Document Transmittal Form

To: NRC ATTN:DOC. CONTROL
ONE WHITE FLINT NORTH
WASHINGTON, DC 20555-001

ID: 0148

Date: 6/12/2012

Please update your controlled set of documentation with the following documents:

Document ID	Revision	Status	Quantity	Format	RecNo
72-0048 212 FSAR SPENT FUEL	7	A	1	H	KP06121202

For questions concerning distributions, please contact Records Management @ (856) 339-1063

If acknowledging via this form ALL PAGES must be returned to:

Records Management, N64
PSEG Nuclear
PO Box 236
Hancocks Bridge, NJ 08038

Acknowledgement may alternately be faxed to (856) 339-2033, emailed to RMPROC@pseg.com or via intranet DCRMS-XMTL application

Your signature below or electronic acknowledgement verifies that:

(1) the above documents have been filed and superseded documents have been removed and destroyed or clearly marked as obsolete.

(2) the mailing address and copyholder information are correct or corrections have been identified

☐ Place checkmark here to be removed from controlled distribution.

Signature: ______________________ Date: __________

NMSS26

PSEG
NUCLEAR LLC

Salem/Hope Creek Generating Station

Independent Spent Fuel Storage Installation

10 CFR 72.212 Evaluation Report

NRC Docket 72-0048

Revision 7

REVISION SUMMARY

REV. NO.	CHANGE AND REASON FOR CHANGE
0	Initial issue.
1	Sections 5.4.1.8.1 and 5.4.1.8.4 are revised to reflect reduction of the minimum ambient operating temperature for the HERMIT from 40°F to 0°F and to add Reference 6.31.5 to the list of procedures. Section 5.4.1.10.1 is revised to indicate a slightly lower minimum pressure pulse required to tip over the cask. References 6.31.4, 6.31.5, and 6.31.6 are revised to reflect the revision levels that incorporate this change into the dry cask storage (DCS) operating procedures. Reference 6.47 is revised to reflect the revision level of the analysis that addresses the revised temperature limit.

2

72.212 Section	Description of Change
5.2.1.1	Corrected the reference cited for the 35 ft separation between rows of casks for transporter access.
5.3.1.2, last paragraph	Added "A" after "Appendix."
5.4.1.1, 2nd paragraph	Clarified that the SER section cited is from the Amendment 0 SER for the cask.
5.4.1.3	Added a paragraph to address compliance with the 15 ft/sec maximum flood water velocity from the CoC
5.4.1.7	Revised the subsection to more clearly describe the burial under debris event in the cask FSAR as a bounding analysis for the Salem/Hope Creek Generating Station site.
5.4.1.8.3	Revised the subsection to define the use of soil temperature in the cask thermal analysis and further support the fact that the generic value used by the cask vendor is bounding for the Salem/Hope Creek Generating Station site.
5.4.1.10, 1st paragraph	Corrected typo: "primer" to "prime"
5.4.1.10, 3rd paragraph	Added specific value and reference for the cask crawler pressure loading and provided a reference for the heavy haul path design.
5.4.1.11.2	Added reference for the cooling tower collapse evaluation results.
Table 5.4.1.12-1	Added ECOs 1024-126, 1026-33, 1026-41, 5014-131, and 5014-132 to the list and added revision numbers to other ECOs. Corrected revision number for ECO 1021-63.
Table 5.4.1.12-2 and preceding text	Added references and discussion of pad repair for ISFSI Pad No. 1.
5.5.1	Added discussion pertaining to the receipt of NRC approval of amendment 169 to the Hope Creek license and corrected a typo in the DCP number.
6.0	Added Calculation A-5-DCS-CDC-1963 as Reference 6.28. Corrected procedure title of Reference 6.54.3. Corrected/added revision levels for several references. Corrected VTD number for reference 6.32.7. Corrected

REV. NO.	**CHANGE AND REASON FOR CHANGE**
2 (cont'd)	procedure number for Reference 6.33.5.

72.212 Section	**Description of Change**
Appendix 1, Table 3, Section 2.1.2 and 2.1.3, 2nd paragraph	Revised description of the type of fuel verification performed to accurately use the terms "independent" and "concurrent" in three places to be consistent with terminology in procedure HU-AA-101.
Appendix 1, Table 3, Section 3.4.6, 5th paragraph	Revised text to reflect resolution of ponding repair on ISFSI pad No. 1.
Appendix 2	Added ECO 1024-126, 1026-33, and 1026-41, to the list and deleted the row pertaining to FSAR changes not specifically associated with cask hardware. Corrected the dates for the initial loading campaign to reflect actual information.

3	Revised throughout to adopt Amendment 3 to the HI-STORM 100 System CoC and Revision 5 to the HI-STORM 100 FSAR for the second Hope Creek loading campaign, and to reflect cask loading procedure revisions and re-numbering. Editorial improvements and typographical corrections are also made. See 72.48 Screening 08-01 for a detailed listing of the changes made in this revision.
4	Revised throughout to adopt Amendment 5 to the HI-STORM 100 System CoC and Revision 7 to the HI-STORM 100 FSAR for Hope Creek. Editorial improvements and typographical corrections are also made. See 72.48 Coversheet/Screening H10-01 for a detailed listing of the changes made in this revision.
5	Revised throughout to add Salem spent fuel to the ISFSI and make editorial and administrative improvements. Editorial improvements and typographical corrections are also made. See 72.48 Coversheet/Screening S10-02 for a detailed listing of the non-editorial/administrative changes made in this revision.
6	Revised to add cask serial numbers for the 2011 loading campaign at Salem Unit 1, update Part 72 rule section numbers per the 2011 rule change, address seismic restraints at Salem Unit 1, and make editorial improvements.
7	Revised to add cask serial numbers for the 2012 loading campaign at Salem Unit 2, update the compliance discussion for CoC Condition 9, address seismic restraints at Salem Unit 2, and make editorial improvements.

TABLE OF CONTENTS

1.0 Introduction

The United States Department of Energy (DOE) did not meet their legal obligation to begin removing spent nuclear fuel from domestic commercial nuclear reactor sites by January 31, 1998. Therefore, PSEG Nuclear is required to provide additional on-site interim storage for spent fuel from the Salem and Hope Creek nuclear power plants until such time as DOE does begin taking the fuel. PSEG began moving spent fuel from the Hope Creek Generating Station (HCGS) spent fuel pool into dry storage in 2006 to create sufficient wet storage capacity to support safe power operations and maintain full core offload capability in the spent fuel pool. Spent fuel from Salem Generating Station (SGS) Unit 1 was added to the Independent Spent Fuel Storage Installation (ISFSI) beginning in 2010. Spent fuel from SGS Unit 2 was added to the ISFSI in 2012. Spent fuel from both Salem units and Hope Creek will periodically be moved from the respective spent fuel pools to the ISFSI for the foreseeable future.

PSEG Nuclear operates an ISFSI facility for interim storage of SGS and HCGS spent fuel in dry casks under the general license provision of 10 CFR 72, Subpart K at the Salem/Hope Creek Generating Station site. Interim storage of SGS and HCGS spent fuel at the ISFSI occurs in NRC-certified dry storage casks. The Holtec International HI-STORM 100 System has been selected for the storage of spent fuel from SGS and HCGS at the ISFSI. Nuclear Regulatory Commission (NRC) Certificate of Compliance (CoC) No. 72-1014 (Reference 6.1) confers NRC approval of the HI-STORM 100 System design for use by Part 72 general licensees and the HI-STORM 100 System is listed as an NRC-approved dry spent fuel storage cask system in 10 CFR 72.214. The design and licensing basis for the HI-STORM 100 System is provided in the CoC and the supporting HI-STORM 100 System FSAR (Reference 6.2). The HI-STORM 100 System is comprised of the Multi-Purpose Canister (MPC), the HI-STORM overpack, the HI-TRAC transfer cask, and necessary ancillary equipment described in the cask FSAR.

References are identified throughout the body of this report and are listed in Section 6.0. References include analyses, calculation packages, drawings, procedures, correspondence, and other documents. The reference documents are intended to provide supporting or background information and additional detail that the reader may refer to in order to learn more about a particular topic presented in this document, but are generally not considered part of this report. A referenced document shall be considered to be a part of this report only if it is clearly annotated as being "incorporated by reference" in this report. Documents incorporated by reference into this report are subject to the same administrative controls and regulatory requirements as the main report (i.e., changes are controlled by 10 CFR 72.48).

2.0 Purpose

The purpose of this report is to document the written evaluations required by 10 CFR 72.212(b) for use of a dry cask storage system to store spent fuel at an on-site ISFSI under a 10 CFR 72 general license. For the first dry fuel storage campaign at HCGS (casks 1 through 4[1]), the written evaluations were based on HI-STORM 100 System 10 CFR 72 CoC Number 1014, Amendment 2 and HI-STORM FSAR Revision 3, including applicable interim changes (e.g., those authorized under 10 CFR 72.48). As ISFSI operations continue over time, the applicable CoC amendment

[1] PSEG FLOC numbers and manufacturer serial numbers for each cask are provided in Appendix 2 of this report.

and/or FSAR revision may change. However, the cask CoC amendment and FSAR revision under which a cask was originally loaded and deployed at the ISFSI remain in effect during storage operations and for unloading a cask, if necessary (see Reference 6.62). This report will be revised, at a minimum, for each dry fuel storage campaign to list the applicable CoC amendment, FSAR revision, and interim changes for the MPC and overpack serial numbers to be added to the ISFSI and for general changes to the HI-STORM FSAR (see Section 5.4.1.12 and Appendix 2 of this report). Because the HI-TRAC transfer cask is re-useable, only one has been fabricated and its design basis is not expected to change. Certification of the transfer cask for use with later CoC amendments and FSAR revisions is tracked in the HI-STORM FSAR, Section 1.0.2.

Revision 3 to this report adopted HI-STORM CoC Amendment 3 and FSAR Revision 5 for the casks deployed in the 2008 cask loading campaign at HCGS (casks 5 though 12). Revision 4 to this report adopted HI-STORM CoC Amendment 5 and FSAR Revision 7 for the casks deployed in the 2010 cask loading campaign at HCGS (casks 13 through 16) and future HCGS casks. Revision 5 to this report added SGS fuel to the ISFSI. HI-STORM CoC Amendment 5 and FSAR Revision 7 is the licensing basis for the SGS fuel casks deployed in the 2010 cask loading campaign (Casks 17 through 20) and all future SGS fuel casks, until such time as a later licensing basis is adopted by PSEG. See Appendix 2 for details.

Revision 6 to this report adds the serial and FLOC numbers for the MPCs and overpacks to be loaded with Salem Unit 1 fuel in 2011, reflect changes to the Part 72 regulations that became effective in May, 2011, and addresses the addition of seismic restraints for the stackup configuration in the Salem Unit 1 truck bay. The licensing basis for these casks is CoC Amendment 5 and FSAR Revision 7.

Revision 7 to this report adds the serial and FLOC numbers for the MPCs and overpacks to be loaded with Salem Unit 2 fuel in 2012, updates the compliance discussion for CoC Condition 9, and addresses the addition of seismic restraints for the stackup configuration in the Salem Unit 2 truck bay. The licensing basis for these casks is CoC Amendment 5 and FSAR Revision 7.

3.0 Background

In order to provide adequate spent fuel storage capacity for the Salem and Hope Creek Generating Stations, PSEG Nuclear operates an on-site ISFSI. The on-site ISFSI is located inside the Salem/Hope Creek Generating Station protected area near the north boundary of the site, just west of the HCGS cooling tower. The ISFSI location is shown in Section 8.7 of Reference 6.31.3. The Salem/Hope Creek ISFSI is comprised of three concrete storage pads designed to provide storage for up to 200 HI-STORM 100 System storage overpacks containing seal-welded MPCs.

The ISFSI pads are sized and structurally designed for storage of HCGS and SGS spent fuel in any of the following three dry cask storage systems approved by the NRC:

- Holtec HI-STORM 100 – CoC 1014
- Transnuclear Standard NUHOMS System – CoC 1004
- NAC International UMS Universal Storage System – CoC 1015

PSEG Nuclear has elected to deploy the Holtec HI-STORM 100 System for dry storage of SGS and HCGS spent fuel at the ISFSI. Deployment of another CoC holder's dry cask storage system design or use of a different HI-STORM 100 System overpack or MPC model than those described herein will require supplemental evaluations and either a revision of this 72.212 evaluation report or the creation of a separate 72.212 evaluation report reflecting the new dry storage system or components.

The Salem/Hope Creek ISFSI is operated under the conditions of the general license granted in accordance with 10 CFR 72.210. The first four casks containing HCGS fuel were loaded and placed at the ISFSI in 2006-07 under HI-STORM 100 System CoC Amendment 2, its supporting FSAR Revision 3, plus certain changes authorized under 10 CFR 72.48, as shown in Section 5.4.1.12 and Appendix 2 to this report. An additional eight casks containing HCGS fuel were loaded in 2008 in accordance with CoC Amendment 3 and FSAR Revision 5. Casks loaded with HCGS fuel after the first 12, and all casks containing SGS fuel were/are being loaded in accordance with CoC Amendment 5 and FSAR Revision 7. The amendment of the CoC and the revision of the cask FSAR upon which this report is based varies by cask serial number as described above and as shown in Appendix 2. The CoC amendment under which casks are loaded remains the governing CoC amendment for those casks unless altered by an exemption to the Part 72 regulations (Reference 6.62). For convenience, the FSAR revision applicable to those CoC amendments is also retained as the governing document or those casks as shown in Appendix 2.

The spent fuel stored at the ISFSI will eventually be shipped offsite in transportation casks approved for shipment of spent fuel in accordance with 10 CFR 71. The HI-STORM 100 System is the storage-only counterpart of the Holtec HI-STAR 100 System, which uses an identical MPC design. The HI-STAR 100 System is the canister-based, 10 CFR 71-certified transportation package (CoC 71-9261). Because the MPC is designed to meet the requirements of both 10 CFR 71 and 10 CFR 72 for transportation and storage, respectively, the HI-STORM 100 System allows rapid decommissioning of the ISFSI by simply transferring the fuel-loaded MPCs directly from the HI-STORM storage overpack into the HI-STAR 100 transportation overpack for offsite transport without re-packaging.

4.0 Regulatory Requirements and License Conditions

10 CFR 72.210 grants a general license for the storage of spent fuel at an on-site ISFSI to holders of a 10 CFR 50 license at the associated reactor site. 10 CFR 72.212 establishes the conditions for use of a Part 72 general licensee.

One of the general license conditions is to prepare and maintain written evaluations demonstrating compliance with certain regulatory requirements, as discussed in 10 CFR 72.212(b)(2). The various “Evaluation” subsections of this report summarize the written evaluations and analyses performed to ensure that the generic HI-STORM 100 System design criteria bound the site-specific design criteria at the Salem/Hope Creek Generating Station site, or that a site-specific evaluation was performed in certain cases. This report addresses the five regulations required by 10 CFR 72.212(b) to be included, plus, for completeness, two additional regulations pertaining to program enhancements (e.g., 10 CFR 50.54 programs). Compliance with other 10 CFR 72 regulatory requirements applicable to general licensees pursuant to 10 CFR 72.13 is controlled via the associated implementing procedures and programs. The seven regulations addressed in this report are:

- 10 CFR 72.212(b)(5)(i) - Certificate of Compliance Terms, Conditions, and Specifications
- 10 CFR 72.212(b)(5)(ii) - Cask Storage Pad Design
- 10 CFR 72.212(b)(5)(iii) - Dose Analyses Pursuant to 10 CFR 72.104

- 10 CFR 72.212(b)(6) - Review of the Cask FSAR and SER
- 10 CFR 72.212(b)(8) - Review of Part 50 Facility Impact (10 CFR 50.59)
- 10 CFR 72.212(b)(9) - Security Plan
- 10 CFR 72.212(b)(10) - Programs

Unless otherwise noted, all requirements pertaining to the HI-STORM 100 System in the CoC apply equally to casks containing a BWR MPC (for HCGS) or a PWR MPC (for SGS). Likewise, generic analyses and site-specific parameters described in the cask FSAR are bounding for casks containing either a BWR MPC or a PWR MPC.

5.0 EVALUATION

5.1 10 CFR 72.212(b)(5)(i) - Certificate of Compliance Terms, Conditions, and Specifications

10 CFR 72.212(b)(5)(i) states that the general licensee shall perform written evaluations, before use and before applying changes authorized by an amended CoC to a cask loaded under the initial CoC or an earlier amended CoC, which establish that:

"The cask, once loaded with spent fuel or once the changes authorized by an amended CoC have been applied, will conform to the terms, conditions, and specifications of a CoC or an amended CoC listed in §72.214."

5.1.1 Evaluation

The NRC confers approval of a dry spent fuel cask storage system by issuance of a CoC in accordance with Subpart L of 10 CFR 72. The HI-STORM 100 System CoC permits the storage of a wide range of BWR and PWR nuclear power plant spent fuel and non-fuel hardware. Therefore, portions of the CoC are applicable to the storage of HCGS spent fuel at the ISFSI and portions are applicable to storage of SGS spent fuel at the ISFSI. Some portions of the CoC are applicable to storage of spent fuel from both stations and/or the ISFSI facility separate from the stored fuel. Use of the HI-STORM 100 System at the ISFSI is conditioned upon compliance with all applicable conditions set forth in the CoC. The conditions in HI-STORM 100 System CoC No. 72-1014, Amendments 2, 3, and 5 that are applicable to storing HCGS and SGS spent fuel at the ISFSI have been evaluated (only CoC Amendment 5 applies to storage of SGS spent fuel). These evaluations are documented in Appendix 1 to this report, Tables 1 through 3. Appendix 2 to this report shows the licensing basis (CoC amendment, FSAR revision, and applicable interim changes) for each licensed component (MPC, overpack, and transfer cask) deployed at the Salem/Hope Creek ISFSI, by manufacturer serial number and PSEG Functional Location (FLOC) number. PSEG Nuclear is not applying the terms, conditions, and specifications of a later HI-STORM CoC amendment to the casks loaded under a previous CoC amendment at this time.

5.1.2 Conclusion

As documented in Appendix 1 to this report, PSEG Nuclear has evaluated HI-STORM 100 System Certificate of Compliance No. 72-1014, including Appendices A and B, against the SGS and HCGS fuel and site-specific conditions and has determined that the applicable conditions in the CoC are met. Therefore, PSEG Nuclear complies with the requirements of 10 CFR 72.212(b)(5)(i).

5.2 10 CFR 72.212(b)(5)(ii) - Cask Storage Pad Design

10 CFR 72.212(b)(5)(ii) states that the general licensee shall perform written evaluations, before use and before applying changes authorized by an amended CoC to a cask loaded under the initial CoC or an earlier amended CoC, which establish that:

"Cask storage pads and areas have been designed to adequately support the static and dynamic loads of the stored casks, considering potential amplification of earthquakes through soil-structure interaction, and soil liquefaction potential or other soil instability due to vibratory ground motion."

5.2.1 Evaluation

The following evaluation is specific to the design of the ISFSI pad for the Holtec HI-STORM 100S Version B-218 overpack model. Use of other HI-STORM overpack models or a different certified dry cask storage system will require re-evaluation and a revision to this report.

5.2.1.1 ISFSI Storage Pad General Description

The primary function of the ISFSI storage pad is to provide a stable foundation for supporting the storage casks under all normal, off-normal and credible accident conditions of storage, including natural phenomena such as earthquakes and tornadoes. Section 3.1.2.3 of the NRC Safety Evaluation Report for Amendment 1 to the HI-STORM 100 CoC (Reference 6.5) states that when the HI-STORM 100 System is deployed in the free-standing mode, the ISFSI pad/basemat is considered not important-to-safety (NITS). If the HI-STORM 100 System is deployed in the anchored condition, the ISFSI pad/basemat is considered important-to-safety (ITS). As shown in the evaluation below, the seismic accelerations at the Salem/Hope Creek ISFSI site are are sufficiently low to permit deployment of the HI-STORM System casks in the free-standing mode. Nonetheless, the ISFSI pad is classified as ITS, Category B (ITS-B) to assure an appropriate level of quality assurance is applied to activities associated with pad design, construction, testing, inspection, and records management.

The ISFSI facility is comprised of three separately constructed, reinforced structural concrete pads[2] and necessary supporting structures, systems, and components. Two of the concrete pads have approximate dimensions of 36 inches thick, 91 feet wide and 260 feet long, and the third differs in length only; it is approximately 248 feet, 8 inches long (Reference 6.18). The three pads are oriented with their long dimension in the north-south direction, approximately 14 feet apart edge-to-edge along their length. The west and middle pads will be used to store 68 casks each, and the east pad will be used to store 64 casks for a total ISFSI capacity of 200 casks. The west

[2] Throughout this document, the text may refer to the ISFSI "pad" or "pads" interchangeably.

and middle pads each support 68 casks in two 2x17 arrays. The east pad supports 64 casks in two 2x16 arrays. The casks are approximately 11’-2” in diameter (Reference 6.32.6). The center-to-center spacing between each cask in an array is 15 feet. There is a 35-foot separation distance along the center of each pad between arrays for crawler access (Reference 6.14, Figure 2).

The pad foundations are designed to accommodate storage of fuel-loaded HI-STORM 100S Version B-218 overpacks and the other NRC-approved dry storage casks listed in Section 3.0 above. The size and layout of the ISFSI storage pads were chosen to be compatible with operation of the vertical cask transporter (VCT). The pads are essentially flush with the surrounding grade, where needed, in order to permit the VCT to be driven directly onto the pad surface. The area surrounding the ISFSI storage pad is compacted gravel (References 6.18 and 6.37).

5.2.1.2 Storage Pad Design

Each of the three ISFSI storage pads is designed to adequately support the static weight of the maximum capacity of fuel-loaded HI-STORM casks (defined in Section 5.2.1.1) in accordance with 10 CFR 72.212(b)(5)(ii). In addition, the pads are designed for appropriate combinations of the effects of normal, off-normal and accident conditions, including the effects of natural phenomena in accordance with 10 CFR 72.122(b). The design of the ISFSI storage pad meets the requirements of 10 CFR 72 for the loads and load combinations specified in the HI-STORM FSAR, NUREG-1567, NUREG-1536, and ACI 349-01 (References 6.2, 6.6, 6.7, and 6.8). The structural analysis of the pad considered the sequential, partial and total load of fuel-loaded HI-STORM overpacks and the VCT loads. The live load on the pad due to operation of the VCT was also considered in the analysis (Reference 6.18).

The ISFSI storage pad is designed to remain functional under earthquake loading conditions so as to preclude a cask tip-over event or excessive cask sliding. According to 10 CFR 72.102(f), the ISFSI design basis earthquake (DBE) must be equivalent to the safe shutdown earthquake (SSE) for the nuclear power plant. The Salem/Hope Creek ISFSI is physically located near the HCGS. Therefore, the ISFSI storage pads are designed to remain functional under the influence of the SSE for the HCGS.

Details of the ISFSI pad design, design criteria, and supporting analyses may be found in References 6.18 through 6.20 and 6.37.

5.2.1.3 Geotechnical Investigation

The site-specific investigations of the ISFSI site, laboratory soils analysis, and foundation structural analysis demonstrate that soil conditions are adequate for the foundation loading to limit total and differential settlements due to both static and dynamic loading conditions once the subsoil has been improved (References 6.18, 6.19, 6.20, and 6.26). See also Section 5.2.1.6 below.

5.2.1.4 Cask Handling Accident and Cask Tip-Over Evaluation

The ISFSI pad and subgrade are designed and constructed to provide adequate support for the dry storage casks under all applicable static and dynamic load combinations. In addition, the CoC holder has made assumptions as to the maximum hardness of the ISFSI pad and subgrade in its design basis cask drop and tipover analyses used in licensing the dry storage cask system. Two sets of physical parameters for the ISFSI pad were used as inputs in the design basis cask drop and tipover analysis models, which establish alternate sets of maximum values not to be exceeded in the design of the ISFSI pad. These two sets of limiting parameters are shown in HI-STORM FSAR Table 2.2.9. ISFSI pad designers may choose either set of parameters as the limits for their design or choose a unique design basis that is shown by analysis to accomplish the required design functions of the storage pad. The designer of the Salem/Hope Creek ISFSI chose the Set 'A' parameters from HI-STORM FSAR Table 2.2.9 as the maximum limits for the pad and subgrade design as shown below:

a. Concrete thickness: ≤ 36 inches
b. Concrete compressive strength: ≤ 4,200 psi at 28 days
c. Reinforcement top and bottom (both directions)
   Reinforcement area and spacing determined by analysis
   Reinforcement shall be 60 ksi yield strength ASTM material
d. Soil Effective Modulus of Elasticity: ≤ 28,000 psi

The ISFSI pad was designed and constructed to be 36 inches thick (maximum) and is placed on compacted engineered fill (soil thickness greater than 36 inches per Reference 6.19). The pad is designed to be structurally adequate with a minimum concrete compressive strength of 3,000 psi. As-built concrete compressive strength has been determined by test to be less than 4,200 psi, in all cases. Steel reinforcing bars are placed in both directions at the top and bottom faces of the pad, and are designed in accordance with ACI 349-01. The steel reinforcement used is ASTM A516 Grade 60 (60 ksi yield strength). However, the soil effective modulus of elasticity was determined to be greater than 28,000 psi (Reference 6.18, Attachment J).

Because the ISFSI pad design does not meet all of the criteria for either ISFSI pad Set 'A' or 'B' from HI-STORM FSAR Table 2.2.9, a site-specific cask tipover analysis was performed (Reference 6.36). The results of this analysis show that the deceleration value at the top of the fuel assemblies is 39.2 g's. This deceleration value is less than the design basis value of 45 g's and is, therefore, acceptable per HI-STORM CoC, Appendix B, Section 3.4.6.a. The site-specific tipover analysis was performed using the analysis method and model described in the HI-STORM FSAR, which bounds both PWR and BWR MPCs. Thus, the site-specific tipover analysis performed for HCGS casks in Reference 6.36 provides a bounding case for SGS casks. A cask drop is not required to be postulated or analyzed because the VCT, which moves the loaded casks to, from, and within the ISFSI, is designed in accordance with ANSI N14.6 and has redundant drop protection features (Reference 6.8).

5.2.1.5 Cask Sliding and Overturning Evaluation

<u>Normal, Dry Conditions</u>

The loaded HI-STORM 100 System cask is designed to withstand a seismic event defined by three orthogonal, statistically independent acceleration time-histories as described in Section 3.4.7 of the HI-STORM 100 System FSAR. The HI-STORM 100 System FSAR states that for the purpose of performing a conservative analysis to determine the maximum Zero Period Acceleration (ZPA) that will not cause incipient tipping, the cask is considered as a rigid body subject to a net horizontal quasi-static inertia force and a vertical quasi-static inertia force. The analysis used in the design of the HI-STORM 100 System uses a finite element model representing the cask, pad, and engineered backfill supported on existing substrata. The input motion used in the following discussion corresponds to the safe shutdown earthquake (SSE) for the HCGS, which is the design basis earthquake for the ISFSI. Strain-dependent soil modulus and damping values (compatible to strains developed during the SSE) are used.

The calculated peak seismic vertical ZPA, G_V, and horizontal ZPA, G_H, expressed as fractions of 'g' at the top of the ISFSI storage pad are as follows (Reference 6.20):

$$G_V = 0.23 \text{ g}$$
$$G_{H,\ Longitudinal} = 0.36 \text{ g}$$
$$G_{H,\ Transverse} = 0.32 \text{ g}$$

The vectorial sum of the two horizontal components is given as:

$$G_{H(R)} = ((0.36)^2 + (0.32)^2)^{1/2}$$
$$G_{H(R)} = 0.482 \text{ g}$$

HI-STORM 100 System CoC, Appendix B, Section 3.4.3.a provides the seismic requirements for deploying the HI-STORM 100 System in the free-standing mode. The following inequality must be met using the above ZPAs:

$$G_H + \mu G_V \leq \mu$$

Where: μ is the Coulomb friction coefficient for the cask/ISFSI pad interface (for sliding).

<u>AND</u>

μ is also the ratio r/h, where 'r' is the radius of the cask and 'h' is the height of the cask center-of-gravity (CG) above the ISFSI pad surface (for tipover).

The inequality must be met for both definitions of μ.

<u>For μ as the coefficient of friction:</u>

In accordance with HI-STORM CoC Appendix B, Section 3.4.3.a, absent testing to demonstrate a higher value is appropriate for use, μ as the coefficient of friction is set equal to 0.53 (References 6.29 and 6.30).

$$0.482 + (0.53)(0.23) = 0.60, \text{ which is greater than } 0.53$$

The inequality is not met.

<u>For μ as the ratio of cask radius to CG height:</u>

There are several values of overpack radius and center-of-gravity in the HI-STORM 100S and 100S Version B overpack drawings (Reference 6.2, Section 1.5, Drawings 3669 and 4116) and FSAR Table 3.2.3, respectively. For the HI-STORM 100S Version B-218 overpack design being used at the Salem/Hope Creek ISFSI, the value of 'μ' is calculated as follows:

r = 132.5”/2 = 66.25” (for the HI-STORM 100S Version B overpack per Reference 6.32.6, Sheet 3)

h = 111.88” (conservatively high value for the HI-STORM 100S Version B-218 with loaded MPC-32 per Reference 6.2, Table 3.2.3)

Then, r/h = 66.25/111.88 = μ = 0.592

0.482 + (0.592)(0.23) = 0.618, which is greater than 0.592

Again, the inequality is not met.

As an alternative to evaluating the inequality using ZPAs, Section 3.4.3.a of Appendix B of the HI-STORM CoC permits use of acceleration time-histories, in which case the values of G_H and G_V may be the coincident values of the instantaneous net horizontal and vertical accelerations. If instantaneous accelerations are used, the inequality must be evaluated at each time step in the acceleration time-history over the total duration of the seismic event.

For $G_V = 0.23$ g, the inequality $G_H + \mu (G_V) \leq \mu$ with $\mu = 0.53$ for sliding governs. That is, if the inequality is satisfied with $\mu = 0.53$, it will also be satisfied for the overturning case, with $\mu = 0.592$.

The inequality $G_H + \mu (G_V) \leq \mu$ with $\mu = 0.53$ has been checked with G_H equal to the vectorial sum of two instantaneous horizontal acceleration components and the corresponding instantaneous vertical acceleration, G_V over the total duration of the seismic event. The results show that the inequality is satisfied (Reference 6.20, Section 6.7).

Because the Salem/Hope Creek ISFSI site-specific seismic criteria are acceptable under both definitions of ‘μ’ required by the HI-STORM CoC, sliding or tipover of the HI-STORM overpack will not occur under dry ISFSI pad conditions and the casks may be deployed at the Salem/Hope Creek ISFSI in the free-standing mode.

ISFSI Pad Icing (Reference 6.4)

Section 3.4.3.b of the HI-STORM CoC requires an evaluation of a degraded pad/cask interface friction (such as due to icing) to ensure that an earthquake will not result in a cask tipping over or falling off the ISFSI pad. Additionally, the evaluation must ensure any impact between casks results in g-loads no greater than 45 g’s.

This evaluation has been performed (Reference 6.20, Appendix K) and the results show that the casks will not tipover, fall off the pad, or experience cask-to-cask impact. Therefore, the HI-STORM 100S Version B overpacks at the Salem/Hope Creek ISFSI comply with the HI-STORM CoC in this regard.

5.2.1.6 Soil Liquefaction Evaluation

During a seismic event, the development of additional pore water pressure[3] in saturated, non-cohesive soil reduces the effective confining pressure and the shear strength to a low value. This phenomenon is called liquefaction. The potential for a soil to liquefy depends on the soil classification, its compaction level (relative density), ground water level, and the intensity of the earthquake. 10 CFR 72.212(b)(5)(ii) requires that soil liquefaction effects be evaluated for the subgrade underneath ISFSI pads.

The Salem/Hope Creek ISFSI pad is founded on improved subsoil. Directly under the pads there is 3 feet of granular engineered fill compacted to a relative density of 85%. The hydraulic fill below the 3 feet of granular engineered fill has been improved by the use of an array of short and long "soilcrete" columns, each with a cross-sectional area of approximately 34.7 square feet. The bottom ends of the long soilcrete columns are located in the stiff-to-hard clay layer approximately 51 feet below the top of the ISFSI pad. The bottom of short soilcrete columns are located approximately 28 feet below the top of the ISFSI pad. The spacing of the long soilcrete columns is about 16.5 feet center-to-center in the east-west direction and about 7.4 feet center-to-center in the north-south direction. The short soilcrete columns have the same center-to-center spacing as the long columns with rows of short columns placed in between the long columns in the east-west direction (Reference 6.32.9).

The groundwater level in the area of the ISFSI pad was considered in the analysis for the bearing capacity and in the evaluation for liquefaction potential during the postulated design basis SSE. The soil liquefaction evaluation (Reference 6.18) is based on soil data for the existing soil strata at the ISFSI pad site provided in the geotechnical investigation performed for the Salem/Hope Creek ISFSI site (Reference 6.26) and foundation soil improved by the soilcrete columns. The Hope Creek SSE seismic ground motion was used in the soil liquefaction and foundation load-carrying capacity analysis (Reference 6.20). The soil liquefaction and foundation load-carrying capacity analysis for the SSE event concludes that:

1. The 3-foot thick engineered fill directly under the pads does not liquefy.

2. The in-situ soil strata below the engineered fill, from a depth of 3 feet to 24 feet has a potential for liquefaction. However, soil improvement (3 feet thick engineered fill and soilcrete columns below the engineered fill) precludes liquefaction of the foundation soil directly below the ISFSI pads. The foundation load carrying capacity analysis shows that the long soilcrete columns are adequate for supporting the design loads imposed upon them.

3. The soil below 24 feet does not liquefy.

---

[3] Pore water pressure is defined as that part of the total normal stress in a saturated soil that is due to the presence of interstitial water (Glossary of Geology, American Geological Institute, 1972). If pressure is exerted on a soil sample, the water does not flow instantaneously. This pressure results in an excess of water in the remaining void space in the soil, hence an increase in the pore water (neutral) pressure. This increase in water pressure reduces the total pressure on a plane to an effective value. As the pore water pressure increases, the effective stress in the soil decreases. If the neutral pressure is significantly increased, the effective stress could be reduced to zero, i.e. a granular soil will possess no shear strength (Foundation Analysis and Design, 3rd edition, J. E. Bowles, 1982).

5.2.2 Conclusion

The Salem/Hope Creek ISFSI pads are designed to adequately support the static and dynamic loads of the HI-STORM 100S Version B-218 overpacks. The ISFSI storage pads are designed for the loads and load combinations specified in NUREG-1567, NUREG-1536, and ACI 349-01, and they meet the requirements of 10 CFR 72 and the HI-STORM 100 System FSAR and Certificate of Compliance for sliding and overturning. Therefore, PSEG Nuclear complies with the requirements of 10 CFR 72.212(b)(5)(ii).

5.3 10 CFR 72.212(b)(5)(iii) – Dose Analyses Pursuant to 10 CFR 72.104

10 CFR 72.212(b)(5)(iii) states that the general licensee shall perform written evaluations, before use and before applying changes authorized by an amended CoC to a cask loaded under the initial CoC or an earlier amended CoC, which establish that:

"…the requirements of §72.104 have been met"

10 CFR 72.104 - "Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS," states the following:

"(a) During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 0.25 mSv (25 mrem) to the whole body, 0.75 mSv (75 mrem) to the thyroid and 0.25 mSv (25 mrem) to any other critical organ as a result of exposure to:

(1) Planned discharges of radioactive materials, radon and its decay products excepted, to the general environment,

(2) Direct radiation from ISFSI or MRS operations, and

(3) Any other radiation from uranium fuel cycle operations within the region.

(b) Operational restrictions must be established to meet as low as is reasonably achievable objectives for radioactive materials in effluents and direct radiation levels associated with ISFSI or MRS operations.

(c) Operational limits must be established for radioactive materials in effluents and direct radiation levels associated with ISFSI or MRS operations to meet the limits given in paragraph (a) of this section."

5.3.1 Evaluation

5.3.1.1 Controlled Area Boundary

The ISFSI is located at the north end of the Salem/Hope Creek Generating Station site inside the Salem/Hope Creek Protected Area (PA) just west of the Hope Creek cooling tower. Therefore, the ISFSI controlled area boundary is the same as the Salem/Hope Creek Generating Station exclusion area boundary. The distance between the ISFSI facility and the nearest point on the controlled area boundary (north direction) is 469 meters (Reference 6.14, Table 38). Therefore, the ISFSI controlled area meets the requirements for a controlled area as defined in 10 CFR 72.3 and 10 CFR 72.106 as being under the authority of the licensee for its use and being at least 100 meters to the nearest boundary. The ISFSI controlled area is not traversed by a highway or railroad. The owner-controlled site access road is a public roadway. The road permits vehicular access to the site and a small number of private residences, and terminates at the Salem/Hope Creek Generating Station. It is not a throughway. A portion of the Delaware River to the west is within the owner controlled area.

Doses were calculated at the shortest distance from a cask on the storage pad to the controlled area boundary (469 meters north of the ISFSI pads) and at the Delaware River shoreline, located 331 meters west of the ISFSI pads (Reference 6.14, Table 38).

5.3.1.2 Dose due to Normal Operations and Anticipated Occurrences

Compliance with the dose limits specified in 10 CFR 72.104(a) is dependent on site-specific considerations such as the number of casks stored on the ISFSI storage pad, cask array configuration, cask contents, and the distance between the casks on the ISFSI and the controlled area boundary. In addition, an evaluation of compliance with 10 CFR 72.104(a) must include doses from other fuel cycle activities (i.e., Salem and Hope Creek plant operations). PSEG Nuclear performed a dose evaluation for normal plant and ISFSI operations and off-normal cask events based on storage of 200 fully loaded HI-STORM 100S Version B overpacks at the ISFSI (Reference 6.14). Storage of additional casks or a different model cask at the ISFSI will require a revision to that evaluation.

The HI-STORM 100 System is designed with a confinement barrier for the radioactive contents to assure that there is no release of radioactive material to the environment under normal operations and off-normal or accident occurrences. Because the confinement boundary (i.e., MPC) remains structurally intact and provides redundant welded closures, the postulated leakage of radioactive material from the confinement boundary was not assumed, consistent with the licensing basis in Chapter 7 of the HI-STORM FSAR and the "leak tight" acceptance criterion for helium leakage testing of the MPC vent and drain port cover plates. Calculations were performed for only direct exposure from the fuel-loaded casks and no effluent dose was considered from the casks (Reference 6.14, Section 4.5). Inhalation (internal) exposures at the ISFSI controlled area boundary due to plant effluents were taken from the Salem/Hope Creek Annual Radiological Effluent Release Report.

The ISFSI dose analysis was performed assuming 200 fully loaded HI-STORM 100S Version B casks in a manner that bounds loading of any fuel authorized by the HI-STORM 100 CoC, with a certain fraction of the casks containing BWR fuel from HCGS and the balance containing PWR fuel from SGS. One hundred eleven (111) of the casks were assumed to contain MPC-68s filled with HCGS spent fuel assemblies and 89 casks were assumed to contain MPC-32s filled with SGS spent fuel assemblies. The 200 casks are stored on three pads as described in Section 5.2.1.1.

Source Terms

The table below shows the fuel-related information used in the dose analyses (Reference 6.14, Section 3.2).

Table 5.3.1-1
Salem and Hope Creek Fuel Source Term Assumptions

PLANT	COOLING TIME (yrs)	AVERAGE BURNUP (MWD/MTU)	ENRICHMENT (wt % ^{235}U)	URANIUM MASS (kg)
SGS	10	57,000	3.8 – 5.0	455
HCGS	5	45,000	3.65	181

Each SGS spent fuel assembly was assumed to include a burnable poison rod assembly (BPRA), which is conservative because it increases the gamma source term. Dose rates from other PWR non-fuel hardware (e.g., thimble plugs, axial power shaping rods, control rod assemblies, etc.) compared to the fuel and BPRAs is considered negligible and is, therefore, not modeled (Reference 6.14, Section 3.2). While the CoC permits loading fuel cooled to as little as three years and with enrichment and burnup higher than that used in this dose analysis, the direct gamma and neutron dose rates computed with these source terms are considered conservative and will bound the measured dose rates.

The fuel and cask loading characteristics are checked during fuel selection to verify that they meet both the CoC and the §72.104 analysis limits. Therefore, SGS and HCGS may load fuel up to the limits for enrichment, decay heat, burnup, and cooling time established in the CoC. The dose analysis performed for use of CoC Amendment 2 at the Salem/Hope Creek ISFSI remains bounding for Amendments 3 and 5 and any fuel authorized by the CoC may be loaded. See Section 7.0 of Reference 6.14 for a more detailed explanation of the bounding nature of the source terms.

Dose Rate Analyses

The computer code MCNP4C3 was used for all dose analyses (Reference 6.14). Two comparative dose rate analyses were performed first to determine the bounding configuration of overpacks containing PWR (MPC-32) and BWR (MPC-68) canisters at the ISFSI. Uniform loading in the MPC was assumed with a homogeneous source modeled.

The direct dose rates were calculated for two cases with a dose receptor at 100 meters to determine whether the BWR or PWR casks on the outside edge of the ISFSI provided the bounding case. In the first case, the 111 overpacks containing MPC-68s were assumed to be arranged on the west side of the ISFSI. In the second case, the 89 overpacks containing MPC-32s were assumed to be arranged on the west side of the ISFSI (see Figures 3 and 4 of Reference 6.14). Per Table 35 of Reference 6.14, the bounding direct dose rate ISFSI configuration occurs when the overpacks containing the 111 BWR MPC-68s are located closest to the dose receptor. Therefore, the BWR and PWR casks may be stored in any configuration at the ISFSI.

The second set of MCNP calculations were performed to determine dose rate versus distance at the nearest controlled area boundary for comparison against the regulatory limit. In addition, dose rates were also calculated in the west direction, for a real individual located at the Delaware River east shoreline for 60 hours per year. In each case, the 111 BWR overpacks were modeled nearest the dose receptor based on the results of the first set of analyses. Dose rates were calculated at various distances from the ISFSI, from one meter to 1000 meters. Direct dose rates, converted to annual doses at the site boundary from the ISFSI, were then added to site boundary effluent and direct doses from plant operation for comparison against the regulatory limits in 10 CFR 72.104. Table 5.1.3-2 below shows the annual dose rates at the locations of interest.

Table 5.3.1-2
Annual Calculated Doses Due to ISFSI and Plant Operations*

LOCATION	CALCULATED WHOLE BODY DOSE (mrem/yr)	WHOLE BODY LIMIT (mrem/yr)	CALCULATED ORGAN DOSE (mrem/yr)	ORGAN DOSE LIMIT (mrem/yr)	CALCULATED THYROID DOSE (mrem/yr)	THYROID DOSE LIMIT (mrem/yr)
NORTH SITE BOUNDARY (full time occupancy)	ISFSI: 1.35E+01	**25**	ISFSI: 0	**25**	ISFSI: 0	**75**
	PLANT: 2.56E-02		PLANT: 1.95E-02		PLANT: 3.43E-01	
	TOTAL: 13.5		**TOTAL: 0.02**		**TOTAL: 0.34**	
EAST SHORE OF DELAWARE RIVER (60 hours/yr occupancy)	ISFSI: 5.35E-01	**25**	ISFSI: 0	**25**	ISFSI: 0	**75**
	PLANT: 2.56E-02		PLANT: 1.95E-02		PLANT: 3.43E-01	
	TOTAL: 0.56		**TOTAL: 0.02**		**TOTAL: 0.34**	

* Reference 6.14, Table 39

The results of the dose evaluation demonstrate that doses at the controlled area boundary due to ISFSI and other fuel cycle facility operations comply with the requirements of 10 CFR 72.104(a). Spent fuel from both SGS and HCGS is assumed to be stored at the ISFSI in the radiological evaluation. The quantities of spent fuel included in the dose analysis are consistent with the quantities of spent fuel currently planned to be removed from each spent fuel pool and stored at the ISFSI. PSEG Nuclear fuel characterization and selection procedures ensure that only fuel authorized by the CoC and also bounded by the dose analysis is loaded into the casks (References 6.31.14, 6.31.15, 6.56.11, and 6.56.12). Furthermore, adherence to the radiation protection program technical specifications in Section 5.7 of Appendix A to the CoC, which require calculation of dose rates for the transfer cask and overpack and measurements of actual dose rates (References 6.31.20 through 6.31.22, 6.56.15, and 6.56.16) for comparison against those calculated values, also provides confidence that the offsite doses will be within those computed in this analysis. Ultimately, the PSEG Nuclear radiological monitoring program, governed by the methodology described in the Hope Creek and Salem Offsite Dose Calculation Manuals (ODCMs), verify that doses at the controlled area boundary, including those due to ISFSI operation do not exceed applicable regulatory limits.

MPC Leak Testing

In 2006, the CoC holder removed a shop helium leak test of the MPC from the HI-STORM FSAR under the provisions of 10 CFR 72.48 and manufactured a number of MPCs without performing the leak test. Eight of those untested MPCs were loaded with HCGS spent fuel in 2008 and are currently being used to store HCGS fuel at the ISFSI. Four additional untested MPCs were delivered for use during the 2010 HCGS loading campaign. In August 2009, the NRC cited the CoC holder for removing the shop helium leak test from the FSAR without prior NRC approval (Reference 6.39). As part of its corrective actions for the violation, the CoC holder committed to restore the test to the FSAR and begin testing all future MPCs. In addition, the CoC holder committed to test all affected MPCs that were delivered to users but not yet loaded, prior to the MPCs being loaded. The four untested HCGS MPCs were successfully leak tested on-site in April 2010 and loaded with HCGS spent fuel in June 2010 (casks 13 through 16). All four MPCs met or exceeded initial fabrication requirements. The four SGS MPCs to be loaded in 2010 were successfully helium leak tested in the fabrication facility and are not affected by this enforcement action.

The eight affected HCGS MPCs in service at the ISFSI (Serial Numbers 1021-147 through -154), while not having been leak tested at the fabrication facility, are still assumed to have no credible leakage. This assumption is consistent with that made by other similarly affected HI-STORM users and is based on the MPC fabrication process (i.e., welding and non-destructive examination, including radiography) being no different for these eight MPCs than the hundreds manufactured before them that <u>were</u> leak tested. In addition, the eight untested MPCs were successfully hydrotested in the plant after fuel loading and MPC lid welding. Lastly, observations of dose over time at the TLDs located near the ISFSI give no indication of any effluent dose contribution from any of the MPCs. Therefore, there is no impact to the 10 CFR 72.104 doses presented in Table 5.3.1-2 as a result of having eight untested MPCs in service. See Reference 6.55 for additional information.

5.3.1.3 Operational Restrictions to meet ALARA Objectives

The Salem and Hope Creek Radiation Protection Programs and procedures have been reviewed in accordance with 10 CFR 72.212(b)(10), and appropriate changes have been made to the implementing procedures to address cask loading operations, transportation of the loaded casks to the ISFSI, and operation of the ISFSI (see Section 5.7 of this report). The Salem and Hope Creek Radiation Protection Programs include appropriate controls to meet as low as reasonably achievable (ALARA) objectives for radioactive materials in effluents and direct radiation exposures during cask loading, cask transport, and ISFSI operations.

5.3.2 Conclusion

PSEG Nuclear has demonstrated by analysis that there is reasonable assurance that the annual dose equivalent to any real individual who is located beyond the ISFSI controlled area boundary during normal operations and anticipated off-normal occurrences will not exceed 10 CFR 72.104(a) dose limits, including doses from other fuel cycle facility operations. PSEG Nuclear implements operational restrictions in conjunction with the SGS and HCGS Radiation Protection Programs to meet ALARA objectives for radioactive materials in effluents and direct radiation levels associated with cask loading and storage operations to ensure that 10 CFR 72.104(a) dose limits will be met. Therefore, PSEG Nuclear complies with the requirements of 10 CFR 72.212(b)(5)(iii).

5.4 10 CFR 72.212(b)(6) — Review of the Cask FSAR and SER

10 CFR 72.212(b)(6) states the following:

"Review the Safety Analysis Report (SAR) referenced in the CoC or amended CoC and the related NRC Safety Evaluation Report, prior to use of the general license, to determine whether or not the reactor site parameters, including analyses of earthquake intensity and tornado missiles, are enveloped by the cask design bases considered in these reports. The results of this review must be documented in the evaluation made in paragraph (b)(5) of this section."

5.4.1 Evaluation

The Salem/Hope Creek ISFSI is located on the Salem/Hope Creek Generating Station site, on which the Salem Units 1 and 2, and Hope Creek reactors are co-located. As part of the process for obtaining the 10 CFR 50 licenses for the three reactors, the characteristics of the site and surrounding area were studied and catalogued in detail, and are well defined. The Salem/Hope Creek ISFSI site parameters are evaluated in this section to ensure that they are enveloped by the HI-STORM 100 FSAR and the NRC Safety Evaluation Report (SER) for the HI-STORM 100 System (References 6.2 and 6.5). The applicable HI-STORM 100 System CoC amendment and FSAR revision, by component serial number, are listed in Appendix 2.

Because the Salem/Hope Creek ISFSI is located much closer to HCGS than SGS, the generic site parameters from the HI-STORM FSAR are compared only to the site parameters described in the HCGS UFSAR. A separate evaluation of the SGS site parameters against the HI-STORM FSAR site parameters would not be meaningful and was not performed, except where events are postulated to occur during cask loading operations taking place at the operating plant or along the heavy haul path. Such events are identified and evaluated, as appropriate, in the following subsections.

The HI-STORM 100 System structures, systems and components important to safety are designed for normal operations and to withstand postulated off-normal and accident events (including natural phenomena) without any unacceptable consequences. The design criteria for the HI-STORM 100 System are given in Chapter 2 of the HI-STORM 100 System FSAR. Accidents are evaluated in Chapter 11 of the HI-STORM FSAR. The parameters identified in the HI-STORM design criteria are evaluated in this section to ensure that they envelope Salem/Hope Creek site-specific conditions. Unique site parameters and events are evaluated as outliers, as required.

5.4.1.1 Fire and Explosion

The HI-STORM 100 System is designed to withstand the effects of fire as described in HI-STORM 100 System FSAR, Sections 2.2.3.3 and 11.2.4, and the effects of explosion as described in FSAR Sections 2.2.3.10 and 11.2.11. Fire and explosion evaluations have been performed for the locations where cask loading and storage operations occur on the Salem/Hope Creek site for comparison against the generic evaluations described in the HI-STORM FSAR. These evaluations are discussed separately below.

Fire

The fire analyses described in the HI-STORM FSAR evaluate the effects of a fire on the HI-STORM overpack and on the HI-TRAC transfer cask, each containing an MPC loaded with fuel at design basis maximum heat load. The fire durations were estimated assuming 50 gallons of transporter fuel distributed in a pool of one meter width around the periphery of the cask. The different diameters of the overpack and transfer cask result in slightly different fire durations (3.6 minutes and 4.8 minutes, respectively). No credit is taken for personnel actions that could suppress the fire.

The generic fire analysis assumes an engulfing fire analyzed with a conservative 1475^{o}F flame temperature for the duration of the fire, which is calculated based on the fuel volume and pool size. This flame temperature is taken from the NRC's radioactive material transportation regulations (10 CFR 71.73(c)(4)) and was found to be acceptable by the NRC for use in Part 72 storage fire analyses (Reference 6.5, Section 11.2.12.2 of the SER for the original HI-STORM CoC). This method of analysis is applicable and bounding for the Salem/Hope Creek site and only a comparison of fuel sources is required for this 72.212 evaluation.

Because the MPC transfer from the HI-TRAC transfer cask to the HI-STORM overpack takes place in either the Hope Creek Reactor Building or the Salem Fuel Handling Building, there is no VCT or prime mover fuel fire threat to the loaded HI-TRAC transfer cask. The fuel tank on the prime mover used to pull the loaded overpack out of the Reactor Building/Fuel Handling Building on the low profile transporter/zero profile transporter is limited to 50 gallons of diesel fuel (Reference 6.43) and is, therefore, bounded by the design basis fire analysis for the overpack described in the HI-STORM FSAR. This design feature also ensures compliance with Section 3.4.5 of Appendix B to the HI-STORM CoC, which limits the "cask transporter" to 50 gallons of diesel fuel, as discussed below.

The VCT fuel volume must be no more than 50 gallons of diesel fuel, in accordance with Section 3.4.5 of Appendix B to the HI-STORM CoC. The amount of fuel in the HCGS VCT's fuel tank is controlled by design. The volume of the fuel tank on the VCT ensures this limit is not exceeded (Reference 6.32.7). The hydraulic fluid in the VCT was also evaluated as a potential source of fuel for the fire. The hydraulic fluid for the VCT has a flash point of 580^{o}F and a fire point of 650^{o}F (Reference 6.49). As such, the VCT hydraulic fluid is considered non-flammable. The Salem ZPT also includes a small amount of hydraulic fluid (less than 5 gallons per Reference 6.70). The MSDS for the ZPT hydraulic fluid indicates that the flash point of the hydraulic fluid is 400^{o}F, which also qualifies as non-flammable. Lastly, a gasoline-powered auxiliary electric generator may be used during cask preparation activities conducted in the SGS FHB.

Based on the National Fire Protection Association's guidance, liquids with flash points greater than 100°F are considered non-flammable (Reference 6.38). Liquids with flash points above 200°F, such as the VCT and ZPT hydraulic fluids, are considered Class IIIB combustible liquids (Reference 6.38). While these non-flammable materials may not be the source of a fire, they can burn in the presence of an existing fire from another source. Given the proximity of these hydraulic fluids and the gasoline to a loaded cask in the ZPT, a supplement to Reference 6.57 was performed.

The supplement addresses the potential fire impact of these additional petroleum-based fluids used with the VCT and ZPT, and establishes a set-back requirement for staging the gasoline-powered auxiliary electric generator. The VCT contains a total of approximately 375 gallons of petroleum products, comprised of 45 gallons of diesel fuel (the VCT fuel tank actual size) and a combined total 330 gallons of hydraulic fluid, motor oil, and transmission fluid (Reference 6.57). The Reference 6.57 supplemental evaluation states that a pool fire around the cask that consumes 400 gallons of fuel, consisting of 50 gallons of diesel fuel and 350 gallons of hydraulic fluid, will have only a minor effect on the cask components and spent nuclear fuel temperature. This analysis bounds the actual 375 gallon value for the SGS arrangement and is, therefore, acceptable.

At HCGS, the VCT receives the loaded HI-STORM overpack outdoors and just south of the Reactor Building receiving bay door. The VCT moves a short distance south, then west, then north along the heavy haul path to the ISFSI site. Both the ISFSI facility and the heavy haul path are evaluated as new fire zones in Reference 6.11. There are no fixed sources of combustible material along the heavy haul path fire zone and no transient combustible material is permitted to be stored at the ISFSI (Reference 6.11, Section 7.2).

At SGS, the VCT receives the loaded HI-STORM overpack outdoors and just west of the Fuel Handling Building of the unit from which fuel is being moved to the ISFSI. The VCT moves a short distance farther west, then north along a new segment of the heavy haul path until it joins the existing heavy haul path at HCGS and follows the existing heavy haul path to the ISFSI. Elevations 100 ft. and 130 ft. of the Fuel Handling Buildings, the egress pads, and the new segment of the heavy haul path have been evaluated in the SGS fire hazards analysis (Reference 6.57)

There are no automatic fire suppression or detection systems at the ISFSI or along the heavy haul paths. As required by procedures (References 6.31.3, 6.31.8, 6.56.3, and 6.56.8), personnel trained in fire protection will accompany the cask during on-site transport between the HCGS Reactor Building or SGS Fuel Handling Building and the ISFSI, along with portable fire suppression equipment, to extinguish any fires that may occur before they jeopardize the cask system. Spatial separation between the overpacks and the fire hazards in the yard area is the primary means of minimizing the effects of fire on the fuel-loaded overpacks. There are no rated or non-rated fire barriers provided in the yard area near the ISFSI or along the heavy haul path for protection of the overpacks. A pre-cask loading campaign walkdown is performed to ensure no unanalyzed transient or permanent combustibles have been located near the heavy haul path (References 6.31.3, Attachment 3 and 6.56.3, Attachment 3).

ISFSI

Reference 6.11, Section 7.1, evaluates combustible materials and other potential fire hazards in and around the ISFSI. Reference 6.15 determines the thermal effects of the sources of combustion. The ISFSI is a fenced-in area comprised of the three concrete pads, surrounding gravel, loaded and unloaded overpacks, and conduit and control boxes associated with the overpack temperature monitoring instrumentation and security equipment. Cabling in rigid metal conduit exists, but does not contribute to the combustible load for the ISFSI fire area. There are no mechanical piping systems carrying potentially combustible, flammable, or explosive fluids either above ground or underground at the ISFSI. Transient combustibles at the ISFSI are controlled by procedure (Reference 6.54.1). Fixed and transient combustibles in the yard area near the ISFSI have been evaluated and found to be acceptable.

The ISFSI Electrical Interface/Security Building is located within the ISFSI and contains numerous cables and electrical cabinets. No fire detection or alarm system is installed in this building. However, portable fire extinguishers are located there. Although a large fire in the Electrical Interface/Security Building is unlikely due to the limited quantity of combustibles and fire detection and protection controls, an evaluation of the effect on the casks of a fire in this building was performed. The results were acceptable provided the building is at least 20 feet away from the nearest overpack (Reference 6.11), which it is. Other yard area buildings in the vicinity of the ISFSI were also evaluated for fire impact on the ISFSI and found to be acceptable.

Heavy Haul Path

Reference 6.11, Section 7.2 identifies the combustible materials and other potential fire hazards near the heavy haul path from HCGS to the ISFSI. Reference 6.15 determines the thermal effects of the sources of combustion. Permanent and transient fire hazards due to structures, tanks, and other fire sources near the heavy haul path have been evaluated and found to be acceptable. Reference 6.57 identifies the combustible materials and other potential fire hazards near the heavy haul path from the SGS egress pads to the point at which it joins the existing heavy haul path near Hope Creek.

There are no mechanical piping systems that carry combustible, flammable, or explosive fluids routed either above ground or below ground near the SGS heavy haul path except a single, underground, 4-inch diameter fuel oil pipeline. Because of its underground location and the fact that the heavy haul path was designed to protect all underground commodities over which it traverses, this fuel oil line was not considered a credible source of fire or a threat to a cask passing above it.

Other permanent and transient sources of fire near the SGS heavy haul path and in the adjacent yard area are identified and evaluated in Reference 6.57. Reference 6.15 determines the thermal effects of the sources of combustion. Permanent and transient fire hazards due to structures, tanks, and other fire sources near the heavy haul path have been evaluated. These sources were found not to be a significant threat to a cask being transported by the VCT on the heavy haul path based on the location and nature of the fire source, fire protection features, and administrative controls that which minimize the probability and effects of fires.

Explosion

No particular explosion analysis was performed generically for the HI-STORM 100 System because of the difficulty in determining a generic explosion hazard that would bound most or all ISFSI sites. Instead, the HI-STORM 100 System MPC and overpack are designed for specific external pressures that are compared to the site-specific explosion hazards, if any. The MPC is designed for 60 psig external pressure and the ventilated HI-STORM overpack is designed for 10 psig instantaneous and 5 psig steady-state external pressure (Reference 6.2, Table 2.2.1).

References 6.15 and 6.57 address the overpressure effects of the sources of explosion on the HI-STORM overpack as it travels along the heavy haul path and during storage operations at the ISFSI. The sources of explosion include hydrogen, gasoline, diesel fuel and fuel oil in storage containers and in parked and driven vehicles. All calculated overpressures are less than 1.0 psig (References 6.15, Table 6.1-1, and 6.57). Therefore, the design basis external pressures for the MPC and overpack are not exceeded.

5.4.1.2 Tornado Wind and Missiles

Tornado Wind

The HI-STORM 100 System is designed to withstand pressures, wind loads, and missiles generated by a tornado as described in HI-STORM 100 System FSAR, Sections 2.2.3.5 and 11.2.6. HI-STORM FSAR Table 2.2.4 provides the wind speeds and pressure drops that the HI-STORM overpack is designed to withstand while maintaining kinematic stability.

The generic HI-STORM 100 System design basis tornado has a rotational wind speed of 290 MPH and a translational wind speed of 70 MPH for a total effective wind speed of 360 MPH. The assumed pressure drop due to a tornado is 3.0 psi. These design criteria are consistent with those specified for Region I sites in Regulatory Guide (RG) 1.76 (Reference 6.3). The HCGS-specific design basis tornado wind characteristics are described in HCGS UFSAR Section 3.3.2.1 and are identical to the generic HI-STORM 100 System design values. The design basis for tornado wind at SGS is 360 MPH, comprised of a 300 MPH peripheral velocity and a 60 MPH translational velocity as discussed in SGS UFSAR Section 3.3.2.1. Therefore, the generic tornado wind design criteria and analysis bounds the site specific tornado design basis winds and both HCGS and SGS.

Tornado Missiles

HI-STORM FSAR Table 2.2.5 provides the tornado missile data used in the analysis of the HI-STORM overpack and HI-TRAC transfer cask. Information from HI-STORM FSAR Table 2.2.5 is repeated in the following table:

Table 5.4.1.2-1
HI-STORM 100 System Design Basis Tornado Missiles

Missile Description	Mass (kg)	Velocity (MPH)
Automobile (large missile)	1,800	126
Rigid solid steel cylinder (8 inch diameter) (intermediate missile)	125	126
Solid sphere (1 inch diameter) (small missile)	0.22	126

These postulated tornado missiles are consistent with the "Spectrum I" missiles in NRC NUREG-0800, Standard Review Plan (SRP) 3.5.1.4 (Reference 6.21). The large missile was evaluated for its ability to tip over the cask. The intermediate missile was evaluated for penetration through the overpack. The small missile was evaluated for damage due to its passage through a penetration in the overpack (i.e., an inlet or outlet air duct).

The Salem/Hope Creek Generating Station site is located in tornado Region I as defined in RG 1.76. The missiles listed in Table 5.4.1.2-1 above are consistent with the "Spectrum II" missiles for tornado Region I listed in SRP 3.5.1.4. SRP 3.5.1.4 permits the use of Spectrum I or II missiles in performing design work. Therefore, the generically analyzed missiles are appropriate examples of the types of tornado missiles that could impact the dry storage casks in-transit to, or at the Salem/Hope Creek ISFSI. The NRC's Safety Evaluation Report (SER) for the HI-STORM 100 System CoC (original issue) states in Section 3.4.2 (Reference 6.5):

> *"The staff concludes that the tornado and tornado missile analyses are adequate and acceptable. The phenomena analyzed are considered to envelop the corresponding phenomena at all points on U.S. territory."*

This SER statement applies to the original HI-STORM 100 overpack design, which is not being used at the Salem/Hope Creek ISFSI. In SER Section 3.4.2.2 for HI-STORM CoC Amendment 1, the NRC affirms that the tornado missile analysis performed for the HI-STORM 100 overpack design remains bounding for the HI-STORM 100S overpack design. The HI-STORM 100S Version B overpack design being used at the Salem/Hope Creek ISFSI, which is a variation of the HI-STORM 100S overpack design, was authorized by Holtec under the provisions of 10 CFR 72.48. The 10 CFR 72.48 evaluation for the HI-STORM 100S Version B overpack design concludes that the Version B overpack design continues to provide adequate tornado missile protection as discussed in HI-STORM FSAR Section 3.4.8.1. Therefore, the HI-STORM 100S Version B overpack design is governed by the CoC Amendment 1 SER statement approving the tornado missile protection design features of the 100S overpack design.

<u>Hope Creek Tornado Missile Evaluation</u>

The HCGS site-specific design basis tornado missile characteristics are described in HCGSUFSAR Table 3.5-12 and are repeated in the following table. Weights (masses) have been converted from pounds to kilograms and velocities have been converted from feet per second to miles per hour for comparison with the generic HI-STORM System design basis missiles in Table 5.4.1.2-1. The missiles listed in Table 5.4.1.2-2 below are consistent with the "Spectrum II" missiles for tornado Region I listed in SRP 3.5.1.4.

Table 5.4.1.2-2
Hope Creek Site Design Basis Tornado Missiles

Missile Description	Mass (kg)	Velocity (MPH)
Automobile	1,814	131.7
Utility Pole	511	122.3
12-inch diameter Schedule 40 pipe	340.7	104.9
6-inch diameter Schedule 40 pipe	130.3	116.3
Wood Plank	52.1	185.7
1-inch diameter steel rod	4.0	114.1

Based on a comparison of the quantity mv^2/d, where 'm' is the mass of the missile, 'v' is the velocity of the missile, and 'd' is the equivalent diameter of the missile, the automobile and the 8-inch diameter steel cylinder analyzed generically in the HI-STORM FSAR are not bounding evaluations for Hope Creek. Therefore, a site-specific analysis of the Hope Creek large and intermediate missiles was performed (Reference 6.50). The results of that analysis are:

- The large missile will not cause the cask to tip over
- The intermediate missile will not penetrate the one-inch outer steel shell of the overpack
- The intermediate missile will not penetrate the 3-inch vent shield lid of the overpack
- Away from impact locations, the stresses in the overpack are less than ASME Code Level D stress limits

The results of the site-specific tornado missile analysis show that the HCGS spent fuel is adequately protected and the overpack will not tipover. Therefore, the HCGS tornado missile analysis is acceptable. A tornado missile evaluation for the HI-TRAC transfer cask at HCGS is not required because a fuel-loaded transfer cask never leaves the Reactor Building.

<u>Salem Tornado Missile Evaluation</u>

The Salem site-specific design basis tornado missile characteristics are described in Salem UFSAR Section 3.5.2.1 and are repeated in the following table. Weights (masses) have been converted from pounds to kilograms and velocities have been converted from feet per second to miles per hour for comparison with the generic cask design basis missiles in Table 5.4.1.2-1

Table 5.4.1.2-3
Salem Site Design Basis Tornado Missiles

Missile Description	Mass (kg)	Velocity (MPH)
Automobile	1818	132
Utility Pole 1	713	150
Utility Pole 2	677	144(H) 115(V)
1-inch diameter steel rod	3.6	215(H) 172(V)

Based on a comparison of the kinetic energy $mv^2/2$, where 'm' is the mass of the missile and 'v' is the velocity of the missile, the automobile missile has similar effects as the HCGS automobile missile. However the utility poles and steel rod missiles are not bounded by the analyzed HCGS missiles. Therefore, a site-specific analysis of the SGS utility pole (intermediate size) and steel rod (small size) missiles was performed (Reference 6.63). The results of that analysis are:

- The intermediate missile will not cause the cask to tip over
- The intermediate and small missiles will not penetrate the one-inch outer steel shell of the overpack
- The intermediate and small missiles will not penetrate the 3-inch vent shield lid of the overpack
- Away from impact locations, the stresses in the overpack are less than ASME Code Level D stress limits

The results of the bounding HCGS missile analyses and the SGS site-specific tornado missile analyses show that the SGS spent fuel is adequately protected and the overpack will not tipover and will not be fully penetrated. Therefore, the SGS tornado missile analysis is acceptable. A tornado missile evaluation for the HI-TRAC transfer cask at SGS is not required because a fuel-loaded transfer cask never leaves the Fuel Handling Building.

5.4.1.3 Flood

The HI-STORM 100 System is designed to be capable of withstanding pressure and moving water forces associated with a flood as described in HI-STORM FSAR Sections 2.2.3.6 and 11.2.7. Table 2.2.8 of the HI-STORM 100 System FSAR shows that the MPC enclosure vessel is designed for a 125 foot static head of water without collapsing, buckling, or otherwise allowing water to intrude into the confinement boundary. The cask system, including the overpack and the MPC is also designed to withstand the forces of flood water up to a velocity of 15 feet per second without sliding or tipping over.

Based on Section 2.4 of the Hope Creek UFSAR, PSEG plant datum is 89 feet above Mean Sea Level (MSL). UFSAR Table 2.4-6 indicates that the maximum still water level at the power block due to the probable maximum hurricane is 24.8 ft. MSL, or 113.8 ft. PSEG plant datum.

According to Section 3 of Reference 6.45, elevation 113.8 ft. PSEG plant datum equals elevation 23.96 ft. North American Vertical Datum (NAVD) 1988 Datum, the frame of reference on which the ISFSI flood analyses are based.

The top of the ISFSI pad is located at elevation 15.0 ft. NAVD 1988 Datum (References 6.32.8 and 6.45) or 104.84 ft. PSEG plant datum. The HI-STORM 100S-218 Version B overpack is 18 feet 2-1/2 inches tall (Reference 6.32.6), placing the top of the cask at just over elevation 33 feet NAVD 1988 Datum when on the ISFSI pad. The HCGS site-specific design basis flood results in flood water to an elevation of about elevation 24 ft. NAVD 1988 Datum, as discussed above. Therefore, the design basis flood would submerge about half of the height of the cask. Because the depth of submergence is less than 125 ft., the MPC confinement boundary will remain intact by design.

The design basis flood would block all air flow through the overpack until such time as the flood waters recede and uncover the inlet air ducts. Section 3.4.9 of Appendix B to the CoC provides the requirements for evaluating site-specific flooding events. The cooling provided by the water at maximum flood height being in contact with the MPC outer shell would compensate for the loss of air flow.

An evaluation of an external flood was performed to determine the maximum water level at the ISFSI due to a probable maximum hurricane and the duration that the water level would be above the ISFSI grade (Reference 6.17). This information was used to estimate the amount of time the overpack inlet vents would be submerged by flood water. This evaluation, performed before the ISFSI design was finalized, considers an ISFSI elevation of 102.5 feet PSEG plant datum, which is lower than the actual ISFSI as-built elevation of about 105 feet and is, therefore, a conservative evaluation. The total duration of the event is estimated to be 24 hours.

The HI-STORM FSAR includes, as an accident condition, the total blockage of all inlet air ducts. Section 11.2.13 and Table 11.2.9 of Reference 6.2 show that the fuel cladding and all cask component temperatures remain below their respective accident allowables for up to 72 hours with full air vent blockage, which provides a bounding case for the 24-hour flood at the Salem/Hope Creek ISFSI. Therefore, the HCGS site-specific design basis flood event is bounded by the generic blocked duct analysis and does not jeopardize safe spent fuel storage at the ISFSI.

The forces on the storage casks at the ISFSI caused by moving floodwater from a Probable Maximum Hurricane (PMH) have been calculated. The calculated water flow velocity is 4.38 ft/sec (Reference 6.45, Section 5.0.B). This value is less than the design basis value of 15 ft/sec per Reference 6.1, Appendix B, Section 3.4.4 and is, therefore, acceptable.

5.4.1.4 Tsunami and Hurricane

Tsunami

The Salem/Hope Creek Generating Station site in southern New Jersey is not located near the ocean and is not otherwise located in an area subject to significant magnitude tsunamis due to the relatively low seismicity of the northern Atlantic Ocean, the Northeast United States in general, and the southern New Jersey area in particular. HCGS UFSAR Table 2.4-6 states that the maximum tsunami wave height is 18.1 ft. MSL, or 107.1 ft. NAVD. Therefore, the flood effects of tsunami water height are bounded by the design basis flood water height discussed in Section 5.4.1.3 above.

Hurricane

Wave action on the cask resulting from a postulated design basis hurricane moving up the Delaware Bay has been evaluated as a site-specific hazard (References 6.45 and 6.48). The factors of safety against sliding and overturning due to wave action are 1.43 and 3.18, respectively, per Reference 6.48. Therefore, the casks will not slide or turn over due to design basis wave action at the ISFSI caused by the probable maximum hurricane.

5.4.1.5 Earthquake

The HI-STORM 100 System is required to withstand loads due to a seismic event in accordance with HI-STORM 100 System FSAR, Sections 2.2.3.7 and 11.2.8. In particular, the HI-STORM overpack is required to resist overturning and sliding on the ISFSI storage pad due to a seismic event. The inequality shown in HI-STORM FSAR Table 2.2.8 is not met for the Salem/Hope Creek ISFSI site seismic accelerations. However, the cask is shown not to slide or tip over using the alternative time-history analysis permitted by the CoC and described in Section 5.2.1.5 of this report. The ISFSI pad is designed to withstand the dynamic effects of an earthquake, including soil-structure interaction and soil liquefaction, as discussed in Sections 5.2.1.2 and 5.2.1.6 of this report. Therefore, the free-standing HI-STORM 100 System and the ISFSI pad are qualified for use at the Salem/Hope Creek ISFSI.

5.4.1.6 Lightning

HI-STORM 100 System overpacks are stored on an unsheltered ISFSI storage pad. The Salem/Hope Creek ISFSI is located adjacent to the HCGS Unit 1 cooling tower, which is over 500 feet high, compared to the cask height of approximately 18 feet. Therefore, the cooling tower is the most probable lightning strike target in the vicinity of the ISFSI and a strike on a cask at the ISFSI is unlikely. Nevertheless, there is a small potential for lightning to strike the HI-STORM overpacks. Sections 2.2.3.11 and 11.2.12 of the HI-STORM 100 System FSAR address the lightning strike as an accident event. The HI-STORM FSAR indicates that the HI-STORM overpack steel outer shell provides a direct path to ground, and the cask can safely conduct lighting strikes without the need for any supplemental protection against lightning strikes. Because of the mass of steel in the overpack, there is adequate protection for the MPC and the confinement boundary is unaffected.

Administrative controls are used to prohibit cask transportation on-site during severe weather (Reference 6.31.3). Therefore, while the cask system is designed to withstand a lightning strike, a lightning strike on the cask transporter while carrying a loaded HI-STORM overpack between the fuel building and ISFSI is considered very unlikely.

5.4.1.7 Burial Under Debris

Section 2.2.3.12 of the HI-STORM 100 System FSAR states that "the HI-STORM System must withstand burial under debris," and "siting of the ISFSI pad shall ensure that the storage location is not located near shifting soil." Section 5.2 of this report discusses the ISFSI pad design and the subsoil, including liquefaction, and finds the Salem/Hope Creek ISFSI pad design acceptable. The above-referenced HI-STORM FSAR section also states that "such debris may result from floods, wind storms, or mud slides." It goes on to state that "mud slides, blowing debris from a tornado, or debris in flood water may result in duct blockage, which is addressed in Subsection 2.2.3.13 [100% Blockage of Air Inlets]." Short term cask duct blockage can be caused by a flood at the Salem/Hope Creek ISFSI and is addressed in Section 5.4.1.3 of this report.

Section 11.2.14 of the HI-STORM FSAR describes the evaluation of the burial under debris accident and states that "burial of the HI-STORM System under debris is not a credible event" and provides justification for this statement. However, complete thermal isolation of the cask is analyzed in the HI-STORM FSAR assuming the material covering the cask acts as a perfect insulator and the contents of the cask undergo a transient heat up under adiabatic conditions.

The Salem/Hope Creek ISFSI is sited on a man-made island, surrounded by the Delaware River, that does not have any nearby mountains, significantly sized hills, mounds of soil, or other accumulated debris that could cause a burial-under-debris event. There are no active volcanoes in this region of the world. Therefore, complete burial of one or more of the casks under debris at the Salem/Hope Creek ISFSI due to these events is not credible.

No analysis has been performed to determine the potential for debris build-up around the casks as a result of a flood or tornado. However, any debris that would build up due to a flood or tornado at the ISFSI would likely not completely cover the casks in a thermally insulating manner, based on engineering judgment. In the case of the flood, this is supported by the fact that the maximum flood water only reaches about the half height of the cask, per Section 5.4.1.3. For a tornado, the debris field would be comprised of various types, sizes, and shapes of items such as loose lumber, trees, and other small, solid objects. Based on these arguments, debris from a flood or tornado would not create the complete coverage and thermal insulation of the casks as described in the HI-STORM accident analysis. Therefore, the burial under debris accident analyzed in the HI-STORM FSAR is bounding for the casks at the Salem/Hope Creek ISFSI.

5.4.1.8 Environmental Temperatures

5.4.1.8.1 Minimum Air Temperature During Handling Operations

Section 2.2.1.2 of the HI-STORM 100 System FSAR specifies that handling operations of the loaded HI-TRAC transfer cask or HI-STORM overpack is limited to working area ambient temperatures greater than or equal to 0°F. This limitation is specified to ensure that a sufficient safety margin exists before brittle fracture might occur during handling operations with the HI-TRAC transfer cask. At HCGS, a site-specific component (the HERMIT[4]), used to prevent cask tipover inside and for a short distance just outside the Reactor Building during a seismic event (see Reference 6.34 for details), also has a minimum operating temperature requirement of 0°F. In addition, at SGS, the ZPT has a minimum operating temperature of 0°F (Reference 6.71, Section 5.15). Applicable cask loading and handling procedures (References 6.31.3, 6.31.5, 6.31.7, 6.31.8, 6.31.9, 6.56.3, 6.56.5, 6.56.7, 6.56.8, and 6.56.9) ensure that the working area temperatures are within the required ranges for the components involved in the particular cask handling operations.

[4] "HERMIT" is an acronym for Holtec EaRthquake MITigator.

5.4.1.8.2 Normal Air Temperature

The HI-STORM 100 System was designed and analyzed assuming a specific ambient temperature in order to establish a normal condition thermal design basis for the storage system that ensures long-term fuel integrity. The design basis normal ambient temperature (annual average) for the HI-STORM 100 System is 80°F per HI-STORM 100 System FSAR Section 2.2.1.4, Table 2.2.2, and Section 3.4.1 of Appendix B to the HI-STORM CoC. Short-term daily exceedance of this value (e.g., during summer months) is acceptable because the thermal inertia of the cask system is so large that it precludes any significant effect on the fuel caused by these daily temperature swings. The annual mean ambient temperature at the HCGS site is 11.7°C (53.1°F) per HCGS UFSAR Table 2.3-9. Therefore, the design basis normal ambient temperature for the HI-STORM 100 System bounds the site value.

The normal ambient (annual average) temperature for operations involving the HI-TRAC transfer cask is not permitted to exceed 80°F per Table 2.2.2 of the HI-STORM 100 System FSAR. This value bounds the annual site mean temperature of 53.1°F and is therefore, acceptable. Administrative controls ensure that the working area ambient temperature inside the HCGS Reactor Building and SGS Fuel Handling Building remain less than 100°F during transfer cask operations. Ambient temperatures during all phases of cask loading operations, both indoors and outdoors, are checked in accordance with procedures (References 6.31.3, 6.31.5, 6.31.7, 6.31.8, 6.31.9, 6.56.3, 6.56.5, 6.56.7, 6.56.8, and 6.56.9).

5.4.1.8.3 Soil Temperature

Section 2.2.1.4 and Table 2.2.2 of the HI-STORM 100 System FSAR limit the annual average normal soil temperature to 77°F. In the thermal analysis, this is the temperature assumed for the soil underneath the ISFSI pad (Reference 6.2, Section 2.2.1.4). This temperature was chosen by Holtec as a conservative maximum value based on the average annual soil temperature for Key West, Florida. It is conservative because lower soil temperature would result in higher heat transfer from the cask through the ISFSI pad. The location of the Salem/Hope Creek Generating Station site is over 1,000 miles north of Key West, Florida. Therefore, by simple geographic comparison, the average annual soil temperature below the ISFSI pad at Artificial Island, NJ is lower than that of Key West. However, a second check was also performed to try to quantify this value, as described below.

No site-specific soil temperature is available for the Salem/Hope Creek ISFSI site. Because the Salem/Hope Creek ISFSI pad is approximately 36 inches thick, the ground water temperature for the appropriate region of the United States was used for comparison with the limit. Approximate groundwater temperature in the Salem/Hope Creek Generating Station area is in the mid- 50 degrees Fahrenheit (Reference 6.27, pp. 31.18 – 31.20), which is well below 77°F and is, therefore, acceptable for meeting this soil temperature limit.

5.4.1.8.4 Off-Normal Environmental Temperature

The HI-STORM 100 System is designed to withstand the effects of off-normal environmental temperatures as described in HI-STORM FSAR Section 11.1.2. Table 2.2.2 of the HI-STORM 100 System FSAR specifies upper and lower bound off-normal temperature limits for the HI-STORM overpack and the HI-TRAC transfer cask. The upper bound off-normal temperature limit for the HI-STORM overpack, and for the HI-TRAC transfer cask, is defined as a 3-day average maximum ambient temperature of 100°F. The lower bound off-normal ambient temperature is -40°F for the overpack and 0°F for the transfer cask.

The maximum measured hourly temperature at the HCGS site is 34.5°C (94.1°F) per HCGS UFSAR Table 2.3-9. Therefore, the design basis maximum off-normal ambient temperature for the HI-STORM 100 System bounds the site value. The minimum measured hourly temperature at the HCGS site is -18.5°C (-1.3°F) per HCGS UFSAR Table 2.3-9. Therefore, the design basis minimum off-normal ambient temperature for the HI-STORM overpack bounds the site value.

5.4.1.8.5 Extreme Environmental Temperature

The HI-STORM 100 System is designed to withstand extreme environmental temperatures as an accident condition as described in HI-STORM FSAR Section 11.2.15. The accident level environmental temperature (3-day average) for the HI-STORM overpack is 125°F per Section 2.2.3.14 and Table 2.2.2 of the HI-STORM 100 System FSAR. The maximum measured hourly temperature at the HCGS site is 34.5°C (94.1°F) per HCGS UFSAR Table 2.3-9. Therefore, the design basis maximum extreme ambient temperature for the HI-STORM 100 System bounds the site value.

5.4.1.9 Snow and Ice

The HI-STORM 100 System is designed to withstand pressure loads due to snow and ice. Section 2.2.1.6 and Table 2.2.8 of the HI-STORM 100 System FSAR state that the HI-STORM 100 System is designed for snow pressure loading of 100 pounds per square foot (psf). From Section 2.3.1.2.2 of the HCGS UFSAR, the extreme snow load on the ground at HCGS site is 123 psf. Therefore, a site-specific evaluation of the 123 psf snow loading on the cask was performed (Reference 6.50). The results of this evaluation show that this snow load on the overpack body and lid is bounded by the loads imposed by the transfer cask stack-up during MPC transfer operations and the design-basis vertical end drop, respectively, and is therefore, acceptable. HCGS UFSAR Tables 2.3-2 and 2.3-3 indicate that the maximum measured snowfall in the area is on the order of two feet. A snow event of this magnitude in southern New Jersey is not common. Even several major snow storms in series would result in less than the equivalent snow loading used in the generic analysis. Lastly, due to the heat generated by the fuel, the overpack lid top plate remains warm (190°F for design basis heat load per Table 4.4.7 of Reference 6.2) and will prevent the accumulation of any significant amount of snow or ice on top of the cask.

A significant snowfall event could result in an accumulation of snow in front of the air inlet ducts at the bottom of the overpack. Procedural guidance is in place (Reference 6.31.24, Condition M) to monitor snow accumulation and remove snow to prevent any extended duration blockage of the air inlet vents.

An evaluation of the effect of ISFSI pad icing per NRC Information Notice 2003-16 (Reference 6.4) has been performed and is summarized in Section 5.2.1.5 of this report.

5.4.1.10 Cask Transport Route (Heavy Haul Path)

The HCGS portion of the heavy haul path is shown in Section 8.7 of Reference 6.31.3.

<u>HCGS Heavy Haul Path</u>

The loaded HI-STORM overpack, resting atop the HERMIT device and low profile transporter (LPT), exits the HCGS Reactor Building through the receiving bay door to the south. The LPT includes a number of Hilman rollers to facilitate movement along two parallel rails running from inside the receiving bay to the egress pad just outside the Reactor Building receiving bay door. The LPT is pulled to the egress pad by a prime mover (similar to an airplane "tugger" vehicle).

The overpack is moved to the egress pad and the overpack lid is installed. The cask is attached to the VCT at a point approximately 50 feet south of the Reactor Building receiving bay door. The VCT is a tracked vehicle that lifts the overpack off the ground only as high as necessary to clear any undulations in the haul path between the Reactor Building and the ISFSI pad. The VCT turns 90 degrees west and travels for approximately 100 feet and turns 90 degrees north. At this point, it travels straight to the ISFSI and to the pre-determined pad location for each cask. The normal speed for the transporter is 0.4 MPH (Reference 6.32.7).

The heavy haul path is primarily an asphalt roadway with concrete turning pads at certain locations, including an "egress pad" south of the Reactor Building receiving bay door. Between the Reactor Building and the egress pad, the path is designed to support the loaded overpack on either the LPT (with the HERMIT) or suspended from the cask transporter. From the egress pad to the ISFSI, the roadway is designed to support the weight of a loaded overpack suspended from the VCT. The maximum road pressure from the tracked VCT is 50 lb/in^2 (Reference 6.28). The heavy haul path has been appropriately designed for this pressure load and the expected number of VCT and semi-truck trips over the life of the ISFSI (Reference 6.35).

<u>SGS Heavy Haul Path</u>

During transfer of the loaded MPC from the HI-TRAC transfer cask to the HI-STORM overpack, the loaded overpack rests on the truck bay floor but inside the collar of the ZPT (see Reference 6.71, page 12). The ZPT lifts the HI-STORM overpack using a hydraulic system which powers four structural "tabs" that engage the air inlet ducts at the bottom of the cask. The non-powered ZPT is designed to lift the overpack just high enough to clear the floor and be moved along two parallel rails running from inside the truck bay to the egress pad to the west, just outside the Fuel Handling Building truck bay door. The ZPT is pulled out of the Fuel Handling Building to the egress pad by a prime mover and a suitably designed tow bar.

The overpack lid is installed at the egress pad. The overpack is attached to the VCT at a point approximately 50 feet west of the Fuel Handling Building truck bay door. The VCT lifts the overpack off the ground only as high as necessary to clear any undulations in the haul path between the Fuel Handling Building and the ISFSI pad (Reference 6.56.3). The VCT turns 90 degrees north from the egress pad and travels to the HCGS heavy haul path. From this point, it travels to the ISFSI and to the pre-determined pad location for each cask. The VCT with the overpack travels at a normal speed of 0.4 MPH (Reference 6.32.7).

The heavy haul path is primarily an asphalt roadway with concrete turning and protective pads at certain locations, including an egress pad west of the Fuel Handling Building truck bay door. The egress pads west of each Fuel Handling Building door are designed to support the loaded overpack on either the ZPT or suspended from the VCT. From the egress pad to the Hope Creek heavy haul path, the roadway is designed to support the weight of a loaded overpack suspended from the VCT. The maximum road pressure from the tracked VCT is 50 lb/in^2 (Reference 6.28). The SGS portion of the heavy haul path has been appropriately designed for this pressure load and the expected number of VCT trips (Reference 6.35).

5.4.1.10.1 Cask Movement to the Egress Pad

During its movement from the Reactor Building or the Fuel Handling Building to the egress pad, the overpack is supported by the LPT and HERMIT (at HCGS) or the ZPT (at SGS), and is pulled by the prime mover along rails embedded in the floor of the Reactor Building receiving bay or the Fuel Handling Building truck bay and the egress pads just outside. During this movement, the overpack lid is not installed. Analyses have been performed (References 6.47 6.64, and 6.69) to verify that the overpack will remain stable and will not tip over as a result of dynamic events while located on the LPT or ZPT.

In all cases, the analyses demonstrate that the overpack will not tip over, preventing discharge of the fuel-loaded MPC. The minimum pressure pulse required to cause tipover of the cask on the LPT/HERMIT was calculated to be 104,250 lb-sec. This value bounds all explosive overpressure events at the egress pads for HCGS and SGS (using the ZPT rather than the LPT), and is therefore acceptable.

While resting inside the ZPT collar, the air flow into the overpack air inlet ducts at the bottom of the cask provide is partially obstructed. However, Holtec has determined that this partial obstruction to cooling air flow is bounded by the off-normal event where half of the air inlet ducts are assumed to be blocked (Reference 6.75). The off-normal event is analyzed as a steady state condition and all component temperatures remain within their allowables for an unlimited amount of time.

PSEG has also evaluated the condition where the loaded HI-STORM overpack is not immediately moved from the SGS FHB truck bay to the egress pad. In this situation, the mating device is left installed on the overpack to provide shielding until the overpack is moved out of the FHB. Holtec has instructed that the mating device drawer be left open at least one foot during this time to provide an equivalent amount of airflow to that occurring when the HI-STORM lid is installed, thus ensuring adequate heat transfer from the MPC during this time (Reference 6.67). It is also recognized that the HI-STORM overpack will be located in the ZPT during this delay time. References 6.67 and 6.75, together, ensure adequate heat transfer is occurring during this evolution until such time as the overpack is moved outside, the HI-STORM lid is installed, and the cask is moved to the ISFSI.

5.4.1.11 Collapse of Site Structures

Potential interactions between site structures near the ISFSI and along the heavy haul path and the DCS storage casks have been evaluated (Reference 6.44). There are two tornado or earthquake-induced collapse hazards evaluated for the ISFSI pad: 1) collapse of a utility pole or high mast light pole and 2) collapse of the HCGS cooling tower. There are no collapse hazards identified or evaluated along the HCGS heavy haul path. For SGS, an evaluation of the potential collapse of site structures along the heavy haul path was performed (Reference 6.44) and concluded that the HI-STORM stability and local penetration resistance are adequate to withstand an impact load resulting from the structural failure of site structures along the heavy haul path (i.e. controlled facility building, high mast light pole, and standard light pole).

5.4.1.11.1 High-Voltage Transmission Line Towers

The on-site heavy haul path and the ISFSI are located on the opposite side of the power plant from the switchyard and associated transmission towers and high voltage power lines. Therefore, there is no threat of a transmission tower collapse or transmission line drop onto a loaded overpack.

5.4.1.11.2 Effect of a Postulated Collapse of Structures Near the ISFSI Pad

The consequences of the collapse of a utility pole or high mast light pole onto a HI-STORM cask is enveloped by the cask's design basis tornado missile evaluation and is, therefore, acceptable. The consequences of a postulated collapse of the HCGS cooling tower were evaluated in Reference 6.44. The closest estimated distance between the storage casks at the ISFSI and the collapsed part of the cooling tower is 242 feet. Hence, there is no impact of debris from a collapsed cooling tower on a storage cask at the ISFSI. The ground acceleration shock produced by the collapse of the cooling tower is approximately 0.08 g (horizontal) and 0.02 g (vertical). These acceleration values are much smaller than the design basis values for the cask system (0.25 g horizontal and 0.17 g vertical) and are, therefore, acceptable.

5.4.1.12 Deviations from the Dry Cask Storage System FSAR

Revisions 3 and 5 of the HI-STORM FSAR were the licensing basis for the first two loading campaigns at the Salem/Hope Creek ISFSI. Certain changes to Revision 3 of the HI-STORM FSAR were implemented by Holtec International on a generic basis after FSAR Revision 3 was issued. These changes were evaluated in accordance with the Holtec 10 CFR 72.48 program and are listed in prior revisions to this report, which governed previous cask loading campaigns. No CoC amendments as a result of these deviations were identified. Those changes were incorporated into subsequent revisions to the HI-STORM FSAR. No changes to FSAR Revision 5 were made by Holtec that affected the second loading campaign.

The specific details of the changes and the technical (e.g., Holtec Engineering Change Order (ECO)) and regulatory (72.48 screening or evaluation) documentation approving the changes are controlled as separate documents by Holtec. Appendix 2 to this report provides a table that lists the specific serial numbers for the cask components affected by ECO changes affecting the FSAR revision to which the hardware is certified. HI-STORM FSAR Revision 7 is being used for the 2010 and future loading campaigns. The changes to FSAR Revision 7 approved by Holtec under their 10 CFR 72.48 program (or as a result of a CoC amendment) are listed in Table 5.4.1.12-1 and the impacts of the changes on PSEG, if any, are described.

Table 5.4.1.12-1
Holtec-Implemented Changes to HI-STORM FSAR Revision 7*

Source Document	Description and Impact of Changes
ECO 5014-124, Rev. 1	This ECO originally removed the fabrication shop helium leakage test of the MPC from the FSAR. Revision 1 to the ECO added an exemption to testing for MPCs with a heat load less than 20 kW. However, as a result of NRC enforcement action on removal of the test (Reference 6.39), the test was restored under ECO 5014-174. The four MPCs delivered to Hope Creek for the 2010 loading campaign were not leak tested in the shop. These MPCs will be leak tested on-site prior to use.
ECO 5014-162	This ECO makes editorial changes to add a definition of "MPC Transfer" to conform with CoC Amendment 5 and to add MPC transfer from the transfer cask into the overpack as a potential function of the VCT. MPC transfer at Hope Creek is performed with the Reactor Building crane. There is no impact on PSEG site implementation documents or the evaluations summarized in this report.
ECO 5014-164	This ECO makes an editorial change to MPC Enclosure Vessel Licensing Drawing 3923 to add missing ECO numbers to the drawing revision log. There is no impact on PSEG site implementation documents or the evaluations summarized in this report.
ECO 5014-166, Rev. 1	This ECO modifies the HI-STORM FSAR to permit the use of as-rolled SA 516 Grade 70 carbon steel in fabricating the overpacks in addition to normalized steel of the same grade. This change affects material procurement and fabrication under the control of the CoC holder and does not affect PSEG site implementation documents or the evaluations summarized in this report.
ECO 5014-167	This ECO makes two sets of corrections to Chapter 4 of the HI-STORM FSAR, "Thermal Evaluation." The first set makes the text consistent with changes made in CoC Amendments 3 and 5. FSAR Chapter 4 was completely revised and sections re-numbered in Revision 7 of the FSAR as a result of the increased heat load permitted in CoC Amendment 5. The corrections in this ECO include clarifications of the description of the thermal evaluation of the various HI-TRAC transfer cask models, which has no impact on PSEG site implementation documents or the evaluations summarized in this report. The second set of changes in this ECO involves MPC unloading and affects a PSEG implementation procedure. FSAR Revision 7, Section 4.5.4 is completely replaced with the text from FSAR Revision 6, Section 4.5.1.1.6 (Section 4.5.1.1.6 was previously deleted in FSAR Revision 7). New Section 4.5.4 of FSAR Revision 7 pertains to direct re-flooding of the MPC for unloading operations as permitted by CoC Technical Specification LCO 3.1.3, which was revised in CoC Amendment 3 and retained in CoC Amendment 5. The FSAR now identifies an example of a limiting re-flooding rate of 3715 lb/hr to prevent overpressurization of the MPC during direct re-flooding, without pre-cooling, at design basis heat load. PSEG may use this limiting re-flooding rate or calculate a site-specific re-flooding rate based on a lower MPC heat load for the MPC being unloaded. This limiting re-flooding rate will be included in the MPC unloading procedure. The information removed from FSAR Section 4.5.4 by this ECO includes discussion of the methods for pre-cooling the MPC cavity gas prior to re-flooding. CoC Amendment 5 does not require pre-cooling of MPCs to be unloaded prior to re-flooding. FSAR Revisions 3 and 5 apply to previously loaded casks and they retain this pre-cooling requirement.
ECO 5014-168	This ECO makes changes to the HI-STORM 100U storage system design, which was under NRC review at the time the ECO was issued. PSEG does not use the HI-STORM 100U design. Thus, there is no impact on PSEG site implementation documents or the evaluations summarized in this report.

Table 5.4.1.12-1
Holtec-Implemented Changes to HI-STORM FSAR Revision 7*

Source Document	Description and Impact of Changes
ECO 5014-169	This ECO changes the word "crawler" to Vertical Cask Transporter in several locations in the FSAR. This is an editorial change and has no impact on PSEG site implementation or the evaluations summarized in this report.
ECO 5014-170, Rev. 1	This ECO allows the use of pre-cast lead sections or sheets in lieu of poured molten lead for shielding in the HI-TRAC 125D transfer cask. PSEG uses the HI-TRAC 100D model and the PSEG HI-TRAC was fabricated with poured lead before this ECO was issued. There is no impact on PSEG site implementation documents or the evaluations summarized in this report.
ECO 5014-171	This ECO adds the details of how to perform the thermal air flow test required by CoC Condition 9 to FSAR Chapter 8. This testing was performed by another HI-STORM user and was found acceptable for use by PSEG Nuclear. See discussion in Appendix 1, CoC Condition 9.
ECO 5014-172	This ECO deletes a fuel rod buckling analysis from FSAR Section 3.5, and replaces it with an alternate method of predicting fuel cladding behavior under g-loads that is described in NUREG 1864. This is an internal licensing matter between Holtec and the NRC that does not affect use of the cask in the field. There is no impact on PSEG site implementation documents or the evaluations summarized in this report.
ECO 5014-173	This ECO replaces a suggestion to use MPC water flushing with a requirement to do so, in the event the time-to-boil is approached during MPC preparation operations. Reference 6.31.6 ensures that MPC flushing will be performed if the time-to-boil is approached or exceeded by using the word "shall" for this operation..
ECO 5014-174	This ECO restores shop helium leakage testing of all MPCs to the FSAR and deletes the exemption canisters with heat loads ≤ 20 kW. This is related to the resumption of leakage testing described in the discussion of ECO 5014-124 R.1, above. All four Hope Creek MPCs delivered without having been shop-tested were tested at the site prior to use.
ECO 5014-175	This ECO revises FSAR Section 3.1.2.1.1.4 so that the description of the consideration of explosions and their pressure waves is consistent with the Technical Specifications in Appendix A the CoC. The hazards analysis performed for the PSEG ISFSI is consistent with the CoC for explosion/overpressure consideration. Therefore, there is no impact on PSEG site implementation documents or the evaluations summarized in this report.
ECO 5014-176	This ECO changes descriptions of the impact limiter for the HI-STAR transportation cask. Therefore it has no implications for the HI-STORM storage system. There is no impact on PSEG site implementation documents or the evaluations summarized in this report.
ECO 5014-177	This ECO makes changes to the HI-STORM 100U storage system design in support of a CoC amendment request. PSEG does not use the HI-STORM 100U design. Thus, there is no impact on site implementation documents or the evaluations summarized in this report.
ECO 5014-179	This ECO revises the FSAR to add constraints to use of the Supplemental Cooling System (SCS) that were made necessary by the increase in allowed maximum canister heat load approved in CoC Amendment 5. If and when PSEG needs to use the SCS, these constraints will be implemented by procedure (References 6.56.6 and 6.56.7).
ECO 5014-180	This ECO clarifies information in FSAR Section 3.5 regarding fuel cladding under g-loads that was introduced by ECO 5014-172. This is an internal licensing matter between Holtec and the NRC that does not affect use of the cask in the field. There is no impact on PSEG site implementation documents or the evaluations summarized in this report.
ECO 5014-182	This ECO adds a new set of material yield strength requirements for the MPC lid and updated results for lid lifting structural analyses. There is no impact on site implementation documents or the evaluations summarized in this report.

Table 5.4.1.12-1
Holtec-Implemented Changes to HI-STORM FSAR Revision 7*

Source Document	Description and Impact of Changes
ECO 5014-183	This ECO makes changes to the HI-STAR transportation package impact limiter design in support of a CoC amendment request. PSEG does not use the HI-STAR package for spent fuel storage. Thus, there is no impact on site implementation documents or the evaluations summarized in this report.
ECO 5014-184	This ECO corrects the helium backfill range for MPC-32 to match the value in the CoC. This is an editorial change and has no impact on site implementation or the evaluations summarized in this report.
ECO 5014-185	This ECO makes changes to the HI-STAR transportation package impact limiter design in support of a CoC amendment request. PSEG does not use the HI-STAR package for spent fuel storage. Thus, there is no impact on site implementation documents or the evaluations summarized in this report.
ECO 5014-187	This ECO makes changes to the HI-STAR transportation package Safety Analysis Report in support of a CoC amendment request. PSEG does not use the HI-STAR package for spent fuel storage. Thus, there is no impact on site implementation documents or the evaluations summarized in this report.
ECO 5014-188	This ECO makes several changes to the HI-STORM FSAR, one of which pertains to vacuum drying, which is not applicable to Salem, where FHD is used for canister dehydration. Any impacts on vacuum drying procedures at Hope Creek that need to be made as a result of this ECO will be performed prior to the next use of the VDS system. Another change made to the FSAR in this ECO was the permission to use nitrogen for MPC blowdown. Neither Salem nor Hope Creek use nitrogen for blowdown and currently has no plans to do so. The remaining changes were editorial corrections and clarifications. There is no impact on site implementation documents or the evaluations summarized in this report.
ECO 5014-191	This ECO adds clarifying information for coating materials used on the HI-TRAC and HI-STORM. There is no impact on site implementation documents or the evaluations summarized in this report.
ECO 5014-192	This ECO a) permits MPC blowdown with nitrogen as an alternative to helium and b) adds a requirement for tracking thermal cycles during vacuum drying. MPC blowdown is performed using helium at both SGS and HCGS. A change to nitrogen for blowdown would require a procedure revision and associated 72.48 review. Vacuum drying is not used at SGS. Thus there is no impact from change 'b' at SGS. The HCGS vacuum drying procedure will be reviewed and revised as required if HCGS does not switch to forced helium dehydration drying.
ECO 5014-193	This ECO modifies the text of FSAR Section 3.5 to address an NRC violation issued to Holtec. The change to the FSAR pertains to licensing bases analyses and does not affect use of the cask system in the field. Thus, there is no impact on site implementation documents or the evaluations summarized in this report.
ECO 5014-194	This ECO reinstates text in FSAR Section 9.1 that was inadvertently removed in a prior FSAR revision. The text clarifies the applicability requirements for pressure testing the HI-TRAC water jacket. PSEG's HI-TRAC was fabricated several years ago, before the text was inadvertently removed. Thus, there is no impact on site implementation documents or the evaluations summarized in this report.
ECO 5014-195	This ECO makes changes to the HI-STORM 100U storage system design in support of a CoC amendment request. PSEG does not use the HI-STORM 100U design. Thus, there is no impact on site implementation documents or the evaluations summarized in this report.

Table 5.4.1.12-1
Holtec-Implemented Changes to HI-STORM FSAR Revision 7*

Source Document	Description and Impact of Changes
ECO 5014-196	This ECO addresses Holtec International Bulletin 51, Revision 1. These changes are made to provide additional clarification in FSAR Chapters 1, 2, 4, and 12 so the users can correctly determine the approach for calculating the total MPC heat load for various operating scenarios. The heat load limits for long-term storage are not changed. In the Certificate of Compliance certain operations are dependent on total heat load in the MPC. This ECO adds text to the FSAR to provide guidance to the users on how to calculate the total heat load consistent with the assumptions used in the thermal analysis already performed for the system. The SGS fuel selection procedure has been revised to reflect these FSAR changes and the HCGS fuel selection procedure will be revised before casks are loaded at that station.
ECO 5014-197	This ECO modifies the FSAR instructions pertaining to MPC cooldown for unloading MPCs that were loaded under CoC Amendment 3 or later. Per Appendix 2 of this report, PSEG has loaded four MPCs containing HCGS fuel to CoC Amendment 2 and all others to Amendment 3 or later. The MPC unloading procedures for both HCGS and SGS have been reviewed and revised as required to reflect these FSAR changes.
ECO 5014-198	This ECO corrects the Technical Specification Bases in FSAR Appendix 12.A to match changes to the TS approved by the NRC in CoC Amendment 5. There are no changes to the TS requirements themselves. The PSEG DCS procedures are unaffected by these FSAR changes.
ECO 5014-199	This ECO revises the FSAR text in Chapters 8 and 9 to reflect the acceptability of performing NDE on the HI-TRAC trunnions in lieu of annual load testing. The PSEG DCS procedure was previously revised to perform NDE in lieu of load testing.

* Does not include one-time fabrication deviations addressed via the Holtec Supplier Manufacturer Deviation Report process. These are documented in the Holtec Component Completion Record for the affected cask component(s) or other document, such as a Field Deviation Report (FDR) for the ISFSI.

PSEG needed to implement two deviations from Revision 3 to the HI-STORM FSAR for the 2006-07 loading campaign that also carry over to all subsequent campaigns. They involved moving the loaded overpack outside of the Reactor Building without the lid installed and a repair of a ponding problem on one of the ISFSI pads. The location of overpack lid installation is not specifically addressed in the HI-STORM FSAR. Therefore, this evolution was addressed as a deviation under the PSEG 10 CFR 72.48 program for HCGS cask loading operations and found to be acceptable. The 72.48 review for overpack lid installation outdoors at HCGS is also applicable to SGS because the overpack design is the same, potential dose rates from the MPC with the overpack lid off are similar, and the lid is installed in the same manner and under the same procedural requirements at Salem.

The repair of the ISFSI pad ponding problem is discussed in more detail in Appendix 1, Table 3, Section 3.4.6 of this report. Both of these deviations also apply to HI-STORM FSAR Revision 5 and 7. One additional deviation from HI-STORM FSAR Revision 5 was required that permits a periodic inspection of the HI-TRAC lifting trunnions in lieu of a load test. This deviation is consistent with ANSI N14.6, which governs the trunnion design and also applies to HI-STORM FSAR Revision 7.

One additional deviation from FSAR Revision 7 is also being authorized by PSEG for SGS DCS operations only as documented in this report. It was recognized during the SGS DCS project that the Holtec generic passive cooling thermal analysis used for licensing short term operations with the HI-TRAC transfer cask, including drying with the Forced Helium Dehydration (FHD) System, was performed with a model that assumed the transfer cask was located on an open-air deck (i.e., a refuel floor) as described in HI-STORM FSAR Section 4.5.1.1.2. At SGS, draining, drying, and helium backfilling the MPC will be performed in the decontamination pit. This configuration was found to not be addressed by the generic HI-TRAC passive cooling thermal analysis performed by Holtec for licensing the FHD System. The Supplemental Cooling System (SCS) circulates cooling water through the MPC-to-HI-TRAC annulus to cool the MPC and its contents. The SCS provides much better heat removal than passive cooling simply because air in the annulus, which is assumed in the FSAR analysis, is replaced with cooling water.

Use of the SCS is required by the cask FSAR and TS (LCO 3.1.4) under two conditions: First, SCS must be used for MPCs having a heat load greater than 28.74 kW because the passive cooling thermal analysis was performed assuming a decay heat load of 28.74 kW and only MPCs with heat loads up to that value are bounded by the analysis. Second, the computed fuel cladding temperature resulting from the passive cooling analysis listed in cask FSAR Table 4.5.4 is 872^{o}F, which is less than the temperature limit for moderate burnup fuel (1058^{o}F), but exceeds the fuel cladding temperature limit of 752^{o}F for high burnup fuel. Thus, SCS must be used for any MPCs containing one or more high burnup fuel assemblies to ensure the temperature limit is not exceeded, regardless of MPC heat load.

To ensure the SGS condition is always bounded by the licensing basis, PSEG has included in their procedures (References 6.56.3 and 6.56.8) a requirement to use the SCS for all MPCs, regardless of whether they contain high burnup fuel and irrespective of heat load (see precaution and limitations section of each procedure). With these procedural requirements in place, the passive cooling thermal analysis in the cask FSAR does not apply to the PSEG condition with the MPC/HI-TRAC in the decontamination pit.

Table 5.4.1.12-2
PSEG-Implemented Deviations from the HI-STORM FSAR

Description of Deviation	FSAR Revision	Source Document
Installation of HI-STORM overpack lid outdoors	3, 5, 7	DCP 80088459 (HCGS) DCP 80091593 (SGS) Procedures HC/SC.MD-FR.DCS-0003 and HC/SC.MD-FR.DCS -0008
Ponding Repair for ISFSI Pad No. 1	3, 5, 7	SMDR 1410, R2
Inspection of HI-TRAC lifting trunnions in lieu of load testing	5, 7	Procedure NC.MD-PM.DCS-0013
Short term operations with HI-TRAC in the SGS decontamination pit	7	Procedures SC.MD-FR.DCS-0006 and SC.MD-FR.DCS-0008

5.4.1.13 CoC Holder Approval of Cask Operating Procedures

Holtec International has reviewed and approved the site dry cask storage operating procedures as required by HI-STORM FSAR Section 8.0, as documented in Reference 6.52. Holtec International also prepared the first drafts of the new Salem cask operating procedures to adopt CoC Amendment 5 and FSAR Revision 7. The Salem procedures are being created from the Hope Creek procedures previously reviewed and approved by Holtec. Furthermore, Holtec is contracted for loading services for the first Salem cask loading campaign and thus will have input to the procedure development process. Procedures are owned, understood, maintained and revised by PSEG Nuclear after the first loading campaigns at each station.

5.4.1.14 ISFSI Pad Elevation

Section 4.4.4.3 of HI-STORM FSAR Revision 7 requires users to confirm the elevation of the ISFSI pad to determine whether a site-specific thermal analysis is required. The HI-STORM FSAR requires a unique thermal analysis for ISFSI pads situated at elevation 1500 ft or higher. The Salem/Hope Creek ISFSI is located near the eastern shore of the Delaware River on land that slopes very gradually up from the shoreline. The ISFSI pad is situated well below 1500 ft. elevation. Thus, a unique thermal analysis, based on elevation, is not required for the Salem/Hope Creek ISFSI.

5.4.1.15 HI-TRAC Pedestal in SGS Transfer Pools

Because of differences in the geometry of the SGS fuel transfer pools, where MPC fuel loading takes place, and the spent fuel pool, it was necessary to design, construct, and install a pedestal for use in the transfer pools in both SGS units upon which the HI-TRAC transfer cask rests during fuel loading. This pedestal allows the MPC and HI-TRAC to be at an appropriate elevation to permit movement of fuel assemblies from the spent fuel pool into the MPC within the travel limitations of the fuel handing bridge and hoist. The configuration with the loaded HI-TRAC resting upon the pedestal in the fuel transfer pool has been analyzed (Reference 6.72) to ensure the transfer cask remains stable and will not topple over under static and dynamic loading conditions.

5.4.2 Conclusion

PSEG Nuclear complies with 10 CFR 72.212(b)(6)

5.5 10 CFR 72.212(b)(8) — Review of Part 50 Facility Impact (10 CFR 50.59)

10 CFR 72.212(b)(8) states the following:

"Prior to use of the general license, determine whether activities related to storage of spent fuel under this general license involve a change in the facility Technical Specifications or require a license amendment for the facility pursuant to §50.59(c)(2) of this chapter. Results of this determination must be documented in the evaluation made in paragraph (b)(5) of this section."

5.5.1 Evaluation

Hope Creek Evaluation

Several Hope Creek plant modifications required for ISFSI implementation have been performed pursuant to 10 CFR 50.59 using the PSEG configuration change process. In addition, an over-arching Design Change Package (DCP) documenting the acceptance of certain Holtec-generated qualification analyses and otherwise authorizing the conduct of dry cask loading activities in the Hope Creek Reactor Building has been developed. DCP 80088459, "Dry Cask Storage Operations" (Reference 6.34) summarizes cask loading activities and their impact on plant operations, including analyses required to ensure the building structures remain qualified for the expected loads. It also identifies the other DCPs for physical modifications required for ISFSI implementation such as ISFSI pad installation, heavy haul path upgrades, and security modifications. DCP 80088459 also addresses changes made to the Hope Creek UFSAR as a result of ISFSI operations and evaluates these changes under 10 CFR 50.59 (50.59 No. HC 06-006).

A review of the Hope Creek operating license (OL) was performed that indicated an administrative change to OL condition 2.C.(6) was required for ISFSI operations to proceed. That review revealed that Subpart 'a' of OL Condition 2.C.(6) prohibited more than three fuel assemblies to be out of an approved shipping container, the spent fuel racks, or the reactor at any one time. Because the dry storage system being used contains up to 68 fuel assemblies, this OL Condition could not be met during cask loading operations. License Change Request (LCR) H-06-01 (Reference 6.40) was submitted to the NRC on February 23, 2006 to request approval of a change to Subpart 'a' of the license condition to include NRC-approved dry spent fuel storage systems in the list of permissible locations for more than three fuel assemblies. In response to this LCR, Hope Creek operating license amendment 169 was granted by the NRC on August 28, 2006, lifting the three-assembly restriction.

Salem Evaluation

Several Salem plant modifications required for ISFSI implementation have been performed pursuant to 10 CFR 50.59 using the PSEG Design Change Package (DCP) process. In addition, an over-arching DCP documenting the acceptance of certain Holtec-generated qualification analyses and otherwise authorizing the conduct of dry cask loading activities in the Salem Fuel Handling Building has been developed. DCP 80091593, "Salem Dry Cask Storage Operations" was created to authorize implementation of DCS operations at Salem Units 1 and 2 (Reference 6.59). DCP 80091593 also addresses changes made to the Salem UFSAR and Technical Requirements Manual (TRM) as a result of ISFSI operations and evaluates these changes under 10 CFR 50.59 (50.59 No. S09-284).

A review of the Salem Units 1 and 2 operating licenses (OLs) and Technical Specifications (TS) was performed. That review revealed that a change to TS 3/4.9.7 was required for both units for cask loading operations to proceed. TS 3/4.9.7 limited the weight of a load being moved over irradiated fuel to 2,200 lbs. The MPC lid, which necessarily must be moved over the irradiated fuel in the canister during its installation, weighs about 10,000 lbs. License Amendment Request (LAR) S 08-06 was submitted to the NRC on April 9, 2009 to request approval of the relocation of TS 3/4.9.7 to the Salem TRM, In response to this LAR, Salem operating license amendments 277 (Unit 1) and 293 (Unit 2) were granted by the NRC on February 17, 2010, moving TS 3/4.9.7 to the TRM. Changes to the TRM are controlled under 10 CFR 50.59. The 10 CFR 50.59 evaluation associated with DCP 8001593 (S09-284) authorized this weight restriction to be appropriately modified to allow canister lid installation.

As a result of an enforcement issue at the site of another Holtec HI-STORM user, PSEG has designed, fabricated and installed seismic lateral restraints for the stack-up configuration of the HI-TRAC/HI-STORM while inside the Salem Unit 1 and Unit 2 truck bays during loading campaign. These modifications were performed under design change packages authorized by 10 CFR 50.59 (Reference 6.76). The seismic restraints ensure the transfer cask and overpack, when in the stack-up configuration, will not topple over in an earthquake.

5.5.2 Conclusion

Activities related to storage of spent fuel under the 10 CFR 72 general license were evaluated pursuant to 10 CFR 50.59 under a variety of design change packages and none of these activities resulted in the need to request NRC approval. However, a review of the Hope Creek operating license determined that an administrative change to the operating license was required to permit ISFSI operations to proceed. That amendment was requested and approved (Hope Creek operating license amendment 169). In addition, a review of the Salem OLs and TS revealed that TS changes were required for both units for cask loading operations to be conducted at those plants. Those amendments were requested and approved by the NRC (Salem operating license amendments 277 for Unit 1 and 293 for Unit 2). No other activities or modifications related to ISFSI implementation required prior NRC approval as documented in the associated 10 CFR 50.59 evaluations. Therefore, PSEG Nuclear complies with the requirements of 10 CFR 72.212(b)(8).

5.6 10 CFR 72.212(b)(9) — Physical Security

10 CFR 72.212(b)(9) states the following:

"Protect the spent fuel against the design basis threat of radiological sabotage in accordance with the same provisions and requirements as are set forth in the licensee's physical security plan pursuant to §73.55 of this chapter with the following additional conditions and exceptions:

(i) The physical security organization and program for the facility must be modified as necessary to assure that activities conducted under this general license do not decrease the effectiveness of the protection of vital equipment in accordance with § 73.55 of this chapter;

(ii) Storage of spent fuel must be within a protected area, in accordance with § 73.55(e) of this chapter, but need not be within a separate vital area. Existing protected areas may be expanded or new protected areas added for the purpose of storage of spent fuel in accordance with this general license;

(iii) For purposes of this general license, searches required by § 73.55(h) of this chapter before admission to a new protected area may be performed by physical pat-down searches of persons in lieu of firearms and explosives detection equipment;

(iv) The observational capability required by § 73.55(i)(3) of this chapter as applied to a new protected area may be provided by a guard or watchman on patrol in lieu of closed circuit television;

(v) For the purpose of this general license, the licensee is exempt from requirements to interdict and neutralize threats in §73.55 of this chapter; and

(vi) Each general licensee that receives and possesses power reactor spent fuel and other radioactive materials associated with spent fuel storage shall protect Safeguards Information against unauthorized disclosure in accordance with the requirements of §73.21 and the requirements of §73.22 or §73.23 of this chapter, as applicable."

5.6.1 Evaluation

The Salem/Hope Creek physical security plan and procedures were reviewed and modified, as necessary, to reflect spent fuel cask loading and transport operations on-site, as well as storage operations at the ISFSI. The ISFSI is located inside the site protected area. Procedural and design modifications have also been undertaken to implement ISFSI-related security interim compensatory measures. Those measures were described to the NRC and NRC provided their approval via letter in 2005 (Reference 6.41). The details of the physical security plan and procedures are necessarily security safeguards information and cannot be discussed in this report. The site security fence and intrusion detection system have also been modified to encompass the ISFSI inside the protected area.

5.6.2 Conclusion

Based on the changes to the Salem/Hope Creek physical security plan and procedures, as well as modifications to the protected area fence, PSEG complies with 10 CFR 72.212(b)(9) for protection of the spent fuel against the design basis threat of radiological sabotage in accordance with the same provisions and requirements as are set forth in the Salem/Hope Creek physical security plan pursuant to §73.55.

5.7 10 CFR 72.212(b)(10) — Programs

10 CFR 72.212(b)(10) states the following:

"Review the reactor emergency plan, quality assurance program, training program, and radiation protection program to determine if their effectiveness is decreased and, if so, prepare the necessary changes and seek and obtain the necessary approvals."

5.7.1 Evaluations

Each of the above-mentioned programs, the SGS Fire Protection Plan, and the HCGS Fire Protection Plan (HC Operating License Condition 2.C.(7)) has been evaluated for impact by the implementation of ISFSI operations. The evaluation of each program plan is summarized below with appropriate cross-references to the plan documents and implementing procedures.

5.7.1.1 Emergency Plan

The SGS and HCGS Event Classification Guides (ECGs) and Emergency Action Levels (EALs) were reviewed for impact as a result of implementing dry cask storage at the two generating stations. The ECGs were revised appropriately to address ISFSI operations, through the creation of new EAL 6.4.1.c and associated bases for both stations.

Any significant increase in the dose rate from a cask would indicate a loss of shielding effectiveness rather than a change to the source term inside the cask. This is because the amount of radioactive material in the cask is fixed at the time of loading and cannot increase (although it could re-locate due to gravity effects after a cask drop or other dynamic event). In fact, due to radioactive decay, the source term in the cask will decrease over time and dose rates would be expected to decrease, given the same amount of shielding with the source in approximately the same location inside the MPC.

The Reportability Action Levels (RALs) in the ECGs at both stations were also reviewed and revised to take into consideration new reportability requirements in the HI-STORM CoC and the Part 72 regulations. Based on the changes to the PSEG ECGs and the new ISFSI RALs and EALs, the requirements of 10 CFR 72.212(b)(10) pertaining to the reactor emergency plan are met.

5.7.1.2 Quality Assurance Program

Cask design, fabrication, assembly, and related activities are performed under Holtec's NRC-approved quality assurance program as described in Chapter 13 of the HI-STORM FSAR.

As allowed by 10 CFR 72.140(d), the existing PSEG Nuclear NRC-approved 10 CFR 50, Appendix B Quality Assurance Program for SGS and HCGS is being applied to ISFSI activities (Reference 6.10). The description of PSEG's Quality Assurance program in the PSEG Quality Assurance Topical Report (Reference 6.33.2) has been revised to include activities related to ISFSI operations as described in Appendices A and E. It was determined that these changes did not reduce the effectiveness of the QA program and could be implemented without prior NRC approval.

The graded approach to quality for ISFSI and dry cask storage structures, systems, and components and activities is implemented consistent with the guidance in NUREG/CR-6407 (Reference 6.23) and the HI-STORM FSAR for classifying structures, systems and components associated with cask loading, on-site transport, and ISFSI operations. Procedural controls are in place to appropriately classify ISFSI-related structures, systems, and components (SSCs) according to the NUREG/CR-6407 guidance (Reference 6.54.2). These classifications are then used to govern the applicable quality requirements for activities involving these SSCs.

Based on the Holtec QA Program and the PSEG Nuclear Appendix B QA Program, and Reference 6.54.2, the requirements of 10 CFR 72.212(b)(10) pertaining to quality assurance are met.

5.7.1.3 Training Program

The PSEG Nuclear Training program is summarized in Reference 6.33.4. Important-to-Safety operations for the cask system are conducted by trained and qualified personnel under the direction of trained supervisors. Training is performed under the existing station training program utilizing the Systematic Approach to Training, as described in Reference 6.33.4. Specific Fuel Handler training courses have been developed to cover the cask system. Welding system operations (MPC lid installation, NDE, and weld removal) and helium leak testing for the cask system are performed by qualified outside specialty vendor(s) under procedures approved by PSEG.

Support activities for the cask systems are performed by Design Engineering, System Engineering, Reactor Engineering, Fuels, Maintenance, Radiation Protection and Security personnel. The scope of training applicable to these personnel is covered under lesson plans created by PSEG training personnel with support from dry cask storage subject matter experts.

The physical and health requirements applicable to Salem/Hope Creek ISFSI operations are included in the SGS and HCGS DCS procedures, as appropriate (References 6.31 and 6.56). Supervision by a first line supervisor is adequate to ensure activities are performed within the capability of the crew.

Prior to first use at HCGS and again at SGS, PSEG performed, or will perform dry run training exercises that meet the requirements of Condition 10 of the HI-STORM 100 CoC, with certain exceptions. The bases for those exceptions are discussed in Table 1 of Appendix 1 to this report. Cask loading activities performed at both HCGS and SGS were only dry run at SGS to the extent the activity in whole, or in part, is performed differently at SGS, or different equipment was used For example, the Forced Helium Dehydration and Supplemental Cooling Systems (required by the HI-STORM CoC for certain MPC contents per LCOs 3.1.1 and 3.1.4) are being used for the first time at SGS.

Based on the modifications made to the PSEG training program and associated implementation of classroom training and dry run exercises, the requirements of 10 CFR 72.212(b)(10) pertaining to training are met.

5.7.1.4 Radiation Protection Program

Radiation protection personnel have been trained and procedures revised to support dry cask storage loading operations in the plant and ISFSI operation. The radiation protection program (Reference 6.33.1) and relevant implementing procedures have been revised or new procedures created to support cask loading, on-site transportation, and storage operations in an ALARA manner.

Based on the modifications made to the PSEG radiation protection program and associated implementation of new and revised procedures that support cask loading, transport, and storage operations, the requirements of 10 CFR 72.212(b)(10) pertaining to radiation protection are met.

5.7.1.5 Fire Protection Plan

HCGS Evaluation

The fire hazards analysis for HCGS has been revised to address cask loading activities inside the Reactor Building, at the ISFSI, and during on-site cask transportation activities between the Reactor Building and the ISFSI. Based on the findings of the fire hazards analysis, appropriate changes have been made to the fire protection implementing procedures to assure adequate protection of the plant and dry storage casks during all phases of cask loading, transport, and storage operations (Reference 6.33.5). This includes control of transient combustible material at both the ISFSI and along the heavy haul path, as well as during fuel transfer operations in the Reactor Building. In addition, fire suppression equipment and personnel trained in its use accompany the cask while in transit from the Reactor Building to the ISFSI. The specific fire and explosion hazards associated with dry cask storage are discussed in more detail in Section 5.4.1.1 of this report. The fire protection programmatic standard (Reference 6.33.5) was reviewed and no changes were required.

SGS Evaluation

The fire hazards analysis for SGS has been revised to address cask loading activities inside the Fuel Handling Building and during on-site cask transportation activities between the Fuel Handling Building and the Hope Creek portion of heavy haul path. Based on the findings of the fire hazards analysis, appropriate changes have been made to the SGS fire protection implementing procedures to assure adequate protection of the plant and dry storage casks during all phases of cask loading and transport operations (Reference 6.33.5). This includes control of transient combustible material along the SGS portion of the heavy haul path as well as during fuel transfer operations in the Fuel Handling Building. In addition, fire suppression equipment and personnel trained in its use accompany the cask while in transit from the Fuel Handling Building to the ISFSI. The specific fire and explosion hazards associated with dry cask storage are discussed in more detail in Section 5.4.1.1 of this report.

Based on the modifications made to the PSEG fire protection program and associated implementation procedures that support cask loading, transport, and storage operations, the requirements of 10 CFR 72.212(b)(10) pertaining to fire protection are met.

5.7.2 Conclusion

All relevant program plans have been reviewed and evaluated for ISFSI impact and modified as necessary to include changes to reflect ISFSI operations. The details of those changes are maintained in the program plan documents and associated change packages (i.e., evaluations pursuant to 10 CFR 50.54)

6.0 REFERENCES

The reference documents listed below provide the bases for the factual statements in the body of this report. Revision levels of PSEG-controlled procedures are not provided because the PSEG 72.48 screening process requires ISFSI-related procedure changes to be checked against the content of the 212 Report to determine if a revision of the 212 Report is required. Thus, it may be assumed that the latest revisions of PSEG procedures listed in this section are applicable.

6.1 HI-STORM 100 Cask System 10 CFR 72 Certificate of Compliance No. 1014, Amendments 2, 3, and 5, PSEG VTD No. 400004 (001), (002), and (003), respectively.

6.2 HI-STORM 100 Cask System Final Safety Analysis Report, Holtec Report No. HI-2002444, Revisions 3, 5, and 7, Docket No. 72-1014, PSEG VTD No. 400006 (001), (002), and (003), respectively.

6.3 NRC Regulatory Guide 1.76, “Design Basis Tornado for Nuclear Power Plants,” April 1974.

6.4 NRC Information Notice 2003-16, “Icing Conditions Between Bottom of Dry Storage System and Storage Pad,” October 2003.

6.5 NRC Safety Evaluation Report for the HI-STORM 100 Cask System (through Amendment No. 5 to CoC No. 1014), PSEG VTD No. 400004 (003).

6.6 NUREG-1567, “Standard Review Plan for Spent Fuel Dry Storage Facilities,” March 2000.

6.7 NUREG-1536, “Standard Review Plan for Dry Cask Storage Systems,” January 1997.

6.8 Holtec International Letter to PSEG, “Hope Creek VCT Compliance with the HI-STORM CoC,” Holtec Document ID 1332046, dated June 19, 2006, PSEG VTD No. 400002 (001).

6.9 PSEG Nuclear Contract with Holtec International SCM-09-NUC-391, Attachment 5, Exhibit 3, Item 10.

6.10 PSEG Nuclear Letter to NRC, “Notification of Intent to Apply the 10 CFR 50 Appendix B Quality Assurance Program to Independent Spent Fuel Storage Installation Activities,” dated May 12, 2003, Dockets 50-272, 50-311, and 50-354.

6.11 PSEG Engineering Evaluation A-5-DCS-FEE-1766, “Hope Creek Generation Station Independent Spent Fuel Storage Installation Fire Hazard Analysis,” PSEG Rev. 0.

6.12 Holtec Report No. HI-2043195, "HI-STORM 100 System Overpack Air Temperature Rise at 17.1 kW, Rev.0;" Holtec letter No. 9042868 to Energy Northwest, "HI-STORM Thermal Validation Test Results," dated July 12, 2004; and Energy Northwest Letter No. G02-04-134 to the NRC, "Columbia Generating Station Validation of HI-STORM 100 System Heat Transfer Characteristics," dated July 28, 2004, Docket 72-35.

6.13 PSEG Calculation No. A-5-DCS-MDC-1958, "Source Term Analysis for the Salem & Hope Creek ISFSI," Rev. 0.

6.14 PSEG Calculation No. A-5-DCS-MDC-1957, "Direct Dose Rates in the Vicinity of the Salem & Hope Creek ISFSI," Rev. 0.

6.15 PSEG Calculation No. A-5-DCS-CDC-1986, "ISFSI Fire Radiant Heat and Explosion Overpressure Analysis," Rev. 0.

6.16 Docket No. 50-354, PSEG Nuclear LLC Hope Creek Generating Station Facility Operating License No. NPF-57, through Amendment 169.

6.17 Sargent & Lundy Letter to PSEG Nuclear, "ISFSI Design and Support, External Flood Events," dated May 30, 2003, PSEG VTD No. 400066 (001).

6.18 PSEG Calculation No. A-5-DCS-CDC-1960, "ISFSI Pad Design," Rev. 0.

6.19 PSEG Calculation No. A-5-DCS-CDC-1978, "Soil Parameters for the ISFSI Pad Area," Rev. 1.

6.20 PSEG Calculation No. A-5-DCS-CDC-1964, "Soil Structure Interaction and Time History Calculation," Rev. 0.

6.21 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 3.5.1.4, "Missiles Generated By Natural Phenomena," Rev. 2, July 1981.

6.22 NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," July 1980.

6.23 NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," February 1996.

6.24 PSEG Nuclear Fuels Memorandum NFS 12-041, "HI-STORM 100S Cask System Thermal Validation Testing Using Air Mass Flow Rate," dated April 20, 2012.

6.25 ANSI N14.6-1993, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4500 kg) or more for Nuclear Materials."

6.26 URS Report, "Geotechnical Investigation for Salem/Hope Creek ISFSI," Report Submitted by Steven D. Coppola of URS Corporation to Ms. Shelly Kugler of PSEG, August 13, 2003, PSEG VTD No. 325972 (001).

6.27 ASHRAE Applications Handbook, 1999 Edition, Chapter 31.

6.28 PSEG Calculation No. A-5-DCS-CDC-1963, "Underground/Above Ground Utilities Evaluation," Rev. 0.

6.29 Holtec International Project Procedure HPP-5014-22, "Cask Storage Pad/HI-STORM Interface Friction Coefficient Determination," Rev. 1, PSEG VTD No. 400062 (001).

6.30 Holtec International Letter No. 1332044 to PSEG Nuclear, "ISFSI Pad Broom Finish," dated May 24, 2006, PSEG VTD No. 400064 (001).

6.31 **HCGS Procedures**

6.31.1 HC.MD-FR.DCS-0001, "HI-STORM System Receipt Inspection."

6.31.2 HC.MD-FR.DCS-0002, "Offloading and Receiving Dry Storage Components."

6.31.3 HC.MD-FR.DCS-0003, "Transport Loaded and Unloaded HI-STORM and HI-TRAC."

6.31.4 HC.MD-FR.DCS-0004, "MPC Preparation for Loading."

6.31.5 HC.MD-FR.DCS-0005, "Handling and Loading MPC."

6.31.6 HC.MD-FR.DCS-0006, "Sealing, Drying, and Backfilling of a Loaded MPC."

6.31.7 HC.MD-FR.DCS-0007, "Stack-up and Transfer of Loaded MPC."

6.31.8 HC.MD-FR.DCS-0008, "Transporting and Transferring a Loaded MPC for Unloading."

6.31.9 HC.MD-FR.DCS-0009, "Unloading a Loaded MPC."

6.31.10 Not used.

6.31.11 Not used.

6.31.12 Not used.

6.31.13 HC.OP-AB.MISC-0004, "ISFSI-Spent Fuel Storage Cask (SFSC) Heat Removal System."

6.31.14 HC.RE-FR.DCS-0001, "Dry Cask Storage Fuel Characterization."

6.31.15 HC.RE-FR.DCS-0002, "Dry Cask Storage Fuel Selection for Cask Loading."

6.31.16 HC.RE-FR.ZZ-0001, "Hope Creek Special Nuclear Material and Core Component Movement."

6.31.17 HC.RE-FR.ZZ-0008, "Verification of Fuel Location."

6.31.18 HC.OP-AR.DCS-0001, "Dry Cask Storage System Thermal Monitoring System Alarm Response Procedure."

6.31.19 Not used.

6.31.20 RP-HC-303, "HI-TRAC Radiation Survey."

6.31.21 RP-HC-304, "HI-STORM Radiation Survey."

6.31.22 RP-HC-305, "Independent Spent Fuel Storage Installation Radiation Survey."

6.31.23 Not used.

6.31.24 HC.OP-AB.MISC-0001, "Acts of Nature."

6.31.25 HC.OP-DL.ZZ-0026, "Surveillance Log."

6.31.26 Not used.

6.31.27 Not used.

6.32 Drawings

6.32.1 Not used.

6.32.2 Not used.

6.32.3 Holtec Drawing 3928, "MPC-68/68F/68FF Basket Assembly [Licensing Drawing]," Section 1.5 of PSEG VTD No. 400006 (001), (002), and (003) (applicable revision per component CCR).

6.32.4 Holtec Drawing 3923, "MPC Enclosure Vessel [Licensing Drawing]," Section 1.5 of PSEG VTD No. 400006 (001), (002), and (003) (applicable revision per component CCR).

6.32.5 Holtec Drawing 4128, "HI-TRAC 100D Assembly [Licensing Drawing]," Rev. 5, (Section 1.5 of PSEG VTD No. 400006 (002).

6.32.6 Holtec Drawing 4116, "HI-STORM 100S, Version B [Licensing Drawing]," Section 1.5 of PSEG VTD No. 400006 (001), (002), and (003) (applicable revision per component CCR).

6.32.7 Lift Systems Drawing CT201064, "210 Ton Transporter (Hope Creek)," Rev. A, PSEG VTD No. 400036 (005).

6.32.8 PSEG Drawing No. 700002, "Cask Storage Pad Sections and Details," Rev. 0.

6.32.9 Holtec Drawing No. 4532, "Soil Mixing As-Built," Rev. 2, PSEG VTD No. 400001 (001).

6.33 Plan Documents

6.33.1 RP-AA-300, "Radiological Survey Program."

6.33.2 Salem and Hope Creek Generating Stations Quality Assurance Topical Report (QATR), NO-AA-10.

6.33.3 Hope Creek Event Classification Guide (ECG), Emergency Action Level 6.4.1.c.

6.33.4 Training Procedures and T&RMs TQ-AA-210,“TSD Process Activities,” TQ-AA-103, “Instructor Training and Development Program,” TQ-SH-210-9001 “Training System Development,” and TQ-SH-103-9001, “Salem/Hope Creek Instructor Training and Development Program.”

6.33.5 NC.DE-PS.ZZ-0001, “Programmatic Standard for Fire Protection.”

6.34 PSEG Design Change Package 80088459, “Dry Cask Storage Operations,” Rev. 0.

6.35 PSEG Calculation A-5-DCS-CDC-1977, “Design for the ISFSI Heavy Haul Road,” Rev. 0.

6.36 Holtec Report No. HI-2043226, “Non-mechanistic Tipover of HI-STORM 100B at Hope Creek ISFSI Pad,” Rev. 6, PSEG VTD No. 400012 (001).

6.37 PSEG Design Change Package 80057739, “ISFSI Pad,” Rev. 2.

6.38 NFPA-30, “Flammable and Combustible Liquids Code,” National Fire Protection Association, 2000.

6.39 Letter from D. Pstrak, NRC, to T. Morin, Holtec International, EA 09-0190, “Exercise of Enforcement Discretion – Holtec International,” August 5, 2009.

6.40 PSEG Nuclear letter LR-N06-0025 to the NRC, License Change Request H06-01, “Request for Change to Operating License Condition 2.C.(6) for Hope Creek Generating Station,” dated February 23, 2006.

6.41 Letter from P. Harris, NRC, to B. Levis, PSEG Nuclear, “Response to Orders Requiring Implementation of Interim Security Compensatory and Access Authorization Measures for Hope Creek and Salem Generating Stations Independent Spent Fuel Storage Installation,” dated September 29 2005.

6.42 PSEG Calculation HCP.6-0207, “Verification of Hope Creek Cycles 1-12 Bundle Characteristics Against Holtec CoC Amendment 2,” August 4, 2005.

6.43 PSEG Purchase Specification A-5-DCS-NDS-0457, “Dry Cask Storage Project Prime Mover,” Rev. 1.

6.44 PSEG Calculation A-5-DCS-CDC-1965, “Adjacent Facilities Evaluation,” Rev. 1.

6.45 PSEG Calculation A-5-DCS-SDC-1961, “PMH Forces on Storage Cask,” Rev. 0.

6.46 Holtec Report No. HI-2043319, Rev. 9, “Seismic Analyses of the Crawler, HI-STORM, and LPT on the Egress Pad,” PSEG VTD No. 400016 (001).

6.47 Holtec Report No. HI-2063502, Rev. 4, “Miscellaneous Analyses Supporting Cask Loading at Hope Creek,” PSEG VTD No. 400051 (001).

6.48 Holtec Report No. HI-2043197, Rev. 0, "Evaluation of Kinematic Stability of HI-STORM Version B Under the Postulated Probable Maximum Hurricane," PSEG VTD No. 400035 (001).

6.49 Houghton International Product Data Sheet for Cosmolubric® Hydraulic Fluid (included in cask transporter operating and maintenance manual), PSEG VTD No. 400019 (001).

6.50 Holtec Report No. HI-2043313, Rev. 2, "Design Basis Wind, Tornado, and Snow Load Evaluation for Hope Creek Generating Station," Revision 2, PSEG VTD 400021 (001).

6.51 PSEG Notification No. 20249856 and FCR No. 310 to Order 60035559.

6.52 Holtec letter to B. Gustems, PSEG dated July 26, 2006, "Holtec Review of Hope Creek Dry Cask Storage Procedures," PSEG Design Input Record No. H-1-DCS-NDI-0126.

6.53 USNRC Division of Spent Fuel Storage and Transportation Interim Staff Guidance 22, "Potential Rod Splitting due to Exposure to an Oxidizing Atmosphere during Short-Term Cask Loading Operations in LWR or Other Uranium Oxide-Based Fuel," Revision 0.

6.54 **Nuclear Common Procedures**

6.54.1 FP-AA-001, "Precautions Against Fire."

6.54.2 CC-AA-103-1001, "Implementation of Configuration Changes."

6.55 PSEG Notification No. 2042611 and Order 70100870, "MPC Shop Leakage Testing."

6.56 **SGS Procedures**

6.56.1 Not used.

6.56.2 Not used.

6.56.3 SC.MD-FR.DCS-0003, "Transport Loaded and Unloaded HI-STORM and HI-TRAC."

6.56.4 SC.MD-FR.DCS-0004, "MPC Preparation for Loading."

6.56.5 SC.MD-FR.DCS-0005, "Handling and Loading MPC."

6.56.6 SC.MD-FR.DCS-0006, "Sealing, Drying, and Backfilling of a Loaded MPC."

6.56.7 SC.MD-FR.DCS-0007, "Stack-up and Transfer of Loaded MPC."

6.56.8 SC.MD-FR.DCS-0008, "Transporting and Transferring a Loaded MPC for Unloading."

6.56.9 SC.MD-FR.DCS-0009, "Unloading a Loaded MPC."

6.56.10 Not used.

6.56.11 SC.RE-FR.DCS-0001, "Dry Cask Storage Fuel Characterization."

6.56.12 SC.RE-FR.DCS-0002, "Dry Cask Storage Fuel Selection for Cask Loading."

6.56.13 SC.RE-FR.DCS-0003, "Fuel Spacer Matrix."

6.56.14 SC.RE-FR.ZZ-0001, "Fuel Handling."

6.56.15 RP-SA-303, "HI-TRAC Radiation Survey."

6.56.16 RP-SA-304, "HI-STORM Radiation Survey."

6.56.17 SC.RE-FR.ZZ-0007, "Verification of Fuel Locations."

6.57 Sargent & Lundy Report No. 009670, "Salem Generating Station Independent Spent Fuel Storage Installation Fire Hazards Analysis," Revision 1, PSEG VTD 901554 (001).

6.58 PSEG Nuclear letter LR-N09-0034 to the NRC, License Amendment Request S08-06, "License Amendment Request to Relocate Communications, Manipulator Crane, and Crane Travel Requirements from Technical Specifications," April 9, 2009, and Salem Units 1 and 2 License Amendments 277 and 293.

6.59 PSEG Design Change Package 80091593, "Salem Dry Cask Storage Operations," latest revision.

6.60 Not used.

6.61 **Contractor Procedures**

6.61.1 Holtec Procedure HPP-1746-600, "Procedure for MPC Cooldown and Weld Removal for MPC Unloading at Hope Creek and Salem Plants," Revision 2.

6.61.2 PCI procedure PI-900995-01, "Closure Welding of Multi-Purpose Canisters at Hope Creek and Salem," Revision 2.

6.62 U.S. NRC Enforcement Guidance Memorandum 09-006, "Enforcement Discretion for Violations of 10 CFR 72, Subpart K, Regarding Implementation of Certificate of Compliance Amendments to Previously Loaded Spent Fuel Storage Casks," September 15, 2009.

6.63 Holtec Report No. HI-2094469, "Design Basis Wind, Tornado, and Snow Load Evaluation for Salem Generating Station," Revision 1, PSEG VTD 901918 (001).

6.64 Holtec Report No. HI-2073864, "Seismic Analysis of the Crawler, HI-STORM and ZPT on the Egress Pad," Rev. 8, PSEG VTD 901298 (001).

6.65 Holtec Operations and Maintenance Manual for the Supplemental Cooling System and Manifold, Ancillary 421, PSEG VTD 901300 (001).

6.66 Not Used.

6.67 Letter from Holtec International, Andrew Fecht, to PSEG Nuclear, Tom Wallender, "Leaving Mating Device Installed Atop a Loaded HI-STORM in a ZPT," dated July 30, 2010.

6.68 Holtec Component Completion Record 1027-421-8, "MPC Supplementary Cooling System," Revision 0.

6.69 Holtec Report No. HI-2073816, "Structural Analysis of Zero Profile Transporter (ZPT)," Rev. 5, PSEG VTD 901426 (001).

6.70 Email from A. Fecht, Holtec, to B. Gutherman, PSEG, "ZPT Hydraulic Fluid Volume and MSDS," dated July 15, 2010.

6.71 Holtec Purchase Specification PS-1129, "HI-STORM Zero Profile Transporter," Revision 5, PSEG VTD 901475 (001).

6.72 Holtec Report No. HI-2104690, "Seismic/Structural Analysis of HI-TRAC/Pedestal at Salem," Rev. 2, PSEG VTD 902010 (001).

6.73 Not used.

6.74 Holtec Report No. HI-2022966, "Forced Helium Dehydrator Sourcebook," Rev. 4, PSEG VTD 902019 (001).

6.75 Holtec 10 CFR 72.48 Evaluation No. 915, "Thermal Evaluation of Loaded HI-STORM in the ZPT," Revision 1.

6.76 PSEG Design Change Package 80103873, "Salem Unit 1 DCS-Stack-up Seismic Lateral Restraints," Rev. 0.

6.77 PSEG Design Change Package 80104315, "Salem Unit 2 DCS-Stack-up Seismic Lateral Restraints," Rev. 0.

APPENDIX 1

HI-STORM 100 CASK SYSTEM CERTIFICATE OF COMPLIANCE EVALUATION

INTRODUCTION

This appendix provides an evaluation of compliance with the HI-STORM 100 System Certificate of Compliance for HCGS and SGS spent fuel and site-specific conditions. This evaluation is presented in the following three compliance evaluation tables:

Table	Title
Table 1	CoC Conditions
Table 2	CoC Appendix A — Technical Specifications
Table 3	CoC Appendix B — Approved Contents and Design Features

The evaluation of compliance with the conditions set forth in the Certificate of Compliance presented in this appendix provides the basis for the conclusion reached in the compliance evaluation of 10 CFR 72.212(b)(5)(i) discussed in Section 5.1 of the main body of this report for HCGS and SGS spent nuclear fuel. In the 2006-07 loading campaign, four casks containing HCGS spent fuel were loaded in accordance with Amendment 2 of the HI-STORM CoC. In the 2008 loading campaign, eight casks containing HCGS spent fuel were loaded in accordance with Amendment 3 of the HI-STORM CoC. Differences between Amendments 2 and 3 of the CoC, if applicable to HCGS, are noted in the following table and the compliance statement is revised, as necessary, to recognize the two amendments as they apply to different casks in accordance with the table in Appendix 2 of this report.

In the 2010 loading campaigns at HCGS and SGS, Amendment 5 of the HI-STORM CoC is the governing CoC amendment. Amendment 5 will continue to be used for loading campaigns at both stations until such time as this 212 Report is revised to adopt a later amendment. Amendment 4 of the HI-STORM CoC pertained only to Indian Point Unit 1. The Amendment 4 changes were not retained in Amendment 5 and are therefore not discussed here. Differences between Amendments 3 and 5 of the CoC, if applicable to HCGS, are noted in the following table and the compliance statement is revised, as necessary, to recognize the two amendments as they apply to different casks in accordance with the table in Appendix 2 of this report. All SGS casks, fuel selection, and loading operations comply with CoC Amendment 5.

Table 1, CoC Conditions

Condition	**Evaluation**
1. CASK a. Model No.: HI-STORM 100 Cask System b. Description	This CoC condition describes the major HI-STORM 100 System components. PSEG Nuclear uses the HI-STORM 100 System components as described in Paragraphs 'a' and 'b' of this CoC condition. The specific components used to store HCGS and SGS spent fuel at the ISFSI are: 1. The HI-STORM 100S-218 Version B overpack. The "218" modifier designates that the 218-inch tall model of the Version B overpack is being used. See Section 1.2.1.2.1 of Reference 6.2. 2. The MPC-68 or MPC-68FF canister may be used to store HCGS spent fuel. PSEG chooses to use the MPC-68 to store HCGS spent fuel. The MPC-24, MPC-24E, MPC-24EF, MPC-32, or MPC-32F may be used to store SGS spent fuel. PSEG chooses to use the MPC-32 to store SGS spent fuel. 3. MPC loading, preparation, and transfer activities in the HCGS Reactor Building and SGS Fuel Handling Building are performed using the 100-ton HI-TRAC 100D transfer cask. Amendment 3 to the CoC made editorial clarifications to Section 1.b that have no effect on the compliance statements above. Amendment 5 to the CoC made editorial clarifications to Section 1.b that have no effect on the compliance statements above.
2. OPERATING PROCEDURES	Chapter 8 of the HI-STORM 100 System FSAR outlines the loading, unloading, and recovery procedures for the HI-STORM 100 Cask System. The procedures provided in the HI-STORM FSAR are prescriptive to the extent that they provide the basis and general guidance for plant personnel in preparing detailed, written, site-specific, loading, handling, storage and unloading procedures. Users are permitted to add, modify the sequence of, perform in parallel, or delete steps as necessary provided that the intent of the guidance given in Chapter 8 is met, and the requirements of the Technical Specifications in Appendix A to Certificate of Compliance No. 1014 are met (Reference 6.2, Section 8.0). PSEG Nuclear uses site-specific written operating procedures for implementation of cask loading, handling, movement, on-site transportation, surveillance, and maintenance of the HI-STORM 100 Cask System at the ISFSI. The site-specific operating procedures are consistent with the technical bases described in HI-STORM 100 System FSAR, Chapter 8 and the CoC. Changes in the operating procedures contained in the HI-STORM FSAR over time have been incorporated into the plant procedures as later FSAR revisions were adopted for use at the Salem/Hope Creek ISFSI.

Table 1, CoC Conditions

Condition	Evaluation
3. ACCEPTANCE TESTS AND MAINTENANCE PROGRAM	This CoC condition requires that written cask acceptance tests and maintenance program [implementation] shall be prepared consistent with the technical basis described in Chapter 9 of the FSAR. The acceptance tests and inspections required during component fabrication are carried out and documented in accordance with the certificate holder's quality assurance program. PSEG Nuclear uses site-specific procedures to implement the performance of maintenance, tests and inspections applicable to use of the storage system in accordance with the technical bases of HI-STORM 100 System FSAR, Chapter 9. Changes in the maintenance, tests, and inspections contained in the HI-STORM FSAR over time have been incorporated into the plant procedures as later FSAR revisions were adopted for use at the Salem/Hope Creek ISFSI. Acceptance tests are performed by the CoC holder and PSEG Nuclear under the applicable QA program and procedures. Normal maintenance of the HI-STORM 100 System is limited to periodic touch-up repairs of the cask coating due to minor nicks and scratches. Maintenance of the cask temperature monitoring system is performed on an as-needed basis.
4. QUALITY ASSURANCE	Activities important to safety are conducted under the appropriate Quality Assurance program having jurisdiction over the activity. Cask and important-to-safety ancillary component design, fabrication, inspection, and testing activities are conducted under the Holtec International 10 CFR 72, Subpart G Quality Assurance Program. Holtec's implementation of quality activities is monitored by PSEG Nuclear via controls imposed through the safety-related procurement for the cask system. On site activities are governed by the applicable portions of either the PSEG Nuclear 10 CFR 50, Appendix B Quality Assurance Program, as augmented to include Part 72 activities, or the QA program of the pool-to-pad services provider. See also Section 5.7.1.2 of the main body of this report.
5. HEAVY LOADS REQUIREMENTS	Changes to the HCGS and SGS Part 50 UFSARs have been made to address HI-STORM 100 Cask System loading operations performed inside the HCGS Reactor Building and SGS Fuel Handling Buildings. Each lift of a HI-STORM 100 System MPC, HI-TRAC transfer cask, HI-STORM overpack, or other heavy load associated with dry cask operations that is performed inside plant structures governed by 10 CFR 50, is made in accordance with approved PSEG Nuclear procedures that have been evaluated in accordance with the requirements of 10 CFR 50.59 and comply with the applicable site and station heavy load handling program. The 10 CFR 50.59 evaluation performed in accordance with 10 CFR 72.212(b)(8) for each plant addresses the heavy load handling aspects of ISFSI implementation. These activities are addressed in more detail in the 10 CFR 50.59 evaluation for the Design Change Package (DCP) documented under PSEG Order Nos. 80088459 and 80091593, "Dry Cask Storage Operations" (References 6.34 and 6.59).

Table 1, CoC Conditions

Condition	**Evaluation**
5. HEAVY LOADS REQUIREMENTS (cont'd)	Movement of a loaded HI-STORM overpack is performed in accordance with approved PSEG Nuclear procedures, and in compliance with HI-STORM 100 Cask System Certificate of Compliance, Appendix A, Section 5.5 (see Table 2 of this appendix). Lifting of a fuel-loaded HI-TRAC transfer cask and MPC is not performed outside of SGS and HCGS structures governed by 10 CFR 50. Therefore, HI-STORM 100 Cask System Certificate of Compliance 1014, Appendix B, Section 3.5, is not applicable to the SGS and HCGS cask loading operations (see also Table 3 of this appendix).
6. APPROVED CONTENTS	Procedural controls are used to ensure that the contents of the HI-STORM 100 Systems at the ISFSI meet the applicable fuel specifications and other requirements in HI-STORM Certificate of Compliance, Appendix B, Section 2.0. The detailed evaluation of compliance with CoC Condition 6 is provided in Table 3 of this appendix, which addresses compliance with the Approved Contents section of CoC Appendix B.
7. DESIGN FEATURES	Features or characteristics for the design and operation of the Salem/Hope Creek ISFSI, cask system, and ancillary equipment are in accordance with HI-STORM 100 System Certificate of Compliance, Appendix B, Section 3.0. The detailed evaluation of compliance with CoC Condition 7 is provided in Table 3 of this appendix, which addresses compliance with the Design Features section of CoC Appendix B.
8. CHANGES TO THE CERTIFICATE OF COMPLIANCE	Certificate of Compliance Condition No. 8 states the 10 CFR 72.244 regulatory requirement that the holder of the certificate who desires to make changes to the CoC, including appendices, must submit an application for amendment of the CoC to the NRC. This condition applies only to the CoC holder. Therefore, no action or implementing procedures are required by PSEG Nuclear.
9. SPECIAL REQUIREMENTS FOR FIRST SYSTEMS IN PLACE	**<u>CoC Amendments 2 and 3:</u>** CoC Condition 9 was not modified from Amendment 2 to Amendment 3. The CoC requirements were as follows in Amendments 2 and 3: "The heat transfer characteristics of the cask system will be recorded by temperature measurements for the first HI-STORM Cask Systems (for each thermally unique MPC basket design – MPC-24/24E/24F, MPC-32/32F, and MPC-68/68F/68FF) placed into service by any user with a heat load equal to or greater than 10 kW. An analysis shall be performed that demonstrates the temperature measurements validate the analytic methods and predicted thermal behavior described in Chapter 4 of the FSAR. "Validation tests shall be performed for each subsequent cask system that has a heat load that exceeds a previously validated heat load by more than 2 kW (e.g., if the initial test was conducted at 10 kW, then no additional testing is needed until the heat load exceeds 12 kW). No additional testing is required for a system after it has been tested at a heat load equal to or greater than 16 kW. "Each first time user of a HI-STORM 100 Cask System Supplemental Cooling System (SCS) that uses components or a system that is not essentially identical to components or

Table 1, CoC Conditions

Condition	Evaluation
9. SPECIAL REQUIREMENTS FOR FIRST SYSTEMS IN PLACE (cont'd)	a system that has been previously tested, shall measure and record coolant temperatures for the inlet and outlet of cooling provided to the annulus between the HI-TRAC and MPC and the coolant flow rate. The user shall also record the MPC operating pressure and decay heat. An analysis shall be performed, using this information, that validates the thermal methods described in the FSAR which were used to determine the type and amount of supplemental cooling necessary. Letter reports summarizing the results of each thermal validation tests and SCS validation test and analysis shall be submitted to the NRC in accordance with 10 CFR 72.4. Cask users may satisfy these requirements by referencing validation test reports submitted to the NRC by other cask users." **<u>CoC Amendments 2 and 3 Compliance Evaluation (Casks 1-12):</u>** The first part of this condition requires temperature monitoring and reporting to the NRC for HI-STORM 100 Systems loaded with decay heat between 10 kW and 16 kW to confirm the heat removal system is operating as designed. Other HI-STORM 100 System users have fulfilled this CoC condition by loading MPC-68-series canisters (including MPC-68 and -68FF) up to and over 16 kW. Specifically, Energy Northwest loaded an MPC-68-series canister with heat load greater than 16 kW at Columbia Generating Station and fulfilled this CoC condition for all other users under these amendments (Reference 6.12). The second part of this CoC requirement pertaining to the Supplemental Cooling System (SCS) applies only to general licensees using the Holtec HI-STORM 100 System to load high burnup (HBU) fuel (burnup > 45,000 MWD/MTU). In Amendments 2 and 3 of the CoC, Appendix A, LCO 3.1.4 requires the SCS to be used only if HBU fuel is loaded into the MPC. PSEG Nuclear did not load any HCGS MPCs in accordance with CoC Amendment 2 or 3 that contained HBU fuel. Therefore, this requirement of the CoC is not applicable to the first 12 HCGS casks loaded and placed into storage at the ISFSI in accordance with CoC Amendments 2 and 3. **<u>CoC Amendment 5:</u>** In CoC Amendment 5, Condition 9 was revised to replace the first two paragraphs with one new first paragraph. The first paragraph of Condition 9 now reads as follows: "The air mass flow rate through the cask system will be determined by direct measurements of air velocity in the overpack cooling passages for the first HI-STORM Cask Systems placed into service by any user with a heat load equal to or greater than 20 kW. The velocity will be measured in the annulus formed between the MPC shell and the overpack inner shell. An analysis shall be performed that demonstrates the measurements validate the analytic methods and thermal performance predicted by the licensing-basis thermal models in Chapter 4 of the FSAR."

Table 1, CoC Conditions

Condition	Evaluation
9. SPECIAL REQUIREMENTS FOR FIRST SYSTEMS IN PLACE (cont'd)	The previous third paragraph of this CoC condition (now the second paragraph) was not changed in CoC Amendment 5. **CoC Amendment 5 Compliance Evaluation (Casks 13 and higher)** HCGS Compliance Evaluation (Casks 13 and higher) HCGS did not load any high burnup fuel and none of the casks have a heat load above the thresholds requiring air mass flow rate measurements or use of the SCS. Therefore, this CoC condition does not apply for HCGS casks 13-16. Condition 9 for HCGS casks beyond number 16 that have heat loads greater than 20 kW or require use of the SCS, is covered by the compliance evaluation for SGS casks 17 and higher, because the same overpack design is used for both stations' fuel and the same SCS system would be placed into service when required. See SGS discussion below. SGS Compliance Evaluation (Casks 17 and higher) SGS has loaded casks with contents and/or heat loads requiring cask air mass flow rate testing (per this CoC condition) and use of the SCS (per LCO 3.1.4). For the air mass flow rate testing requirement, Reference 6.24 confirms that the air mass flow rate test performed at Arkansas Nuclear One (ANO) on a cask with heat load exceeding 20 kW is applicable to the casks at the PSEG Nuclear ISFSI. Therefore, the cask air mass flow rate test portion of this CoC condition for Amendment 5 is met. SCS inlet and outlet temperatures and water flow rates were measured and recorded in accordance with an approved station procedure in 2010 during Salem Unit 1 cask loading. Based on a review of the PSEG SCS performance data, it can be concluded that the system was operating in accordance with Appendix 2.C of the HI-STORM FSAR. A formal submittal to the NRC summarizing the SCS validation test and analysis is in progress and is expected to be transmitted in mid-2012.

Table 1, CoC Conditions

Condition	Evaluation
10. PRE-OPERATIONAL TESTING AND TRAINING EXERCISE	Dry run training is conducted using the same procedures that are used in loading casks with actual spent nuclear fuel. The dry run training program addresses each of the items in CoC Condition 10.a through 10.k, except as follows: • Condition 10.h, which pertains to transfer cask upending and downending, is not demonstrated because the cask loading procedures at SGS and HCGS do not require the fuel-loaded HI-TRAC transfer cask to be upended or downended. • Condition 10.g, which pertains to use of the Supplemental Cooling System (SCS), was not demonstrated at HCGS through 2010 because the SCS was not required by the CoC to be used for the HCGS spent fuel loaded through the first 16 casks.. SCS training and dry runs will be provided at SGS for the first loading campaign and, in the future at HCGS, if and when SCS use is required by the CoC for the fuel being loaded (see also discussion under CoC Condition 9, and LCO 3.1.4). • **CoC Amendment 2 (applicable to casks 1 - 4):** Condition 10.k, which pertains to fuel cooldown and unloading was not demonstrated at HCGS because this evolution has been previously demonstrated on the HI-STORM 100 System at other plants. By design, an MPC is never expected to have to be unloaded. If some unforeseen event requires the unloading of an MPC, PSEG has the administrative controls in place to acquire the necessary equipment and trained personnel to perform these operations (Reference 6.9). • **CoC Amendment 3 (casks 5 – 12):** Condition 10.k was revised to delete the phrase "cooling fuel assemblies" to reflect the revision of LCO 3.1.3, which no longer requires pre-cooling of the MPC cavity before re-flooding. No action is required for this revised CoC condition. • **CoC Amendment 5 (casks 13 and higher):** Condition 10.g, which pertains to SCS training and dry runs was revised to add "if applicable" to the training requirement to clarify that no training or dry run operations are required if the system is not being used. The system is not being used at HCGS based on the fuel selected for dry storage. The SCS is being used at SGS. PSEG will provide appropriate training and dry run operations for the SCS for the station where it is required by the CoC to be used.
11. EXEMPTION FROM 10 CFR 72.236(f) FOR SUPPLEMENTAL COOLING SYSTEM	This CoC condition pertains to the use of the Supplemental Cooling System (SCS) for on-site loading and transportation of high burnup (HBU) spent fuel (burnup > 45,000 MWD/MTU) in the HI-TRAC transfer cask. The NRC has granted an exemption from the 10 CFR 72.236(f) requirement that adequate heat removal capacity must be provided without reliance on an active cooling system. Because this is a simple statement acknowledging the exemption, no action to demonstrate compliance is required for this CoC condition.

Table 1, CoC Conditions

Condition	Evaluation
12. AUTHORIZATION	By virtue of holding three 10 CFR Part 50 licenses, PSEG Nuclear also holds a general license for the storage of spent fuel at an ISFSI pursuant to 10 CFR 72.210. This CoC condition states that general licensees are authorized to use the HI-STORM 100 System under a 10 CFR 72 general license and provides direction regarding use of previously approved amendments to the CoC. PSEG Nuclear used Amendment 2 to CoC 1014 to load the four HCGS casks in the 2006-07 loading campaign and Amendment 3 to load the eight HCGS casks in the 2008 loading campaign. CoC Amendment 5 was/will be used to load casks from HCGS and SGS in 2010 and later. Revisions to this report will address the use of future CoC amendments, as necessary. See Appendix 2 to this report for the CoC amendment, FSAR revision, and approved interim design and licensing basis changes applicable to each cask loading campaign and licensed storage system component.

Table 2, CoC Appendix A — Technical Specifications

Technical Specification	Evaluation
1.0 USE AND APPLICATION	The Use and Application section of Appendix A to the HI-STORM 100 CoC provides definitions of terms used in the technical specifications (TS) in Section 1.1, and explanatory information on the interpretation of logical connectors (e.g., AND and OR), completion times, and frequency in Sections 1.2, 1.3, and 1.4, respectively. This section provides the necessary information on how to interpret and implement the requirements in the TS and is used in training. No other compliance actions are required.
2.0 INTENTIONALLY BLANK	None
3.0 LIMITING CONDITIONS FOR OPERATION (LCO) APPLICABILITY	The HI-STORM 100 System CoC technical specification Limiting Conditions for Operation (LCOs) specify the minimum capability or level of performance that is required to assure that the HI-STORM 100 System can fulfill its safety functions. LCOs 3.0.1 through 3.0.5 provide the over-arching rules for complying with LCOs located elsewhere in the TS. HI-STORM 100 System Technical Specification LCOs and the Required Actions and Completion Times to be performed if an LCO is not met, are implemented through approved SGS, HCGS, Nuclear Common, and contractor procedures (see procedures listed under References 6.31, 6.54, 6.56, and 6.61).
3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY	The HI-STORM 100 System TS Surveillance Requirements (SRs) specify actions to be taken and acceptance criteria to be met to verify that equipment important to safety is operable. SRs 3.0.1 through 3.0.4 provide the over-arching rules for complying with the SRs elsewhere in the TS. HI-STORM 100 System Technical Specification SRs are implemented through approved SGS, HCGS, Nuclear Common, and contractor procedures (see procedures listed under References 6.31, 6.54, 6.56, and 6.61).
3.1 SFSC INTEGRITY	This section of CoC Appendix A provides LCOs for ensuring the long-term integrity of the MPC confinement boundary and the stored fuel. Each LCO is discussed individually below.
3.1.1 Multi-Purpose Canister (MPC)	The MPC is required to be dried, backfilled with helium, and the vent and drain port cover plates helium leak tested before declaring the system ready for MPC transfer to the HI-STORM overpack, and subsequent on-site transportation to the ISFSI. LCO 3.1.1 requires the MPC to be dry and helium filled during transport operations and storage operations. Surveillance Requirements (SRs) 3.1.1.1, 3.1.1.2, and 3.1.1.3 are used to assure that the MPC is dried and backfilled with helium in accordance with the applicable acceptance criteria in TS Tables 3-1 and 3-2, and leak-tested per ANSI N14.5-1997 before being declared ready for on-site transport and storage operations. Several changes to this TS were made in CoC Amendment 5, as described below, which apply to the loading of HCGS (casks 13 and higher) and SGS (casks 17 and higher).

Table 2, CoC Appendix A — Technical Specifications

Technical Specification	Evaluation
3.1.1 Multi-Purpose Canister (MPC) (cont'd)	**<u>CoC Amendment 5:</u>** 1) The LCO was revised to add a 40-hour limit for vacuum drying time of casks with heat loads between 23 kW and 28.74 kW and prohibit use of the vacuum drying system (VDS) on canisters with heats loads exceeding 28.74 kW (TS Table 3-1 further restricts VDS use to 26 kW). 2) Required Action A.2 was revised to replace "return the MPC to an analyzed condition" to "return the MPC to compliance with Table 3-1." 3) New Condition B and Action B.1 were added to reflect the time limit on vacuum drying. The remaining Conditions and Required Actions were re-lettered appropriately. 4) Required Action C.2 (previously B.2) was revised to add "by adding helium to or removing helium from the MPC" to the end of the action statement. 5) New Required Action C.2.2 was added to permit an option to Action C.2 to demonstrate by analysis that all limits for cask components and contents can be met in the event the helium backfill limit is not met. 6) Required Action D.2 (previously C.2) was revised to replace "return the MPC to an analyzed condition" to "return the MPC to compliance with SR 3.1.1.3." 7) SR 3.1.1.1 was revised to refer to the vacuum drying time limits for higher heat load casks. 8) SR 3.1.1.2 was revised to add a statement that re-performance of the SR is not required after successful completion of Required Action C.2.2. 9) SR 3.1.1.3 was revised to make a grammatical correction. The above changes to this technical specification have been reflected, as appropriate, in the SGS and HCGS cask loading procedures. At HCGS, the acceptance criteria for the MPC-68/68FF are applicable, and compliance with the MPC drying and backfilling acceptance criteria is demonstrated by procedure (Reference 6.31.6). At SGS, the acceptance criteria for the MPC-32/32F are applicable, and compliance with the MPC drying and backfilling acceptance criteria is demonstrated by procedure (Reference 6.56.6). Helium leakage testing of the vent and drain port cover plates is performed in accordance with ANSI N14.5 with a "leaktight" acceptance criterion, and is also demonstrated in the same procedures.
3.1.2 SFSC Heat Removal System	LCO 3.1.2 requires the natural ventilation heat removal system of the HI-STORM 100 System to be operable at all times during storage operations at the ISFSI. Surveillance Requirement 3.1.2.1 requires periodic inspection of the overpack inlet

Table 2, CoC Appendix A — Technical Specifications

Technical Specification	Evaluation
3.1.2 SFSC Heat Removal System (cont'd)	and outlet air ducts to verify that they are free of blockage. Alternately, a periodic check of the temperature rise of the air from the cask air inlet (or ambient) to a minimum of two cask air outlets may be performed to verify heat removal system operability. Several changes to this TS were made in CoC Amendment 5, as described below, which apply to HCGS (casks 13and higher) and SGS (casks 17 and higher). **<u>CoC Amendment 5:</u>** 1) A note was added to the Applicability to clarify that the SFSC heat removal system is operable provided 50% or more of the inlet and outlet vent areas are unblocked and available for flow or when air temperature measurements are met. 2) A new Condition A and Required Action A.1 were added to remove partial blockage less than 50%. No completion time is applicable because the heat removal system is still considered operable. Previous Condition A and Required Action A.1 were changed to B and B.1, respectively. Previous Condition B and Required Actions B.1, B.2.1, and B.2.2 were changed to C, C.1, C.2.1 and C.2.2, respectively. 3) The Completion Times for Required Actions C.2.1 and C.2.2 (previously B.2.1 and B.2.2) were revised to be heat load-dependent. 4) SR 3.1.2.1 was changed to SR 3.1.2. 5) The visual inspection part of the SR was revised to add "from solid debris or floodwater." 6) The temperature monitoring part of the SR was revised to separate the acceptance criterion for PWR and BWR fuel and increase the value from 126°F for all fuel to 155°F for PWR fuel and 137°F for BWR fuel. The casks at the Salem/Hope Creek ISFSI are equipped with the instrumentation needed to use the temperature monitoring option as the primary means of verifying heat removal system operability. HCGS personnel implement this Surveillance Requirement for all casks at the ISFSI, regardless of whether they contain HCGS or SGS fuel. Visual inspection of the inlet and outlet air ducts may be used as a backup method for meeting the LCO if the temperature monitoring system is inoperable or otherwise unavailable. Procedural controls are used to implement the SR and verify whether LCO 3.1.2 is met (Reference 6.31.25). An alarm response procedure is used to respond to any alarms from the temperature monitoring system (Reference 6.31.18). If the alarm is determined to be valid, an abnormal procedure is used to implement the Required Actions for not meeting the LCO (Reference 6.31.13).

Table 2, CoC Appendix A — Technical Specifications

Technical Specification	**Evaluation**
3.1.3 Fuel Cool-Down **(for casks 1-4)**	By design, the HI-STORM 100 System is never expected to be required to be unloaded of fuel. However, if MPC unloading is required for some unforeseen reason, LCO 3.1.3 (Amendment 2) requires the MPC cavity bulk helium temperature to be less than a specific value before re-flooding of the MPC is permitted in preparation for fuel removal in the spent fuel pool. Meeting this LCO precludes significant fuel quenching or MPC pressurization due to water flashing during re-flooding. Procedural controls are used to verify whether LCO 3.1.3 is met via implementation of SR 3.1.3.1 (Reference 6.31.9) and are also used to implement the Required Actions if the LCO is not met (Reference 6.31.26). SR 3.1.3.1 allows establishing the MPC cavity bulk helium temperature prior to re-flooding by analysis or by direct measurement. If the predicted or measured bulk helium temperature is above the LCO limit, any appropriate cooling method is acceptable to reduce the bulk helium temperature to below the LCO limit to allow re-flooding operations to proceed. See also Reference 6.9.
3.1.3 MPC Cavity Re-flooding **(for casks 5-20)**	This LCO (as revised in CoC Amendment 3 and remaining in Amendment 5) requires MPC cavity pressure to be less than 100 psig prior to, and during re-flooding. Meeting this LCO precludes significant fuel quenching or MPC pressurization due to water flashing during re-flooding. Procedural controls are used to verify that LCO 3.1.3 is met via implementation of SR 3.1.3.1 (References 6.31.9 and 6.56.9) and are also used to implement the Required Actions if the LCO is not met (Reference 6.61.1). SR 3.1.3.1 allows ensuring the MPC cavity pressure prior to re-flooding meets the LCO limit by analysis or by direct measurement. If the predicted or measured pressure is above the LCO limit, re-flooding must be stopped and cannot resume until the pressure is within the LCO limit and the MPC vent port is verified not to be blocked. Because there are HCGS casks at the ISFSI that were loaded to Amendments 2, 3 and 5 of the CoC, References 6.31.9 and 6.61.1 contain instructions for meeting both sets of requirements. See Appendix 2 for the CoC amendment applicable to each cask component serial number.
3.1.4 Supplemental Cooling System	**<u>CoC Amendments 2 and 3 (casks 1 – 12):</u>** The Supplemental Cooling System (SCS) is required to be operable when high burnup (HBU) fuel (> 45,000 MWD/MTU) is in an MPC inside the HI-TRAC transfer cask. Use of the SCS ensures the HBU fuel cladding temperature remains below the applicable limit during on-site transfer cask operations. The SCS is not required for on-site transfer cask operations if all of the fuel in the MPC is burned less than or equal to 45,000 MWD/MTU. No HBU fuel was placed into dry storage in the first 12 HCGS casks. Thus this LCO is not applicable to the first 12 casks of HCGS fuel in storage at the ISFSI.

Table 2, CoC Appendix A — Technical Specifications

Technical Specification	Evaluation
3.1.4 Supplemental Cooling System (cont'd)	**CoC Amendment 5 (casks 13 and higher):** The applicability of this LCO was modified to require SCS use for casks with a heat load greater than 28.74 kW in addition to any cask containing at least one high burnup fuel assembly. In the 2010 cask loading campaign at HCGS, no HBU fuel was placed in storage nor were any casks exceeding 28.74 kW heat loaded. Thus, this LCO was not applicable for HCGS casks 13 – 16. At SGS, HBU fuel will be loaded or casks may exceed the threshold heat load for SCS. Thus, the SCS will be required at SGS and its use is governed by procedure (References 6.56.6, 6.56.7 and 6.56.8)
3.2 SFSC RADIATION PROTECTION	This section of CoC Appendix A provides one LCO that addresses radiological controls for the HI-TRAC transfer cask.
3.2.1 Deleted	None.
3.2.2 Transfer Cask Surface Contamination	LCO 3.2.2 establishes limits on loose radioactive contamination for the HI-TRAC transfer cask if MPC transfer operations occur outside of the "Fuel Building." The "Fuel Building" is defined in the HI-STORM technical specifications as the "site-specific power plant facility, governed by the regulations of Part 50, where the loaded overpack or transfer cask is transferred to or from the transporter." At HCGS, the "Fuel Building" is the secondary containment of the Reactor Building, where MPC transfer from the HI-TRAC transfer cask to the HI-STORM overpack takes place on elevation 102 ft. At SGS, the "Fuel Building" is the Fuel Handling Building where MPC transfer from the HI-TRAC transfer cask to the HI-STORM overpack takes place in the truck bay on elevation 100 ft. This LCO includes a note that states the LCO is not applicable to the transfer cask if MPC transfer operations occur inside the "Fuel Building." Because MPC transfer operations take place in the HCGS Reactor Building and the SGS Fuel Handling Building (the "Fuel Building" for this LCO), this LCO does not apply to the transfer cask, but does apply to the MPC. SR 3.2.2.1 is used to verify that the LCO limits on loose contamination are met for accessible portions of the MPC prior to transport operations. Procedural controls are used to implement the SR and ensure that LCO 3.2.2 is met. Procedural controls are also used to implement Required Actions if the LCO is not met (References 6.31.5, 6.31.20, 6.31.22, 6.33.1, 6.56.5, and 6.56.15).
3.2.3 Deleted	No action required.
3.3 SFSC CRITCALITY CONTROL	This section of CoC Appendix A provides an LCO to assure the physical environment in the MPC is consistent with the supporting criticality analysis.
3.3.1 Boron Concentration	LCO 3.3.1 establishes minimum soluble boron concentration requirements for the water in the MPC during loading and unloading of PWR fuel in the plant spent fuel pool. This LCO is not applicable to HCGS dry cask storage operations because HCGS is a BWR plant. Thus, no action is required for HCGS spent fuel storage campaigns. This LCO is required for loading SGS spent fuel into the PWR MPCs used in the HI-STORM 100 System. The requirements of this LCO for MPC-32 are implemented by procedure (References 6.56.5, 6.56.6, and 6.56.9).

Table 2, CoC Appendix A — Technical Specifications

Technical Specification	Evaluation
Table 3-1	CoC Table 3-1 augments LCO 3.1.1 by providing the maximum permissible heat loads for use of the vacuum drying system (VDS), above which the Forced Helium Dehydration (FHD) system must be used for MPC drying. The FHD System is also required to dry any MPCs containing one or more HBU fuel assemblies. In CoC Amendment 5 the table was revised: 1) Different heat load thresholds for the various MPC models are established to require FHD System use for MPCs containing no HBU fuel. Below these heat load thresholds, MPCs containing no HBU fuel may be dried using the VDS. (Note that the VDS threshold of 26 kW in Table 3-1 is lower than the VDS threshold of 28.74 kW in LCO 3.1.1 – PSEG uses the more conservative limit of 26 kW.) 2) A new maximum heat load value of 36.9 kW is provided for any MPC. 3) A new Note 3 is added to the table that requires the HI-TRAC-to-MPC annulus to either be filled or continuously flushed with water during vacuum drying operations, based on heat load thresholds. The HCGS cask loading procedure implementing these requirements has been revised, as required, to reflect these new requirements when vacuum drying is used (Reference 6.31.6). All SGS MPCs loaded in accordance with CoC Amendment 5 will be dried using the FHD System, regardless of heat load. Thus, the vacuum drying requirements do not apply to SGS MPCs. The SGS drying and sealing procedure implements the MPC drying requirements (Reference 6.56.6).
Table 3-2	CoC Table 3-2 augments LCO 3.1.1 by providing the required helium backfill pressure ranges based on MPC model and heat load. The SGS and HCGS cask loading procedures have been appropriately revised to reflect these requirements (Reference 6.31.6 and 6.56.6).
4.0 INTENTIONALLY BLANK	None.
5.0 ADMINISTRATIVE CONTROLS AND PROGRAMS	This section of CoC Appendix A provides requirements for certain programmatic controls necessary to ensure the dry storage system is used on site in a manner consistent with the regulations and the generic cask design. Each program is addressed individually below.
5.1 Deleted	None.
5.2 Deleted	None.
5.3 Deleted	None.

Table 2, CoC Appendix A — Technical Specifications

Technical Specification	Evaluation
5.4 Radioactive Effluent Control Program	CoC Administrative Program 5.4 requires the general licensee to maintain a radioactive effluent control program in accordance with 10 CFR 72.44(d), including an environmental monitoring program and annual reports. The HI-STORM 100 System does not release any radioactive materials or require any radioactive waste treatment systems because the MPC is leak tight. The design of the MPC and the surveillance requirements of LCO 3.1.1, "Multi-Purpose Canister (MPC)," provide assurance that there are no radioactive effluents from the ISFSI under all normal, off-normal, and credible accident conditions. Therefore, specific operating procedures for control of radioactive effluents and maintenance of radioactive waste treatment systems are not required for the ISFSI. The Radiological Environmental Monitoring Program (REMP) requirements of SGS and HCGS have been expanded to include the ISFSI. The Radiological Effluent Controls (REC) program, as part of the Offsite Dose Calculation Manual (ODCM), implements the procedural details of the REMP. Because the casks used at the on-site ISFSI provide confinement yielding no radioactive gaseous or liquid effluents, assessment of offsite collective dose due to ISFSI storage operations is limited to direct and reflected radiation. Thermoluminescent dosimeters (TLDs) or equivalent will be used to monitor direct gamma radiation levels in and around the Salem/Hope Creek ISFSI site. Placement of environmental monitoring station TLDs is in accordance with the approved REC and ODCM. PSEG Nuclear submits dry cask storage effluent reports for the ISFSI in accordance with 10 CFR 72.44(d)(3) requirements. Annual Radioactive Effluent Release Reports (ARERRs) for the reactor site are submitted to the NRC to meet 10 CFR 50 requirements. Radioactive effluent release information related to dry cask storage activities at the ISFSI is incorporated in the ARERR. The addition of Salem fuel has no impact on the programs already in place that are used to monitor the ISFSI and make the appropriate reports.

Table 2, CoC Appendix A — Technical Specifications

Technical Specification	Evaluation
5.5 Cask Transport Evaluation Program	CoC Administrative Program 5.5 requires the general licensee to evaluate the conditions pertaining to transporting the fuel-loaded cask between the Part 50 facility and the ISFSI. The purpose of this program is to ensure one of two things: 1. The combination of the physical characteristics of the heavy haul path and the carry height for the cask are such that a cask drop event would be bounded by the design basis cask drop event described in the FSAR, or: 2. The cask transporter design features meet certain requirements that allow a drop event to be considered non-credible. Movement of a fuel-loaded HI-STORM overpack and MPC outside of the SGS or HCGS structures governed by 10 CFR 50 is performed in accordance with approved PSEG Nuclear procedures (References 6.31.3, 6.31.8, 6.56.3, and 6.56.8). The HI-STORM CoC requirement for a Cask Transport Evaluation Program (CTEP) is implemented by the cask transportation procedure and the design attributes of the vertical cask transporter (VCT) used to move the fuel-loaded overpack from the Hope Creek Reactor Building or the Salem Fuel Handling Building to the ISFSI. **<u>HCGS Compliance Evaluation</u>** The HI-STORM overpack containing a loaded MPC is moved outside of the HCGS Reactor Building receiving bay on a low profile transporter (LPT) to a location where the VCT can access the cask. The LPT supports the HI-STORM overpack from underneath. Therefore, consistent with Technical Specification 5.5, the Cask Transport Evaluation Program does not apply to movement of a loaded HI-STORM overpack and MPC on the LPT while in the Reactor Building receiving bay and just outside the receiving bay door. The HI-STORM overpack is moved out of the HCGS Reactor Building without its lid installed due to receiving bay door clearance limitations. The lid is installed as soon as possible after the overpack exits the Reactor Building while the cask is still on the LPT. The lid installation occurs within approximately 50 feet of the Reactor Building door and is expected to be complete in approximately 1-2 hours. Procedures and training include instructions to complete the lid installation without interruption (Reference 6.31.3). The outdoor lid installation is not addressed in the HI-STORM 100 System FSAR except for MPC transfers conducted in a Cask Transfer Facility. Therefore, as part of the Design Change Package for dry cask loading operations, a 10 CFR 72.48 screening was performed to authorize the implementation of this operating evolution as a deviation from the HI-STORM 100 System FSAR (Reference 6.34).

Table 2, CoC Appendix A — Technical Specifications

Technical Specification	Evaluation
5.5 Cask Transport Evaluation Program (cont'd)	The loaded overpack is moved out of the Reactor Building on the LPT, outfitted with a Holtec Earthquake Mitigator (HERMIT). The HERMIT ensures that a seismic event will not cause the cask to tip over in the receiving bay or on its journey to the egress pad. The HI-STORM overpack and MPC are moved from just outside the HCGS Reactor Building to the ISFSI pads using the VCT. The VCT is designed in accordance with ANSI N14.6 and has redundant drop protection design features (Reference 6.8). Therefore, in accordance with Technical Specification 5.5, no maximum lift height is established and a cask drop need not be postulated along the heavy haul path. Therefore, Technical Specification 5.5 does not apply to movements of a fuel-loaded HI-STORM overpack and MPC with the VCT. As a defense-in-depth measure, the heavy haul path between the HCGS Reactor Building receiving bay and the ISFSI pad has been evaluated to determine whether the hardness of the path surface is more energy absorptive (i.e., as hard or less hard) than the surface modeled in the design basis cask drop analysis in the HI-STORM FSAR. That evaluation (Reference 6.35) revealed that the entire haul path, except for the egress pad, is bounded by the surface modeled in the FSAR analysis (Reference 6.2, Table 2.2.9). Therefore, the 11-inch cask drop analysis described in the FSAR is bounding for all locations on the heavy haul path. Even though a cask drop is not required to be postulated, the lift height of the cask during transport to the ISFSI pad is maintained by procedure as low as practicable above the surface below for prudence (References 6.31.3 and 6.31.8). A fuel-loaded HI-TRAC transfer cask is never moved outside of HCGS structures that are governed by 10 CFR 50. Therefore, Technical Specification 5.5 does not apply to movements of a fuel-loaded HI-TRAC transfer cask and MPC. **SGS Compliance Evaluation:** The HI-STORM overpack containing a loaded MPC is moved outside of the SGS Unit 1 or 2 Fuel Handling Building truck bay on a zero profile transporter (ZPT) to a location where the VCT can access the cask. The ZPT supports the HI-STORM overpack from underneath. Therefore, consistent with Technical Specification 5.5, the Cask Transport Evaluation Program does not apply to movement of a loaded HI-STORM overpack and MPC on the ZPT while in the Fuel Handling Building truck bay and just outside the truck bay door. The HI-STORM overpack is moved out of the SGS Fuel Handling Building without its lid installed due to truck bay door clearance limitations. The lid is installed as soon as possible after the overpack exits the Fuel Handling Building while the cask is still on the ZPT. The lid installation occurs within approximately 50 feet of the Fuel Handling Building door and is expected to be

5.5 Cask Transport Evaluation Program (cont'd)	complete in approximately 1-2 hours. Procedures and training include instructions to complete the lid installation without interruption (Reference 6.56.3). The HI-STORM overpack and MPC are moved from just outside the SGS Unit 1 or 2 Fuel Handling Building to the ISFSI pads using the VCT. The VCT is designed in accordance with ANSI N14.6 and has redundant drop protection design features (Reference 6.8). Therefore, in accordance with Technical Specification 5.5, no maximum lift height is established and a cask drop need not be postulated along the heavy haul path. Therefore, Technical Specification 5.5 does not apply to movements of a fuel-loaded HI-STORM overpack and MPC with the VCT. Even though a cask drop is not required to be postulated, the lift height of the cask during transport to the ISFSI pad is maintained by procedure as low as practicable above the surface for prudence (Reference 6.56.3 and 6.56.8). A fuel-loaded HI-TRAC transfer cask is never moved outside of SGS structures that are governed by 10 CFR 50. Therefore, Technical Specification 5.5 does not apply to movements of a fuel-loaded HI-TRAC transfer cask and MPC at SGS.
5.6 Deleted	None.
5.7 Radiation Protection Program	The SGS and HCGS Radiation Protection Programs have been augmented to address fuel loading, cask handling, and ISFSI operations, as applicable. Implementing procedures ensure that each of the elements of the program required by Technical Specification 5.7 is addressed (References 6.31.20 through 22, 6.56.15, 6.56.16, and 6.33.1). Programmatic changes to the PSEG radiation protection program are discussed in Section 5.7.1.4 of this report.

Table 3, CoC Appendix B — Approved Contents and Design Features

<table>
<tr><th>Approved Contents and Design Features</th><th>Evaluation</th></tr>
<tr><td>1.0 DEFINITIONS</td><td>This section of CoC Appendix B provides definitions of terms used elsewhere in the appendix. Defined terms are shown in capitalized text.</td></tr>
<tr><td>2.0 APPROVED CONTENTS</td><td>This section of CoC Appendix B provides the limits for the material permitted to be stored in the HI-STORM 100 System. It includes limits on such things as fuel physical parameters, cooling time, enrichment, burnup, decay heat, and location of assemblies and non-fuel hardware in the MPC.</td></tr>
<tr><td>2.1 Fuel Specifications and Loading Conditions</td><td><u>Specification 2.1.1.a</u>

Specification 2.1.1.a requires that all fuel assemblies and non-fuel hardware from SGS and HCGS to be loaded into HI-STORM 100 casks and deployed at the on-site ISFSI meet the limits in Table 2.1-1 and other referenced tables. For HCGS spent fuel, the limits specified for the BWR MPC-68 and –68FF apply. These requirements are implemented through the HCGS fuel characterization and selection procedures (References 6.31.14 and 6.31.15). For SGS spent fuel, the limits specified for the PWR MPC-32 and –32F apply. These requirements are implemented through the SGS fuel characterization and selection procedures (References 6.56.11 and 6.56.12).

Procedural controls are used to ensure that the spent fuel assemblies stored in the HI-STORM 100 casks include only those fuel assemblies that meet the fuel limits specified in CoC Appendix B, Section 2.1.1 and associated tables (References 6.31.14, 6.31.15, 6.56.11, and 6.56.12). Control and verification of the movement and location of fuel assemblies in the MPC is also controlled by procedure (References 6.31.16, 6.31.17, 6.56.14, and 6.56.17).

Past and (to the extent it is known) future HCGS and SGS spent fuel has been evaluated for storage in the HI-STORM 100 System. HCGS spent fuel physical parameters are bounded by the following array/classes as shown in HI-STORM CoC Appendix B, Table 2.1-3:

• GE7 (8x8 with two water rods): Array/Class 8x8C
• GE9 (8x8 with one central water rod): Array/Class 8x8D
• SVEA-96+: Array/Class 10x10C
• GE14: Array/Class 10x10A

The HCGS fuel characterization and selection procedures (References 6.31.14, and 6.31.15) are used to ensure the initial enrichment, cooling time, decay heat, and burnup of the assemblies chosen for dry storage comply with the limits in the CoC.</td></tr>
</table>

Table 3, CoC Appendix B — Approved Contents and Design Features

<table>
<tr><th>Approved Contents and Design Features</th><th>Evaluation</th></tr>
<tr><td>2.1 Fuel Specifications and Loading Conditions (cont'd)</td><td>SGS spent fuel types are listed below:

• Westinghouse Standard
• Westinghouse V5H
• Westinghouse Vantage +
• Westinghouse RFA
• Westinghouse RFA-2
• Westinghouse OFA

The physical parameters of these fuel types are bounded by array/classes 17x17A and 17x17B as defined in HI-STORM CoC Appendix B, Table 2.1-2. The SGS fuel characterization and selection procedures (References 6.56.11, and 6.56.12) are used to ensure the initial enrichment, cooling time, decay heat, and burnup of the assemblies chosen for dry storage comply with the limits in the CoC.

<u>Specification 2.1.1.b</u>

Specification 2.1.1.b establishes loading requirements for stainless steel clad fuel mixed with zirconium-based clad fuel. This requirement is not applicable to the Salem/Hope Creek ISFSI because SGS and HCGS fuel rods are all clad with zirconium-based material.

<u>Specification 2.1.1.c</u>

Specification 2.1.1.c establishes loading requirements for BWR fuel in the 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A array/classes. These requirements are not applicable to the Salem/Hope Creek ISFSI because the specified array/classes do not apply to HCGS spent fuel. See the discussion for CoC Section 2.1.1.a above for the array/classes applicable to HCGS spent fuel. This specification does not apply to SGS because it is a PWR plant.

<u>Specification 2.1.1.d</u>

Specification 2.1.1.d establishes loading requirements for BWR fuel in array/classes 10x10D and 10x10E with stainless steel channels. These requirements are not applicable to the Salem/Hope Creek ISFSI because all HCGS 10x10 fuel is in array/class 10x10A or 10x10C as discussed above for CoC Section 2.1.1.a. This specification does not apply to SGS because it is a PWR plant.</td></tr>
</table>

Table 3, CoC Appendix B — Approved Contents and Design Features

Approved Contents and Design Features	Evaluation
2.1 Fuel Specifications and Loading Conditions (cont'd)	**Specifications 2.1.2 and 2.1.3** These requirements pertain to uniform fuel loading and regionalized fuel loading in the MPC, respectively. The location of each spent fuel assembly in the MPC is documented by fuel assembly identification number and MPC storage location to verify compliance with the applicable uniform or regionalized storage requirements. Each spent fuel assembly that is to be loaded into an MPC is concurrently verified to be the correct fuel assembly prior to moving the assembly, and the storage location in the MPC is concurrently verified to be the correct location prior to inserting the fuel assembly into the MPC. A final independent verification of the fuel assembly identification and the MPC fuel storage location is made prior to MPC lid installation. All cask fuel selection and loading activities, including verification of location in the MPC, are governed by the spent fuel characterization and selection procedures (References 6.31.14 through 6.31.17 for HCGS and 6.56.11, 6.56.12, 6.56.14, and 6.56.17 for SGS).
2.2 Violations	Procedures are used to comply with the requirements of CoC Appendix B, Section 2.2 in the event that a fuel loading violation occurs. Fuel loading violations have been incorporated into the site Reportability Action Level (RAL) process to ensure such violations are reported in accordance with this CoC requirement (SGS and HCGS Event Classification Guides, RAL 11.1.3.a).
2.3 Not Used	None required.
2.4 Decay Heat, Burnup, and Cooling Time Limits for ZR-Clad Fuel	This section of CoC Appendix B specifies the limits on decay heat, burnup, and cooling time for fuel permitted to be loaded into the HI-STORM 100 System. All cask fuel selection and loading activities, including verification of these fuel limits, are governed by procedure (References 6.31.15 and 6.56.12).
3.0 DESIGN FEATURES	This section of CoC Appendix B establishes requirements on-site conditions and certain cask and ancillary equipment design features important to safe spent fuel storage at the ISFSI using the generically certified HI-STORM 100 System. Because the ISFSI is located near HCGS, the site parameters in the CoC are compared the information in the HCGS UFSAR and are generally not compared to the information in the SGS UFSAR. If a site condition is specified in the CoC that could apply to the cask during its movement in or around SGS, then the site parameters in the SGS UFSAR would also be considered.
3.1.1 Site Location	Specification 3.1.1 is a simple statement that reiterates the permission granted in 10 CFR 72, Subpart K for 10 CFR 50 license holders to operate an ISFSI under a Part 72 general license using an NRC-certified cask. No further evaluation is required. There is one Part 72 general license (Docket 72-0048) that applies to dry storage of spent fuel discharged from all three site reactors at the ISFSI.

Table 3, CoC Appendix B — Approved Contents and Design Features

Approved Contents and Design Features	Evaluation
3.2 Design Features Important for Criticality Control	This CoC section establishes limits for certain design features deemed important to criticality control by the NRC for the various HI-STORM 100 System MPC models. Specifications 3.2.1 and 3.2.4 apply to 24-assembly PWR MPCs and are not evaluated further at this time because SGS uses the MPC-32. Specification 3.2.3 also does not apply to HCGS fuel because these requirements apply to a specialty BWR MPC (MPC-68F), which was custom-designed for a particular type of BWR fuel not used at HCGS. HCGS fuel may be stored in the MPC-68 and/or -68FF model canisters governed by Specification 3.2.2 as discussed below. Specifications 3.2.6 through 3.2.8 apply to all MPC models used for storage of spent fuel in the HI-STORM 100 System and are evaluated below for use at SGS and HCGS. SGS and HCGS exclusively use MPCs equipped with METAMIC™ neutron absorber. Therefore, CoC requirements related to Boral neutron absorber are not applicable and Boral-equipped MPCs may not be used to store SGS or HCGS spent fuel without a revision to this report.
3.2.2 MPC-68 and MPC-68FF	This specification establishes the following limits on the MPC-68/68FF model fuel basket design and fabrication used to store HCGS spent fuel: 1. Fuel cell pitch: $\geq$ 6.43 inches 2. ^{10}B loading in the METAMIC™ neutron absorber: $\geq$ 0.0310 g/cm^2 The fuel cell pitch and ^{10}B loading of the METAMIC™ neutron absorbers in the MPC are verified as part of MPC fabrication. Certification that each MPC meets these technical specification limits is provided by the CoC holder (Holtec International) in the Component Completion Record (CCR) for each serial number MPC. Each fabricated MPC-68/68FF is quality-control checked to ensure that it meets the specific design features for criticality and certified as such. A CCR cannot be issued if an as-built MPC does not meet these CoC design feature requirements. CCRs for each loaded MPC are part of the quality document file for the hardware provided by the CoC holder.

Table 3, CoC Appendix B — Approved Contents and Design Features

Approved Contents and Design Features	Evaluation
3.2.2 MPC-32 and MPC-32F	This specification establishes the following limits on the MPC-32/32F model fuel basket design and fabrication used to store SGS spent fuel: 1. Fuel cell pitch: $\geq$ 9.158 inches 2. ^{10}B loading in the METAMIC™ neutron absorber: $\geq$ 0.0310 g/cm^2 The fuel cell pitch and ^{10}B loading of the METAMIC™ neutron absorbers in the MPC are verified as part of MPC fabrication. Certification that each MPC meets these technical specification limits is provided by the CoC holder (Holtec International) in the Component Completion Record (CCR) for each serial number MPC. Each fabricated MPC-32/32F is quality-control checked to ensure that it meets the specific design features for criticality and certified as such. A CCR cannot be issued if an as-built MPC does not meet these CoC design feature requirements. CCRs for each loaded MPC are part of the quality document file for the hardware provided by the CoC holder.
3.2.6 Fuel Spacers	Specification 3.2.6 requires that fuel spacers be sized to ensure that the active fuel region of intact fuel assemblies remains within the neutron poison region of the MPC basket with water in the MPC. All HCGS fuel nominally ranges from 176.2 to 176.4 inches in length (Reference 6.42). Therefore, consistent with HI-STORM FSAR Table 2.1.10, no fuel spacers are required to maintain the active fuel region of HCGS spent fuel in the appropriate location in the MPC with respect to the neutron absorber. SGS spent fuel assemblies containing inserts have a maximum length of 167.3 inches, including inserts. Fuel assemblies without inserts are slightly shorter. Spacers are provided for SGS fuel, both with and without inserts in accordance with Table 2.1.9 of the HI-STORM FSAR. The note under that table specifies that "Fuel spacers shall be sized to ensure that the active fuel region of intact fuel assemblies remains within the neutron poison region of the MPC basket with water in the MPC." The selection of fuel spacers is governed by procedure (Reference 6.56.13)
3.2.7 Boron Carbide Content	Specification 3.2.7 requires the boron carbide (B_4C) content in the METAMIC™ neutron absorber to be $\leq$ 33.0 wt. %. Similar to the fuel basket design requirements in Specification 3.2.2, the boron carbide content in the METAMIC™ neutron absorbers is verified as part of MPC fabrication. Certification that each MPC meets this technical specification limit is provided by Holtec in the CCR for each serial number MPC.

Table 3, CoC Appendix B — Approved Contents and Design Features

Approved Contents and Design Features	Evaluation
3.2.8 Neutron Absorber Tests	Specification 3.2.8 incorporates the language in HI-STORM FSAR Section 9.1.5.3 pertaining to neutron absorber testing into the CoC by reference. Neutron absorber testing is verified to meet these requirements by the CoC holder and documented in the CCR for each serial number MPC and/or maintained in Holtec's records management system.
3.3 Codes and Standards	This specification establishes the governing codes for the HI-STORM 100 System. The governing code for the construction and structural design of the HI-STORM 100 System MPC, HI-TRAC transfer cask, and the metal components in the HI-STORM overpack is the 1995 edition of the ASME Boiler and Pressure Vessel Code (ASME Code), with addenda through 1997, except for Sections V and IX. The latest effective editions of Sections V and IX may be used for activities governed by those sections (NDE and welding). The governing code for the concrete in the HI-STORM overpack is American Concrete Institute (ACI) 349-1985, as clarified in cask FSAR Appendix 1.D. NRC-approved alternatives to the ASME Code are listed in Table 3-1 of this CoC section. New or revised alternatives must be submitted to the NRC for approval prior to implementation in accordance with CoC Section 3.3.2. Alternatives to the ASME Code are requested by the CoC holder. Holtec assures that all applicable Code requirements are met during fabrication of the cask components. No PSEG action or further evaluation is required.
3.4 Site-Specific Parameters and Analyses	This CoC section establishes various requirements to be evaluated against site-specific conditions at the plant to ensure the generic cask design is bounding for the site. Each parameter is discussed separately below.
3.4.1 Average Site Temperature	The maximum average yearly temperature on-site must not exceed 80°F. As documented in Table 2.3-12 of the HCGS UFSAR, the mean maximum average yearly annual temperature at the HCGS site is 53.1°F, which is less than the 80°F acceptance criterion, and is therefore in compliance with the CoC.
3.4.2 Extreme Site Temperature	The 3-day average temperature extremes must be greater than -40°F and less than 125°F. The lowest and highest hourly temperatures measured at HCGS site are -1°F and 94°F, respectively, as stated in HCGS UFSAR Table 2.3-12. The low temperature extremes at HCGS site (averaged over a 3 day period) are greater than –40°F, and the high temperature extreme is less than 125°F. Therefore, the HCGS site temperature extremes are in compliance with the CoC.

Table 3, CoC Appendix B — Approved Contents and Design Features

Approved Contents and Design Features	Evaluation
3.4.3 Seismic Criteria	The seismic criteria in Section 3.4.3 of Appendix B to the HI-STORM CoC are presented first as a test to determine whether the site seismic accelerations require the HI-STORM casks to be anchored to the ISFSI pad. If yes, specific criteria for the cask anchorage design must be met. If no, the casks may be deployed in a free-standing configuration, subject to meeting a specific inequality pertaining to cask sliding and overturning on the ISFSI pad. The design basis earthquake (DBE) resultant horizontal and vertical accelerations at the on-site ISFSI storage pad are less than the values in Specification 3.4.3.c.i, as discussed in Section 5.2.1.5 of this report. Therefore, the casks may be deployed in the free-standing mode at the Salem/Hope Creek ISFSI. The site seismic accelerations do not meet the inequality for free-standing casks in Specification 3.4.3.a. The detailed evaluation of DBE effects at the ISFSI storage pad, including a discussion of the alternative to meeting the inequality and the evaluation of degraded pad/cask interface friction (such as due to icing) is provided in Section 5.2.1.5 of this report.
3.4.4 Flooding	A flood water velocity of 15 ft/sec at the cask location and a submergence depth of 125 feet of water must not be exceeded. The HI-STORM 100 System casks at the ISFSI storage pads are not subject to submergence due to flooding to a depth in excess of 125 feet. The 15 fps flood water velocity is also not exceeded. A detailed description of the evaluation of flooding conditions at the Hope Creek site, including hurricane-induced wave action, is provided in Section 5.4.1.3 of this report.
3.4.5 Fire and Explosion	The potential for fire and explosion shall be addressed based on site-specific considerations. This includes the condition that the on-site transporter fuel tank will contain no more than 50 gallons of diesel fuel while handling a loaded overpack or transfer cask. The vertical cask transporter (VCT) that is used to move a loaded cask to the on-site ISFSI storage pad and the prime mover used to pull the HI-STORM overpack out of the HCGS Reactor Building on the LPT and out of the Fuel Handling Buildings at SGS on the ZPT have fuel tanks that are limited by design to hold no more than 50 gallons of diesel fuel (References 6.32.7 and 6.43). The potential for fires and explosions, based on Hope Creek site-specific hazards and the transport route between the Reactor Building and the ISFSI considerations are addressed in Section 5.4.1.1. The potential for fires and explosions, based on SGS site-specific hazards and the transport route between the Fuel Handling Buildings and the HCGS portion of the heavy haul path are addressed in Section 5.4.1.1.

Table 3, CoC Appendix B — Approved Contents and Design Features

Approved Contents and Design Features	Evaluation
3.4.6 Cask Drop and Tipover	For free-standing casks, the ISFSI pad shall be verified by analysis to limit cask deceleration during design basis drop and non-mechanistic tip-over events to less than or equal to 45 g's at the top of the MPC fuel basket. Analyses shall be performed using methodologies consistent with those described in the HI-STORM 100 FSAR. A lift height above the ISFSI pad is not required to be established if the cask is lifted with a device designed in accordance with ANSI N14.6 and having redundant drop protection features. The ISFSI pad thickness, concrete compressive strength, and reinforcing bar design meet the limits for Set 'A' in HI-STORM FSAR Table 2.2.9. However the ISFSI pad subgrade modulus of elasticity exceeds the 28,000 psi limit in the HI-STORM FSAR (Reference 6.18, Attachment J). This CoC specification permits a site-specific drop and tipover analysis to be performed if this is the case. Because the VCT is designed in accordance with ANSI N14.6 with redundant drop protection features (Reference 6.8), a cask drop is not postulated or analyzed. However, the non-mechanistic tipover must be analyzed. The site-specific analysis of a HI-STORM 100S Version B overpack tipover onto the ISFSI pad was performed. The results of the analysis show that the deceleration at the top of the fuel for this event is 39.2 g's (Reference 6.36). This value is less than the HI-STORM 100 System design basis value of 45 g's. Therefore, this CoC requirement is met. Because the HI-STORM overpacks will be handled with a device designed in accordance with ANSI N14.6 and having redundant drop protection features (Reference 6.8), there is no lift height restriction above the ISFSI pad. However, as a defense-in-depth measure, the cask will be carried no higher and no longer than necessary above the pad surface. During construction of the ISFSI pads, a section of the west pad (Pad No.1, Section 1A) was found to have a problem with water ponding. The repair for this problem was performed with grout, which has a compressive strength greater than the maximum permitted value of 4,200 psi. This repair has been evaluated by Holtec as a supplier manufacturing deviation report (SMDR No. 1410, Rev. 2) and found to be acceptable as-is via Holtec 72.48 No. 778, Revision 2. Therefore, this section of pad may be used to deploy casks.
3.4.7 Berms and Shield Walls	In cases where engineered features (i.e., berms and shield walls) are used to ensure that the requirements of 10CFR72.104(a) are met, such features are to be considered important to safety and must be evaluated to determine the applicable Quality Assurance category. No engineered features such as berms or shield walls are credited in the dose analysis performed to demonstrate compliance with the dose limits of 10 CFR 72.104(a) for the Salem/Hope Creek ISFSI. Therefore, this requirement is not applicable.

Table 3, CoC Appendix B — Approved Contents and Design Features

Approved Contents and Design Features	Evaluation
3.4.8 Working Area Ambient Temperature	Loading operations, transport operations, and unloading operations shall only be conducted with working area ambient temperatures greater than or equal to 0°F. Procedures are used to ensure that the working area ambient temperature is greater than or equal to 0° F during loading operations, transport operations, and unloading operations (References 6.31.3, 6.31.5, 6.31.7, 6.31.8, 6.31.9, 6.56.3, 6.56.5, 6.56.7, 6.56.8, and 6.56.9).
3.4.9 Site-Specific Events	For those users whose site-specific design basis includes an event or events (e.g., flood) that result in the blockage of any overpack inlet or outlet air ducts for an extended period of time (i.e., longer than the total Completion Time of LCO 3.1.2), an analysis or evaluation may be performed to demonstrate adequate heat removal is available for the duration of the event. Adequate heat removal is defined as fuel cladding temperatures remaining below the short-term temperature limit. If the analysis or evaluation is not performed, or if fuel cladding temperature limits are unable to be demonstrated by analysis or evaluation to remain below the short-term temperature limit for the duration of the event, provisions shall be established to provide alternate means of cooling to accomplish this objective. The evaluation of flooding at the Salem/Hope Creek ISFSI site that could submerge any overpack inlet or outlet air ducts for an extended period of time is discussed in Sections 5.4.1.3 and 5.4.1.4 of this report. In summary, adequate heat removal is maintained for the duration of the flood event.
3.4.10 Fuel Cladding Exposure to Air (casks 5 - 20)	This requirement was added in CoC Amendment 3. Fuel cladding, while in the MPC, must be covered at all times with water or an inert gas. This requirement reflects a fuel-in-air degradation phenomenon described in Reference 6.53. Except for draining a small amount of water from the MPC for lid welding operations, the HI-STORM FSAR operating procedures (ALARA Warning preceding Step 8.1.5.2.b) and applicable cask loading procedures (References 6.31.3, 6.31.6, 6.56.3, and 6.56.6) require inert gas in the MPC. In this "welding" configuration, water continues to cover the fuel cladding. When the water is completely drained for canister drying, helium is used to assist with blowdown and the canister cavity is never exposed to air during drying. Therefore, this CoC requirement is met.
3.5 Cask Transfer Facility (CTF)	Lifting of a loaded HI-TRAC transfer cask and MPC is not performed outside of HCGS and SGS structures governed by 10 CFR 50. Therefore, no CTF is required and CoC Appendix B, Section 3.5 is not applicable to the Salem/Hope Creek ISFSI.

Table 3, CoC Appendix B — Approved Contents and Design Features

Approved Contents and Design Features	Evaluation
3.6 Forced Helium Dehydration System	This specification establishes requirements for the Forced Helium Dehydration (FHD) System, if used for canister drying instead of vacuum drying. Holtec International has certified that the FHD System used at SGS meets the design criteria requirements in Specification 3.6.2 and has performed the thermal analysis required by Specification 3.6.3, as documented in Reference 6.74.
3.7 Supplemental Cooling System	The Supplemental Cooling System (SCS) is a water circulation system for cooling the MPC inside the HI-TRAC transfer cask during on-site transport. Use of the SCS is required by LCO 3.1.4 for HI-TRAC operation with an MPC containing one or more high burnup (> 45,000 MWD/MTU) fuel assemblies or if the MPC has a total heat load greater than 28.74 kW. Use of the SCS will be dictated by procedure at the time of loading based on the contents and heat load of a given MPC. Holtec International has certified that the SCS used at PSEG meets the design criteria requirements in TS Section 3.7.2 (Reference 6.68). Backup power supplies (TS Section 3.7.2.2) are provided in accordance with procedures (References 6.56.6 and 6.56.8).
3.8 Combustible Gas Monitoring During MPC Lid Welding	During MPC lid welding operations, combustible gas monitoring of the space under the MPC lid is required, to ensure that there is no combustible mixture present in the welding area. This requirement is implemented by procedure (References 6.31.6, 6.56.6, 6.61.1 and 6.61.2).

APPENDIX 2

CASK CoC AND FSAR APPLICABILITY

The table below documents the applicable HI-STORM 100 System Certificate of Compliance (CoC) amendment and Final Safety Analysis Report revision (including approved interim changes[5]) for each serial number Multi-purpose Canister (MPC), HI-STORM overpack, and HI-TRAC transfer cask. This is a living table, updated for each fuel loading campaign.

FUEL LOADING CAMPAIGN	PLANT	COMPONENT MODEL AND SERIAL NUMBER (S/N)	HI-STORM COC AMNDT	HI-STORM FSAR REVISION	APPROVED INTERIM CHANGES*	FUNCTIONAL LOCATION[6] (FLOC)
2006-07 (All casks)	HC	HI-TRAC 100D Transfer Cask Serial No. 1026-4***	2	3	1026-30R0 1026-31R0 1026-32R0 1026-33R0 1026-40R0 1026-41R0	N/A. The HI-TRAC transfer cask is re-used for each cask loaded.
2006-07 (Casks 1-4)	HC	MPC-68 Serial Nos. 1021-143 through –146**	2	3	1021-63R2 1021-67R0	C1DCS-00S5MPC-239 (S/N 145) C1DCS-00S5MPC-240 (S/N 143) C1DCS-00S5MPC-241 (S/N 144) C1DCS-00S5MPC-242 (S/N 146)
2006-07 (Casks 1-4)	HC	HI-STORM 100S-218 Version B Overpack Serial Nos. 1024-189 through -192	2	3	1024-126R0	C1DCS-00S5HI-STORM-239 (S/N 191) C1DCS-00S5HI-STORM-240 (S/N 189) C1DCS-00S5HI-STORM-241 (S/N 190) C1DCS-00S5HI-STORM-242 (S/N 192)

[5] The term "interim changes" is used here to identify approved, permanent changes to the HI-STORM 100 System licensing and/or design basis made by PSEG Nuclear and/or the CoC holder between formal cask FSAR updates submitted pursuant to 10 CFR 72.248. These changes may have been authorized under the provisions of 10 CFR 72.48 if that regulatory process was determined to be applicable, or under another process (e.g., editorial or administrative change, or program controlled under 10 CFR 50.54). This list does not include one-time changes to address manufacturing deviations that do not result in a change to the generic component design.
[6] To determine which MPC is installed in which HI-STORM, match the FLOC numbers, e.g., MPC FLOC 239 is installed in HI-STORM FLOC 239, etc.

FUEL LOADING CAMPAIGN	PLANT	COMPONENT MODEL AND SERIAL NUMBER (S/N)	HI-STORM COC AMNDT	HI-STORM FSAR REVISION	APPROVED INTERIM CHANGES*	FUNCTIONAL LOCATION[6] (FLOC)
2008 (Casks 5-12)	HC	MPC-68 Serial Nos. 1021-147 through –154	3	5	None	C1DCS-00S5MPC-201 (S/N 152) C1DCS-00S5MPC-202 (S/N 148) C1DCS-00S5MPC-203 (S/N 151) C1DCS-00S5MPC-204 (S/N 153) C1DCS-00S5MPC-235 (S/N 149) C1DCS-00S5MPC-236 (S/N 147) C1DCS-00S5MPC-237 (S/N 150) C1DCS-00S5MPC-238 (S/N 154)
2008 (Casks 5-12)	HC	HI-STORM 100S-218 Version B Overpack Serial Nos. 1024-193 through -200	3	5	None	C1DCS-00S5HI-STORM-201 (S/N 193) C1DCS-00S5HI-STORM-202 (S/N 200) C1DCS-00S5HI-STORM-203 (S/N 198) C1DCS-00S5HI-STORM-204 (S/N 194) C1DCS-00S5HI-STORM-235 (S/N 197) C1DCS-00S5HI-STORM-236 (S/N 196 C1DCS-00S5HI-STORM-237 (S/N 199) C1DCS-00S5HI-STORM-238 (S/N 195)
2010 (Casks 13-16)	HC	MPC-68 Serial Nos. 1021-155 through -158	5	7	1021-96R1	C1DCS-00S5MPC-101H1 (S/N 155) C1DCS-00S5MPC-102H1 (S/N 156) C1DCS-00S5MPC-103H1 (S/N 157) C1DCS-00S5MPC-104H1 (S/N 158)
2010 (Casks 13-16)	HC	HI-STORM 100S-218 Version B Overpack Serial Nos. 1024-201 through -204	5	7	None	C1DCS-00S5HI-STORM-101H1 (S/N 201) C1DCS-00S5HI-STORM-102H1 (S/N 202) C1DCS-00S5HI-STORM-103H1 (S/N 203) C1DCS-00S5HI-STORM-104H1 (S/N 204)
2010 (Casks 17-20)	Salem 1	MPC-32 Serial Nos. 1023-93 through -96	5	7	1023-57R0 1023-58R0	C1DCS-00S5MPC-135S1 (S/N 93) C1DCS-00S5MPC-136S1 (S/N 96) C1DCS-00S5MPC-137S1 (S/N 94) C1DCS-00S5MPC-138S1 (S/N 95)
2010 (Casks 17-20)	Salem 1	HI-STORM 100S-218 Version B Overpack Serial Nos. 1024-327 through -330	5	7	None	C1DCS-00S5HI-STORM-135S1 (S/N 327) C1DCS-00S5HI-STORM-136S1 (S/N 328) C1DCS-00S5HI-STORM-137S1 (S/N 329) C1DCS-00S5HI-STORM-138S1 (S/N 330)

FUEL LOADING CAMPAIGN	PLANT	COMPONENT MODEL AND SERIAL NUMBER (S/N)	HI-STORM COC AMNDT	HI-STORM FSAR REVISION	APPROVED INTERIM CHANGES*	FUNCTIONAL LOCATION[6] (FLOC)
2011 (Casks 21-25)	Salem 1	MPC-32 Serial Nos. 1023-128 through -132	5	7	1023-57R0 1023-58R0 1023-59R0 1023-60R0 1023-62R0	C1DCS-00S5MPC-333S1 (S/N 131) C1DCS-00S5MPC-334S1 (S/N 132) C1DCS-00S5MPC-335S1 (S/N 128) C1DCS-00S5MPC-336S1 (S/N 129) C1DCS-00S5MPC-338S1 (S/N 130)
2011 (Casks 21-25)	Salem 1	HI-STORM 100S-218 Version B Overpack Serial Nos. 1024-426 through -430	5	7	None	C1DCS-00S5HI-STORM-333S1 (S/N 426) C1DCS-00S5HI-STORM-334S1 (S/N 427) C1DCS-00S5HI-STORM-335S1 (S/N 428) C1DCS-00S5HI-STORM-336S1 (S/N 429) C1DCS-00S5HI-STORM-338S1 (S/N 430)
2012 (Casks 26-32)	Salem 2	MPC-32 Serial Nos. 1023-133, -209, -210, and -246 through -249	5	7	1023-57R0 1023-58R0 1023-59R0 1023-60R0 1023-62R0 1023-63R0 1023-64R0 1023-65R0	C1DCS-00S5MPC-301S2 (S/N 247) C1DCS-00S5MPC-302S2 (S/N 133) C1DCS-00S5MPC-303S2 (S/N 209) C1DCS-00S5MPC-304S2 (S/N 249) C1DCS-00S5MPC-305S2 (S/N 248) C1DCS-00S5MPC-306S2 (S/N 246) C1DCS-00S5MPC-337S2 (S/N 210)
2012 (Casks 26-32)	Salem 2	HI-STORM 100S-218 Version B Overpack Serial Nos. 1024-431, -567, -568, and -628 through -631	5	7	1024-146R1 1024-151R1	C1DCS-00S5HI-STORM-301S2 (S/N 431) C1DCS-00S5HI-STORM-302S2 (S/N 567) C1DCS-00S5HI-STORM-303S2 (S/N 628) C1DCS-00S5HI-STORM-304S2 (S/N 629) C1DCS-00S5HI-STORM-305S2 (S/N 630) C1DCS-00S5HI-STORM-306S2 (S/N 631) C1DCS-00S5HI-STORM-337S2 (S/N 568)

* Holtec Engineering Change Order (ECO) number. (May not apply to all serials numbers – see Component Completion Records for specific applicability.)

** The Component Completion Records for these serial number MPCs also include ECOs 1021-50, -56, 61, and 62. However, all of these ECOs were incorporated into the revisions of the MPC enclosure vessel and MPC-68 fuel basket licensing drawings included in FSAR Revision 3 (Drawings 3923, Rev. 13 and 3928, Rev. 7).

*** Re-certified by Holtec as compliant with all performance requirements in CoC Amendment 3/FSAR Revision 5 and CoC Amendment 5/FSAR Revision 7 (Reference 6.2, Section 1.0.2).