitting internal shutoff valve to close. At this point, the canister was now isolated. Procedure SO23-I-30.9, Revision 12, Step 6.9.2, provided a caution that "The DSC is now unvented. Proceed through the following steps without delay to minimize DSC heat-up." This occurred at 12:30 p.m. for Canister DSC 41.

The siphon port Swagelok fitting was replaced with a quick connect fitting that was connected to a discharge hose routed to the spent fuel pool. The discharge hose had a drain isolation valve that was closed. The vent port was connected to the vacuum drying system. The vacuum drying system included a 50 psig pressure relief valve. The vent port ball valve located a few inches down-line from the vent port was closed per Step 6.9.7 of Procedure SO23-I-30.9, Revision 12, so the canister was still isolated at this point from any pressure relief valves. The line to the helium bottles was then connected to the vacuum drying system. This line had an 18-psig relief valve located near the helium bottle rack. The valve on the helium line was opened and the line was charged to a maximum of 17 psig. For Canister DSC 41, the line was charged to 13 psig. This charged the line all the way to the closed vent port ball valve located near the vent port on the canister. This charged line included the 50-psig relief valve on the vacuum drying system. The canister was still isolated at this point. It should be noted that previous procedures to Revision 12 had the operator open the vent port ball valve in Step 6.9.7 prior to connecting the helium line to the vacuum drying system. This provided a path from the canister to the 50-psig relief valve on the vacuum drying system. This change in Revision 12 had been made because of concerns that water could get from the canister into the helium fill lines before the helium pressure (max 17 psig) was applied to the lines. Canister DSC 41 was the first to use Revision 12 of the procedure. So prior to Canister DSC 41, the canisters had access to the vacuum drying system, which included the 50-psig relief valve, slightly earlier than with Canister DSC 41.

The process to connect the lines to the vent and siphon ports and turn on the helium was stated by the craft workers and their shift supervisor as requiring a couple of minutes. During the time the canister was isolated, pressure built up in the canister. The drain isolation valve on the siphon port drain line and the vent line ball valve were opened at approximately 12:32 p.m., providing a path to the helium bottles through the vacuum drying system and a drain path to the spent fuel pool through the siphon port. No water was observed being discharged through the drain line from the siphon port, but the 18-psig relief valve near the helium bottle rack lifted, resulting in water being released from the relief valve. Water was sprayed on the wall and a small puddle formed on the floor. The vent port ball valve, then the helium supply line valve, was closed to stop the discharge. The siphon port drain line to the spent fuel pool was left open. The water was cleaned up using a couple of rags and was cool to the touch. The 50-psig relief valve on the vacuum drying system did not open.

The licensee entered Notification NN 200984377 into their corrective action program on June 24, 2010, to document the event in their corrective action program. On June 26, 2010, calculations were completed by SONGS design engineers to bound the event as to the maximum pressure that could have developed inside the canister. Calculations determined that the pressure could have risen to 181 psig based on a 23-minute isolation period. The 23 minutes had been determined by the project manager after looking at the log and the procedures. The time was taken from the initial closing of the two ports (vent and siphon) at 12:30 p.m. to the time the successful helium blow-down was started at 12:53 p.m. However, 2 minutes after the initial isolation occurred at

12:30 p.m., the craft personnel had installed and opened the drain line connected to the siphon port. This vented the canister and was left open throughout the activities that occurred next prior to the successful canister blow-down. So the 23 minutes should have been approximately 2 minutes for the calculation. The calculation that determined the 181 psig was a very simplistic calculation and did not consider heat transfer from the water in the canister to the canister metal shell or heat transfer to the atmosphere. The starting assumption for the calculations also assumed the water temperature at the beginning of the isolation was 200ºF. However, when the dye penetrant test was conducted of the weld after the event, the temperature on the weld surface was 130ºF. By assuming a higher starting temperature, a higher water expansion coefficient is assumed instead of the coefficient associated with the lower temperature of the water. Based on 2 minutes instead of 23 minutes, the 181-psig value can be reduced by a factor of 10, resulting in an internal pressure more in the range of 18 psig, well below the 100 psig design basis accident pressure analyzed in the NUHOMS FSAR (see Section A.2.1.1, Table A.3.1-4, and Table A.4.4-10). The calculation also estimated that the heat build-up increased the water volume in the canister by 1 gallon (page 8 of 35 of NN 200984377). This was not consistent with the observations made by the craft workers concerning the small amount of water released from the 18-psig relief valve.

After the event, the licensee performed a visual and a dye penetrant examination of the final weld on the inner lid cover. The evaluation found no indications in the weld, indicating no stresses had been applied to the weld causing cracking. The lid was also found to have maintained its flat shape with no bulging in the middle, which would have interfered with the placement and welding of the outer lid. The inner lid weld was a partial penetration weld and would have been most susceptible to damage from an over-pressurization event, since the bottom welds and seam welds are full penetration welds. Areva Transnuclear performed a calculation of the stress on the canister based on a 50-psig pressure and a 300ºF water temperature. This calculation was documented as TN Calculation SCE23-0213, Revision 0, issued June 30, 2010. The 50-psig value was used because the 50-psig pressure relief valve did not open on the vacuum drying system, thereby establishing an upper bound for the pressure. The FSAR had analyzed for a maximum of 20-psig pressure during the blow-down phase (FSAR Table A.4.4-10). The TN calculations for the 50-psig pressure found the canister component stresses and the weld stresses to be below the allowable Areva Transnuclear design criteria stresses and the NB-3226 testing limits.

After the incident, the licensee recognized that closing both the siphon and vent line represented an unanalyzed condition in the FSAR and entered Notification NN 200984377 into the corrective action program on June 24, 2010. The “extent of condition” section in NN 200984377 discussed FSAR Section A 4.7.3.2, which stated “The cavity is always vented during the water heat-up transient.” Notification NN 200995333 was entered into the corrective action program on July 1, 2010, and documented an evaluation of Procedure SO23-I-30.9 to determine every step where the vent and siphon ports were closed in Revision 12 of the procedure. The licensee revised the procedure (effective revision date July 3, 2010) to modify any steps that closed both ports at the same time. The new Revision 13 had no steps that would close both ports on the canister at the same time while water was in the canister.

There is no accident analysis in the FSAR that evaluates having both the vent and siphon port closed at the same time and the canister heating up and over-pressurizing. In fact, FSAR Section A.4.7.3 states "Prevention of boiling in the Advanced NUHOMS System is not required to ensure public health and safety for the following reasons . . . (2) the cavity is always vented during the heat-up transient." Clearly, this requirement in the FSAR was not fulfilled prior to the blow-down of Canister DSC 41 by closing both the vent and siphon lines while the connections were being made.

Revision 11 of Procedure SO23-I-30.9, Step 6.9.7, ensured that the vent port valve was opened as soon as it was installed. The vent port valve connected to the vacuum drying system which contained the 50-psig relief valve. Revision 12 changed this step to close the vent port valve after installation, thereby isolating the canister since the siphon port valve had been closed previously. Revision 12, effective date June 23, 2010, was first used with Canister DSC 41. The licensee conducted the required 10 CFR 72.48 safety evaluation of the procedure change from Revision 11 to Revision 12. However, the review did not recognize the significance of the change made in Step 6.9.7 in relation to the statement in the FSAR and approved the revision.

The requirement in 10 CFR 72.48(c)(1) related to making changes in the licensee's programs states, in part "A licensee may make changes to procedures described in the FSAR if the change does not meet any of the criteria in Section C.2." Section C.2 states "A general licensee shall request that the certificate holder obtain a Certificate of Compliance amendment prior to implementing a proposed change, test, or experiment if the change, test, or experiment would . . . (v) create the possibility for an accident of a different type than any previously evaluated in the FSAR." Contrary to this, closure of the vent port valve, as specified in Revision 12 of Procedure SO23-I-30.9, resulted in the isolation of the canister while filled with water, contrary to the statement in FSAR Section A.4.7.3 which states that the cavity is always vented during the water heat-up transient. As such, no accident analysis is included in the FSAR for an over-pressurization situation when both the vent and port valves are isolated while the canister is filled with water. This Severity Level IV violation of 10 CFR 72.48(c)(1) meets the criteria in the Enforcement Policy, Section 2.3.2, as a noncited violation. The issue was licensee identified, entered into their corrective action program, non-repetitive and not willful, and compliance was restored with the issuance of Revision 13 of Procedure SO23-I-30.9 which revised Section 6.9 to have at least one valve open at all times. Revision 13 of the procedure was effective July 3, 2010, prior to loading the next canister.

This event did not meet the reportability requirements of 10 CFR 72.75. The over-pressurization of the canister was bounded by the design limits for an accident in Section A 2.1.1 of 100 psig. The relief valves and associated equipment were not safety related and the cask stress limits were not exceeded. There were no reportability requirements in 10 CFR Part 72 related to a licensee violating requirements in the FSAR unless an emergency is declared, a press release is issued, a contaminated person is transported to an offsite medical facility for treatment, a defect is found in an important-to-safety structure, system, or component, or a "significant" reduction in the effectiveness of the storage confinement system occurs.

The issue related to the over-pressurization of Canister DSC 41 was reported to the NRC Resident Inspector at the SONGS facility by the licensee over the June 26-27, 2010, weekend. The following week, the information was forwarded to the NRC Region IV office. On November 3, 2010, a Region IV inspector met with the licensee personnel involved with the loading and sealing of Canister DSC 41 and reviewed the various documents associated with the incident. Additional information related to the incident was provided during a public meeting held by the NRC on December 14, 2010.

l. <u>Aging Management Program</u>

The SONGS ISFSI had been in operation less than 8 years. The facility was in very good condition with no signs of deterioration, despite being exposed to a marine environment. SONGS had selected the advanced horizontal storage system from Areva Transnuclear because it provided several advantages over the standard design, including a thicker concrete storage module, thick concrete doors, supplemental concrete shield walls, and a five foot thick roof. SONGS also added additional thickness requirements to the steel canister to improve weathering in the marine environment. A special report was issued by Structural Integrity Associates, Inc. in June 2002 entitled "Suitability of Materials for a Dry Fuel Storage Facility for Extended Service in a Marine Atmosphere Environment." The report evaluated the potential life expectancy of the canisters at SONGS and concluded that corrosion would not be a problem over an extended period. The report looked at the corrosion performance of stainless steel in a severe marine environment and considered general corrosion, pitting corrosion, crevice corrosion, stress corrosion cracking, and microbiologically influenced corrosion. The effects of radiation at levels greater than 10,000 R/hr from the stored fuel on the canister was also included.

The review included data from Stainless Steel Type 304 and Type 316 samples that had been archived and isolated in storage for 26 years at Kure Beach, North Carolina, and a 15-year marine atmospheric exposure study on Types 304 and 316 stainless steel in Okinawa, Japan, and Chiba, Japan. Welded specimens were included in these samples. The general corrosion rate was estimated in the report to be less than 0.1 mil (0.0001 inches) in 100 years. The SONGS canisters were constructed to a minimum 0.61" (610 mils) wall thickness. Pitting corrosion rates for the Type 316 stainless steel was estimated to range from 5.9 mils to 10.1 mils over a 100-year period. For the SONGS canisters sheltered inside the AHSMs, the pitting would be expected to be at the lower end.

m. <u>Records</u>

Specific records were required by 10 CFR 72.234(d) related to each canister placed in service. Canister DSC 42 was selected to verify that the licensee could produce the required records. Two main documents contained the required information. The first was the standard letter submitted to the NRC as required by 10 CFR 72.212(b)(1)(ii), which is submitted to the NRC within 30 days of using the canister to store spent fuel. For Canister DSC 42, this letter was dated July 26, 2010 (ML102090124). Canister DSC 42 had been placed on the ISFSI pad on July 13, 2010. The second document was the Certificate of Compliance provided to the licensee from Areva Transnuclear. This letter documented the fabrication dates, certified that the canister was designed, fabricated, tested and repaired in accordance with a quality assurance program accepted by the NRC, and certified that the inspection required by 10 CFR 72.236(j) had been performed and found satisfactory.

1.3 Conclusions

The ISFSI concrete storage modules and pad were being maintained in good physical condition, with no deterioration or cracking observed. There were 55 storage modules located on the pad, with 41 storing spent fuel and one storing GTCC. The next loading campaign this summer will load up to 6 more storage modules.

Spent fuel from all three reactor facilities was loaded in the ISFSI. All Unit 1 spent fuel stored at the SONGS site has been loaded into 17 canisters. Twelve canisters have been loaded with Unit 2 spent fuel and 12 have been loaded with Unit 3 spent fuel. Technical specification requirements related to maximum burn-up, enrichment, heat load, empty slots, and failed fuel were reviewed against records for selected canisters. All Technical Specification requirements had been met.

Records for the environmental radiological monitoring program were reviewed for the period of 2006 through 2010. TLDs placed around the ISFSI pad typically measured exposure levels at background or slightly higher, unless nearby radioactive material influenced the readings. The most significant influence was the storage of the Unit 2 steam generators near the northern side of the ISFSI.

Radiological surveys and contamination surveys of the ISFSI pad continued to confirm no contamination at the ISFSI and radiation levels were low. Radiation levels from the canisters during handling and welding were well within the Technical Specification limits. Accumulated doses to workers to load each canister were continuing to improve as good ALARA practices were implemented.

The training program met the requirements of Technical Specification 5.2.2. Training included the requirements in the license, ISFSI operations, and procedures used for ISFSI activities. Job-specific training, consisting mostly of hands-on training under the direction of a qualified individual, was required prior to an individual working independently.

Temperature monitoring of the AHSMs was being performed in accordance with Technical Specification 5.2.5. Temperature limits had been established for the temperature monitoring system being used by the licensee consistent with the FSAR requirements. No blockage of air vents had been found during tours of the ISFSI. Periodic alarms due to malfunctions, ambient temperature changes, or the insertion of a new canister in an adjacent AHSM were effectively evaluated, resolved, and entered into the corrective action program.

The licensee was implementing their Part 50 quality assurance program for the ISFSI. The quality assurance program had been adequately incorporated into the ISFSI procedures with appropriate control points and sign-offs. However, the quality assurance program had implemented an auditing program that only reviewed the ISFSI programs in a generic fashion.

The licensee was actively documenting problems related to the ISFSI in the site corrective action program. Selected condition reports were reviewed and found to be properly dispositioned. An in-depth review was performed on one selected condition related to a rag found in the siphon line of a canister. The issue was properly dispositioned and changes made in the foreign material control program at the fabrication shop.

One canister of Greater-Than-Class-C (GTCC) waste was stored at the ISFSI. The canister contained irradiated and activated metal from the Unit 1 reactor internals. GTCC is allowed for storage at an ISFSI if the licensee verifies that the GTCC meets the storage requirements of the Part 72 license. Calculations were performed and a revision to the 10 CFR 72.212 evaluation was completed to verify compliance.

The ISFSI had been evaluated for postulated earthquakes and tsunamis that could affect the site. A recent California Coastal Commission report completed a preliminary review of the reactor facilities on the California coast. The report concluded that the San Andreas fault would not produce the same conditions as occurred in Sendai, Japan, in March 2011. However, the licensee was re-evaluating their site preparedness for a major event. The ISFSI pad and storage modules were designed for the postulated earthquake that could occur from a nearby fault. For a tsunami, the estimated water height that could be experienced on the ISFSI pad was 7 feet. This was less than the 50-foot flood level that the AHSMs could withstand based on possible flooding effects analyzed in the FSAR.

The FSAR does not include an accident analysis for a situation where the canister is filled with water after the lid is welded in place, with the vent and siphon port valves closed, resulting in a pressure build-up in the canister. The FSAR does not evaluate this accident because it states in Section A.4.7.3.2 that the canister is always vented during the heat-up transient. Contrary to this, while loading Canister DSC 41 in June 2010, the licensee briefly closed both valves, allowing the canister to build-up pressure. This action was performed in accordance with a revision of the procedure that directed the workers to close both valves. The 10 CFR 72.48 evaluation of the procedure revision did not recognize that the change was inconsistent with the FSAR and should not have been approved. Consequently, an NCV was identified for the failure to perform an adequate 72.48 evaluation of the procedure change.

The licensee had not implemented an aging management program at the site since the ISFSI had become operational only 8 years ago. However, the features of the advanced horizontal storage module system and the thicker canister design used at SONGS provided additional protection of the spent fuel from the marine environment. A study conducted in 2002 found that corrosion of the stainless steel canisters would not create storage problems for a considerable time into the future.

Records required by 10 CFR 72.234(d) related to each canister placed in service were verified for Canister DSC 42 to verify the required records could be retrieved and provided for inspection. The required records were found to include the required information as specified by 10 CFR 72.234(d)

2 Review of 10 CFR 72.48 Evaluations (60857)

2.1 Inspection Scope

The licensee is required to follow the process defined in 10 CFR 72.48 when making changes, under certain conditions, to their ISFSI and procedures as described in the FSAR. A selected sampling of documents was reviewed to verify the licensee was appropriately using the required process and performing adequate safety reviews.

2.2 Observations and Findings

The 10 CFR 72.48 process was being implemented by the licensee using Procedure SO123-XV-44, "10 CFR 50.59 and 72.48 Program," Revision12, and Procedure SO123-XV-44.2 ,"10 CFR 72.48 Program Implementation Guidelines," Revision 1. Since SONGS was a general licensee, changes that involved the design of the ISFSI were sent to Areva Transnuclear for the 72.48 evaluation. The licensee was maintaining a list of qualified reviewers. Thirteen individuals were listed as currently qualified.

Several 72.48 screenings were reviewed. All of the screenings and evaluations listed below were performed by Areva Transnuclear for SONGS.

Screening LR 721029-327 reviewed the storage of four spent fuel assemblies that contained small quantities of rust that had come from the reactor vessel flange. The screening determined that an evaluation was required. Areva Transnuclear performed the evaluation and found that storage in the canisters used at SONGS of spent fuel assemblies containing small amounts of rust was acceptable.

Screening LR 721004-839 reviewed the issue of several small pits found on the underside of the bottom support ring of the OS197-3 transport cask. The screening was initiated as a result of Non-Conformance Report 2010-074. The screening determined that the pitting did not impact the function of the cask.

Screening LR 721029-332 evaluated the over-pressurization incident related to Canister DSC 41. This screening was initiated as a result of the licensee's Condition Report NN 200984377. The screening determined that an evaluation of the potential over-pressurization on the condition and function of the canister was required. Areva Transnuclear's evaluation found that the potential over-pressurization to a maximum limit of 50 psig had not damaged the canister. This issue is discussed further in Section 1.2.k of this inspection report.

Screening LR 721029-295 evaluated the storage implications of leaving a 6-inch long polystyrene bristle on Assembly S2F234. During inspections of the spent fuel prior to placement into a canister, debris was found on five fuel assemblies. The debris was successfully removed from four of the assemblies; however, the debris on the fifth assembly could not be removed. Spectrum analysis was performed on similar debris collected during the inspection. The screening determined that an evaluation was necessary. The evaluation reviewed several possible scenarios that could occur with the bristle inside the canister during loading and long-term storage. The evaluation determined that there would be no adverse effects due to the presence of the bristle on the spent fuel assembly.

Screening 721004-460 evaluated damage to the OS 197 transfer cask during the insertion of Canister DSC 19. During the insertion, the canister was not properly aligned with the AHSM opening and rails. As a result, the ram inserting the canister reached 1350-psig pressure, which was the procedural limit during insertion. The canister was retracted back into the transfer cask. The transfer cask was realigned, raising it approximately 1/8 inch. The canister was then successfully inserted. Upon examination of the interior of the transfer cask after it was backed away from the AHSM, scratches were observed on the inside of the transfer cask. In addition, a small chunk of metal was found. Three issues resulted from this incident: (1) the design basis pressure for the ram during insertion was exceeded, (2) the inside of the transfer cask was damaged, and (3) a chunk of metal was found inside the transfer cask. These issues were documented and evaluated in several documents reviewed during this inspection, including SONGS Action Request 070300343, originated on March 7, 2007, Areva Transnuclear Supplier Non-Conformance Evaluation 2008-027, dated August 26, 2008, Areva Transnuclear 10 CFR 72.48 Screening and Evaluation LR-721004-460, dated March 12, 2007, Areva Transnuclear Supplier Non-Conformance Evaluation SNR AR-M450, dated March 12, 2007, and Areva Transnuclear 10 CFR 72.48 Screening and Evaluation LR-721029-227, Revision 1.

During the insertion of Canister DSC 19 on March 6, 2007, the licensee experienced difficulty and stopped the insertion when the hydraulic pump driving the ram that pushed the canister out of the transfer cask and into the AHSM storage module reached the 1350-psig limit specified in the AHSM loading procedure. The canister was stopped approximately 79 inches short of being fully inserted, then retracted back into the transfer cask. The transfer cask was raised approximately 1/8 inch and the canister successfully inserted into the AHSM. Areva Transnuclear performed an evaluation of the stresses on the bottom of the canister and on the hydraulic ram due to reaching the

1350 psig limit allowed in the licensee's procedures. Areva Transnuclear's 10 CFR 72.48 Screening and Evaluation LR-721029-227, Revision 1, determined that the force exerted during the event reached 155 kilopound-force (kips). However, the design basis force that had been evaluated for the ram was 80 kips for a jammed canister event. The 80 kips equated to 720 psig. Analysis in accordance with ASME NB-3228 was performed to determine the effect of this higher force on the bottom end of Canister DSC 19. Since the canister had been successfully retracted after the initial attempt to insert into the AHSM, there was no evidence of excessive deformation on the outer bottom cover plate of the canister. The analysis found the 155 kips ram loads were below the normal and off-normal ASME service level A and B stress limits. As such, the overload had no affect on the structural or confinement function of the canister.

Procedure SO23-I-30.9, "24PT4-DSC Dry Cask Storage Loading," Revision 14, is the current procedure for operating the ram during insertion of the canister into the AHSM. Section 6.19, "DSC Insertion and Ram Withdrawal," of the procedure provided directions for operating the ram. The ram was an extension arm that had three stages (segments). Each successive staged required a higher pressure for movement. The first stage moved the canister into the AHSM the first 80 inches. Step 6.19.15 limited the ram pressure to 398 psig. The second stage moved the canister from 80 to 160 inches. Step 6.19.15.5 limited this stage to 707 psig. The third and final stage moved the canister from 160 inches to final insertion at 210 inches. This stage of the cylinder stroke limited the ram pressure to 1591 psig. The limit for the second stage at the time of insertion of Canister DSC 19 had been 1350 psig. This pressure was identified as too high for this stage (extension) of the ram arm and should have been limited by the 720 psig. This is when the canister had been 79 inches from insertion (i.e., about 130 inches) into the canister. The current procedure in use at the time of this inspection was found to limit the arm in the second stage to 707 psig and met the 80 kips limit of force on the bottom of the canister. The ram is a not-important-to-safety component as listed in FSAR Table 2.5-1 "Advanced NUHOMS System Major Components and Safety Classification."

The gouging of the inside of the transfer cask was evaluated in several of the documents reviewed, including Areva Transnuclear Supplier Non-Conformance Evaluation SNR AR-M450 and Areva Transnuclear Supplier Non-Conformance Evaluation 2008-027. After Canister DSC 19 was inserted into the AHSM and the transfer cask moved back, an examination was performed of the inside of the transfer cask. Five new scratches on the inside of the transfer cask were found. Of these, three were very minor and would not impact the minimum wall thickness requirement of 0.45 inches in Areva Transnuclear Drawing NUH-06-8002, Revision 6, Sheet 2, Item 2 (Inner Liner Plate), as shown for Zone B-5. One of the scratches was in the weld connecting the liner to the top forging and was not subject to minimum wall thickness requirements. However, one gouged area was 1 ¾ inches long, 1 inch wide and 0.12 inch deep and was determined to be of sufficient depth to require evaluation. Ultrasonic testing was performed to obtain an accurate depth measurement. The cask wall thickness was 0.459 inches. The gouged area was 0.120 inch deep. This resulted in a wall thickness of 0.339 inches at the point of the gouge, which was below the 0.45 inch minimum wall thickness requirement shown on the drawing. For the 125 ton capacity OS197-3 transfer cask used by SONGS, the wall thickness requirement was calculated in Areva Transnuclear Calculation NUH-06-0212, Revision 4. Table 8-7 showed the calculated minimal wall thickness requirement as 0.439 inches for the drop accident and 0.249 inch for normal and off-normal loading conditions. The gouged area met the minimum wall thickness for normal and off-normal events, but did not meet the requirement for the drop accident. Areva Transnuclear evaluated the gouged area in accordance with ASME NC-3217(c), which provided a basis for isolated areas to exceed normal stress levels. Areva Transnuclear determined that the ASME Code NC-3217 Service Level D calculated stress intensities associated with the postulated load drop accident loads only required evaluation of the primary and primary plus bending stresses. The stresses associated with the gouged area were classified as secondary, or Q stresses and did not require evaluation. As such, there was no affect upon the structural strength of the inner liner and the transfer cask could be used as-is.

The small metal piece found inside the transfer cask appeared to be rolled sheets of metal formed into a wedge shape. Action Request 070300343 and Areva Transnuclear Supplier Nonconformance Evaluation 2008-027 provided information related to the metal found. Final Safety Analysis Report, Revision 2, was in effect at the time of construction of Canister DSC 19. Revision 2 of the FSAR, Table A.1.2-1, "Key Parameters of the Advanced NUHOMS System Components," listed the minimum shell thickness for the 24PT4-DSC canister as 0.53 inch. Section A.3.6.1 of the FSAR listed the "nominal" plate thickness for the cylinder shell as 0.625 inch. The stress analysis conservatively assumed the minimum plate thickness of 0.53 inch. Drawing ANUH-01-4001 in the NUHOMS FSAR showed the minimum wall thickness of the shell as 0.53 inch. The canisters being fabricated at SONGS were thicker than the typical NUHOMS canisters to account for the marine environment and to provide extra margin for potential aging effects that could occur at the site. The SONGS canisters were fabricated to a minimum wall thickness of 0.61 inch. The measured minimum wall thickness for Canister DSC 19 was 0.666 inch determined during fabrication.

The metal sample was sent to an independent laboratory for analysis. The sample was determined to be Type 316 stainless steel, which is the construction material for the canister, with a small amount of Type 304 stainless steel, which is the construction material for the rails inside the AHSM. The laboratory concluded that the sample consisted of multiple thin layers scraped from the canister shell by the structural rails of the AHSM. The maximum depth of the scrape was less than 0.012 inch and was most likely around 0.004 inch deep. Based on these independent laboratory results, Areva Transnuclear determined the canister thickness of the gouged area was still acceptable. The minimum wall thickness of Canister DSC 19, determined during the fabrication process, was 0.666 inch. The gouge reduced the canister wall thickness to 0.654 inch. This reduced thickness in the damaged area still exceeded the minimum wall thickness required by the FSAR of 0.53 inch and the minimum wall thickness specified for the SONGS canisters of 0.61 inch. Areva Transnuclear dispositioned the canister as "use-as-is."

The cask certificate holder, Areva Transnuclear, issued a biennial 72.48 report as required by 10 CFR 72.48(d)(2). The March 23, 2007, report (ML070860267) and the

August 12, 2010 (ML1102290084), reports were reviewed. No issues were identified during the reviews. SONGS was reviewing and verifying that 72.48 evaluations issued by Areva Transnuclear were incorporated into their program when applicable.

2.3 Conclusions

The licensee and the cask vendor were performing the required safety screenings and safety reviews required by 10 CFR 72.48. A sampling of these evaluations was reviewed to determine if an adequate evaluation was being performed. For those reviewed, no issues or concerns were identified related to the conclusions reached.

3 Decommissioning Performance and Status Review at Permanently Shutdown Reactors (71801)

3.1 Inspection Scope

The inspectors toured the footprint of the former Unit 1 reactor facilities and met with staff and management representatives involved with the dismantlement of the Unit 1 reactor. Records associated with ongoing radiological monitoring of the site were reviewed.

3.2 Observations and Findings

The inspectors met with licensee representatives associated with the Unit 1 dismantlement and the ongoing environmental monitoring of the site. The inspectors also toured the area of the former Unit 1 facilities to observe status. During the tour of the Unit 1 footprint, yard drains and sumps were noted to be clear of interference and in good condition. Using a personal dose rate meter (NRC Tag 086962, calibration due June 2, 2011), the inspectors noted that ambient radiation levels were influenced by the removed steam generators from the operating units, the unit 1 reactor vessel package, and the ISFSI. Radiation areas were properly posted.

The inspectors reviewed actions taken by the licensee to characterize and monitor groundwater tritium concentrations in the area around the former unit 1 footprint. Levels measured were below those previously reported in 2006, and for comparison purposes, were well below EPA drinking water standards for tritium. This tritium in the groundwater is being reported in the licensee’s Annual Radioactive Effluent release report. The licensee is preparing an extraction plan to pump this water to the Offsite Dose calculation manual monitored discharge pathway.

3.3 Conclusions

The footprint of the former Unit 1 reactor was being used to store hardware and intermodal containers associated with the operating units’ steam generator replacement project. Yard drains and sumps were clear and in good condition. The area around the Unit 1 reactor vessel package was properly posted and secure.

4 Exit Meeting

The inspectors reviewed the scope and findings of the inspection during an exit meeting conducted at the conclusion of the onsite inspection on April 21, 2011. The licensee did not identify any information as proprietary that was provided to or reviewed by the inspectors.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF CONTACTS

B. Ashe-Everest, Maintenance Engineering Supervisor
B. Bordelon, Reactor/Fuel Maintenance Supervisor
K. Ceribble, Reactor Maintenance Supervisor
K. Deboi, Dry Cask Storage Project Manager
L. Delgado, Canister Fabrication Team Supervisor
C. Gabel, Nuclear Fuel Management Engineer
R. Granaas, Maintenance Engineering Senior Engineer
K. Gribble, Reactor/Fuel Maintenance Supervisor
P. Gyswyt, Design Engineering Organization Senior Nuclear Engineer
N. Hansen, Environmental Technical Specialist/Scientist
R. Hansen, Nuclear Oversights Division Technical Specialist/Scientist
D. Karr, Nuclear Oversight Division Technical Specialist/Scientist
M. Maheu, Reactor/Fuel Maintenance Supervisor
J. Morales, P.E., Projects Manager
V. Nazareth, Nuclear Fuels Management Supervisor
L. Pham, Design Engineering Organization Nuclear Engineer
S. Sherman, Health Physics Technical Specialist/Scientist
A. Sistos, Nuclear Oversight Division Senior Engineer
J. Tipton, Fuel Handling Superintendent
M. Stevens, Regulatory Affairs

INSPECTION PROCEDURES USED

IP 60855.1	Operations of an ISFSI at Operating Plants
IP 60857	Review of 10 CFR 72.48 Evaluations
IP 71801	Decommissioning Performance and Status Review of Permanently Shutdown Reactors

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

NCV	72-41/1101-001	Failure to Perform Adequate 72.48 Evaluation for Procedure Revision

Discussed

None

Closed

NCV	72-41/1101-001	Failure to Perform Adequate 72.48 Evaluation for Procedure Revision

LIST OF ACRONYMS

AHSM	advanced horizontal storage module
CFR	*Code of Federal Regulations*
DSC	dry shielded canister
FSAR	Final Safety Analysis Report
GTCC	Greater-than-Class-C waste
GTRF	Grid-to-Rod Fretting
ISFSI	Independent Spent Fuel Storage Installation
kips	kilopound-force
mR/hr	milliRoentgen/hour
mrem/hr	millirem/hour
NCV	noncited violation
NRC	Nuclear Regulatory Commission
INPO	Institute of Nuclear Power Operations
SCE	Southern California Edison Company
SONGS	San Onofre Nuclear Generating Station
TLD	Thermoluminescent Dosimeter
TN	Areva Transnuclear
μR/hr	microRoentgen/hour

ATTACHMENT 2

LOADED CASKS AT THE SONGS ISFSI

LOADING ORDER	DSC SERIAL No.	AHSM No.	UNIT FUEL	DATE ON PAD	HEAT LOAD (Kw)	BURNUP MWd/MTU (max)	MAXIMUM FUEL ENRICHMENT %	PERSON-REM DOSE
1	DSC003	001	Unit 1	10/03/03	9.35	36.0	4.0006	1.366
2	DSC007	002	Unit 1	11/07/03	9.36	37.3	4.0006	0.667
3	DSC005	003	Unit 1	11/19/03	9.31	37.0	4.0006	0.681
4	DSC006	004	Unit 1	12/12/03	8.84	37.6	4.0006	0.673
5	DSC009	005	Unit 1	01/03/04	8.34	34.9	4.0006	0.695
6	DSC008	006	Unit 1	05/23/04	7.50	40.0	4.02	1.021
7	DSC011	007	Unit 1	06/06/04	7.46	37.6	4.02	1.043
8	DSC010	008	Unit 1	06/20/04	8.08	38.5	4.02	0.776
9	DSC012	009	Unit 1	07/03/04	8.44	38.1	4.02	0.734
10	DSC014	010	Unit 1	07/18/04	7.74	36.7	4.02	0.689
11	DSC015	011	Unit 1	07/30/04	7.46	38.8	4.02	0.633
12	DSC013	012	Unit 1	08/16/04	8.45	40.6	4.02	0.545
13	DSC016	013	Unit 1	08/20/04	7.87	40.3	4.02	0.560
14	DSC017	014	Unit 1	08/31/04	7.23	43.2	4.02	0.348
15	DSC018	015	Unit 1 GTCC	09/02/04	N/A	N/A	N/A	0.555

LOADING ORDER	DSC SERIAL No.	AHSM No.	UNIT FUEL	DATE ON PAD	HEAT LOAD (Kw)	BURNUP MWd/MTU (max)	MAXIMUM FUEL ENRICHMENT %	PERSON-REM DOSE
16	DSC002	016	Unit 1	05/29/05	7.37	42.8	4.00	0.570
17	DSC004	017	Unit 1	06/10/05	8.15	38.3	4.00	0.431
18	DSC001	018	Unit 1	06/28/05	8.11	41.8	4.00	0.453
19	DSC019	019	Unit 2	03/06/07	11.96	48.3	4.04	0.849
20	DSC020	020	Unit 2	03/19/07	12.05	44.8	3.97	0.819
21	DSC021	021	Unit 2	04/05/07	12.13	44.8	3.97	0.502
22	DSC022	022	Unit 2	04/16/07	10.51	38.5	3.95	0.262
23	DSC023	023	Unit 2	04/30/07	11.89	38.6	3.96	0.308
24	DSC024	024	Unit 2	05/14/07	11.07	41.1	3.97	0.192
25	DSC025	025	Unit 2	05/29/07	9.66	38.3	3.97	0.126
26	DSC026	026	Unit 3	03/05/08	8.29	38.2	3.96	0.149
27	DSC027	027	Unit 3	03/18/08	9.99	38.2	3.96	0.262
28	DSC028	028	Unit 3	04/01/08	9.68	37.1	3.96	0.217
29	DSC029	029	Unit 3	04/23/08	11.14	50.1	3.97	0.275
30	DSC030	030	Unit 3	05/29/08	8.49	36.7	3.96	0.143
31	DSC031	031	Unit 3	06/09/08	7.78	29.5	3.46	0.074
32	DSC032	032	Unit 2	03/19/09	12.37	46.0	4.49	0.328
33	DSC033	033	Unit 2	04/08/09	12.85	42.5	4.49	0.177

LOADING ORDER	DSC SERIAL No.	AHSM No.	UNIT FUEL	DATE ON PAD	HEAT LOAD (Kw)	BURNUP MWd/MTU (max)	MAXIMUM FUEL ENRICHMENT %	PERSON-REM DOSE
34	DSC034	034	Unit 2	04/30/09	13.64	42.9	4.49	0.248
35	DSC035	035	Unit 2	05/13/09	13.0	45.8	4.48	0.180
36	DSC036	036	Unit 2	05/30/09	14.71	46.0	4.49	0.220
37	DSC037	037	Unit 3	05/13/10	13.16	48.0	4.6	0.360
38	DSC038	038	Unit 3	05/26/10	14.46	46.0	4.6	0.342
39	DSC039	039	Unit 3	06/09/10	13.16	48.0	4.5	0.278
40	DSC040	040	Unit 3	06/21/10	15.04	47.0	4.6	0.326
41	DSC041	041	Unit 3	07/03/10	15.26	47.0	4.6	0.326
42	DSC042	042	Unit 3	07/13/10	15.39	47.0	4.6	0.288

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