SAFETY EVALUATION REPORT

DOCKET NO. 72-1014

HOLTEC INTERNATIONAL

HI-STORM 100 CASK SYSTEM

CERTIFICATE OF COMPLIANCE NO. 1014

AMENDMENT NO. 4

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SUMMARY

By letter dated June 23, 2006, Holtec International (Holtec) submitted an application to the United States Nuclear Regulatory Commission (NRC) to amend Certificate of Compliance (CoC) No. 1014 for the HI-STORM 100 Cask System (License Amendment Request (LAR) 1014-5, Revision 0), in accordance with U.S. Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste and Reactor-Related Greater than Class C Waste," Title 10, Part 72 (10 CFR Part 72). By letter dated August 11, 2006, the NRC staff, hereafter referred to as the staff, informed Holtec that the LAR 1014-5 application contained sufficient information for the staff to begin a technical review.

The application (LAR 1014-5, Revision 0) requested changes to the CoC, Technical Specifications (TS), and Final Safety Analysis Report (FSAR) to modify the HI-STORM 100 Cask System. The amendment proposed to add site specific options to the CoC to permit use of a modified HI-STORM 100 Cask System at the Indian Point Unit 1 (IP1) Independent Spent Fuel Storage Installation (ISFSI). These options include the shortening of the HI-STORM 100S Version B, Multi-Purpose Canister (MPC) -32 and MPC-32F and the HI-TRAC 100D to accommodate site specific restrictions. Additional changes proposed address; the Technical Specifications (TS) definition of "TRANSPORT OPERATIONS," and associated language in the Safety Analysis Report; the soluble boron requirements for Array/Class 14x14E IP1 fuel; the helium gas backfill requirements for Array/Class 14x14E IP1 fuel; the addition of a fifth damaged fuel container design under the TS definition for Damaged Fuel Container; addition of separate burnup, cooling time, and decay heat limits for Array/Class 14x14E IP1 fuel for loading in an MPC-32 and MPC-32F; addition of antimony-beryllium secondary sources as approved contents; the loading of all IP1 fuel assemblies in damaged fuel containers; the preclusion of loading of IP1 fuel debris in the MPC-32 or MPC-32F; the reduction of the maximum enrichment for Array/Class 14x14E IP1 fuel from 5.0 to 4.5 wt% ^{235}U; changes to licensing drawings to differentiate the IP1 MPC-32 and MPC-32F from the previously approved MPC-32 and MPC-32F; and other editorial changes including replacing all references to US Tool and Die (UST&D) with Holtec Manufacturing Division (HMD).

In addition to the revision to the CoC, TS, and FSAR, proposed by the applicant to reflect the above changes, changes to the CoC were made, consistent with information described in the FSAR, to clarify the configuration and materials used for the transfer cask, to clarify a distinction regarding overpack variants, and clarify the definition of damaged fuel assemblies.

This Safety Evaluation Report (SER) documents the review and evaluation of the revised FSAR through FSAR Revision 5 submitted June 21, 2007, supplemental materials, and proposed CoC changes. The FSAR follows the format similar to that of the U.S. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems," NUREG-1536, January 1997, (NUREG-1536) with differences implemented for clarity and consistency.

I. REVIEW CRITERIA

The staff's evaluation of each of the proposed changes is based on whether the applicant meets the applicable requirements of 10 CFR Part 72 for independent storage of spent fuel and of 10 CFR Part 20 for radiation protection. The staff's evaluation focused only on modifications requested in the LAR 1014-5 and did not specifically reassess previously approved portions of the CoC, TS, and the FSAR or those areas of the FSAR modified by Holtec as allowed by 10 CFR 72.48. The technical objectives for the following review disciplines are as described below for each of the proposed changes.

The objectives of the structural review is to assess the safety analysis of the structural design features, the structural design criteria, and the structural analysis methodology used to evaluate the expected structural performance capabilities under normal operations, off-normal operations, accident conditions and natural phenomena events for those structures, systems, and components important to safety included in this amendment.

The structural review was conducted in accordance with the regulations set forth in 10 CFR 72.236 that identify the specific requirements for spent fuel storage cask approval and fabrication. The unique characteristics of the spent fuel to be stored are identified as required by 10 CFR 72.236(a) so that the design basis and the design criteria that must be provided for the structures, systems, and components important to safety can be assessed under the requirements of 10 CFR 72.236(b). LAR 1014-5 was also reviewed to determine whether the modifications to the HI-STORM 100 Cask System fulfill the acceptance criteria listed in Section 3 of NUREG-1536.

The objectives of the thermal review are to ensure that the cask component and fuel material temperatures of the HI-STORM Cask System and HI-TRAC Spent Fuel Transfer System will remain within the allowable values or criteria for normal, off-normal, and accident conditions. These objectives include confirmation that the fuel cladding temperature will be maintained below specified limits throughout the storage period to protect the cladding against degradation that could lead to gross ruptures. This portion of the review also confirms that the cask thermal design has been evaluated using acceptable analytical techniques and/or testing methods.
The review was conducted in accordance with the regulations set forth in 10 CFR 72.236 that identify the specific requirements for spent fuel storage cask approval and fabrication. The unique characteristics of the spent fuel to be stored are identified as required by 10 CFR 72.236(a) so that the design basis and the design criteria that must be provided for the structures, systems, and components important to safety can be assessed under the requirements of 10 CFR 72.236(b). This amendment was also reviewed to determine whether the modifications to the HI-STORM 100 Cask System fulfills the acceptance criteria listed in Section 4 and 11 of NUREG-1536 as well as associated Interim Staff Guidance (ISG) documents including, Interim Staff Guidance No. 11, Revision 3, "Cladding Considerations for the Transportation and Storage of Spent Fuel," November 17, 2003.

The objective of the shielding review is to ensure that there is adequate protection to the public and workers against direct radiation from the cask contents. The review ensures that both changes to the shielding features and contents provide adequate protection against direct radiation to the operating staff and members of the public. Further, the review ensures that any direct radiation exposures will be within the regulatory limits for normal operating, off-normal, and design-basis accident conditions. The objective includes review of changes to the shielding design description, radiation source definition, shielding model specification, and shielding analyses for the HI-STORM 100 Cask System proposed by LAR 1014-5.

The regulatory requirements for providing adequate radiation protection to licensee personnel and members of the public include 10 CFR Part 20, 10 CFR 72.104, 10 CFR 72.106(b), 10 CFR 72.212, and 10 CFR 72.236(d). Because 10 CFR Part 72 dose requirements for members of the public include direct radiation, effluent releases, and radiation from other uranium fuel-cycle operations, an overall assessment of compliance with these regulatory limits was evaluated.

The objective of the criticality review is to ensure that the spent fuel will remain subcritical under all credible normal, off-normal, and accident conditions encountered during handling, packaging, transfer, and storage. The objective includes a review of the changes to the criticality design criteria, features and fuel specifications, a verification and review of the configuration and material properties for the HI-STORM 100 Cask System, and a review of the criticality analyses that may include computer programs, benchmark comparisons, and multiplication factors proposed in LAR 1014-5.

The applicant proposed several modifications to the HI-STORM 100 Cask System design. The staff reviewed the proposed changes to the HI-STORM 100 Cask System criticality safety analysis to ensure that all credible normal, off-normal, and accident conditions have been identified and their potential consequences on criticality considered such that the HI-STORM 100 Cask System, as revised, meets the following regulatory requirements: 10 CFR 72.124(a), 72.124(b), 72.236(c), and 72.236(g). The staff's review also involved a determination on whether the cask system fulfills the acceptance criteria listed in Section 6 of NUREG-1536.

The objective of the confinement review is to ensure that radiological releases to the environment will be within the limits established by the regulations and that the spent fuel cladding and fuel assemblies will be sufficiently protected during storage against degradation that otherwise might lead to gross ruptures.

The objective of review of the operating procedures is to ensure that the applicant's FSAR presents acceptable operating sequences, guidance, and generic procedures for key operations. The staff reviewed the proposed changes to ensure the changes in the operating procedures meet the following regulatory requirements: 10 CFR 72.104(b), 72.122(I), 72.212 (b)(9), 72.234(f), and 72.236(h) and (I). LAR 1014-5 was also reviewed to determine whether the cask system fulfills the acceptance criteria listed in Section 8 of NUREG-1536.

The objective of review of the radiation protection program is to ensure that the capability of the current and revised radiation protection design features, design criteria, and the operating procedures, as appropriate, of the HI-STORM 100 Cask System, can meet regulatory dose requirements for the proposed changes. The regulatory requirements for providing adequate radiation protection to site licensee personnel and members of the public include 10 CFR Part 20, 10 CFR 72.104(a), 72.106(b), 72.212(b), and 72.236(d).

Calculated occupational exposures from the HI-STORM 100 system are based on the direct radiation dose rates calculated in Chapter 5 of the FSAR and the operating procedures discussed in Chapter 8 of the FSAR. Calculated doses to individuals beyond the controlled area boundary (members of the public) are determined from the direct radiation (including skyshine) dose rates calculated in Chapter 5 of the FSAR.

The objective of the accident analysis review is to evaluate the applicant's identification and analysis of hazards, as well as the summary analysis of system responses to both off-normal and accident or design-basis events. This ensures that the applicant has conducted thorough accident analyses, as reflected by the following factors:

1) Identified all credible accidents
2) Provided complete information in the FSAR
3) Analyzed the safety performance of the cask system in each review area
4) Fulfilled all applicable regulatory requirements

The proposed modifications to the applicant's description and conclusions regarding the cause of an event, detection of an event, summary of event consequences, regulatory compliance, and corrective course(s) of action were reviewed. The regulatory requirements applicable to accident analysis changes proposed by this amendment include 10 CFR 72.104(a), 10 CFR 72.106(b), 10 CFR 72.122(b)(2), (3), (d), (g), (h)(4), (l), and (l), 10 CFR 72.124(a), 10 CFR 72.236(c), (d), and (l), and 10 CFR 72.212(b). LAR 1014-5 was also reviewed to determine whether the modifications to the HI-STORM 100 Cask System fulfill the acceptance criteria listed in Section 11 of NUREG-1536.

The objective of the review of the TS is to assess the proposed modifications to CoC 1014 "Conditions" and Appendices A and B to the CoC (Technical Specifications (TS)) and determine if the changes are appropriate to accommodate the design modifications requested by the amendment. This review focused on evaluating whether the Conditions and TS had been revised to ensure that all safety limits and regulations were met. The proposed changes to the CoC and TS for the HI-STORM 100 Cask System were reviewed to the above criteria.

II. GENERAL DESCRIPTION OF THE CASK DESIGN

The HI-STORM 100 Cask System is a dry cask storage system for spent light water reactor fuel. The system comprises three discrete components: the multi-purpose canister (MPC), the HI-TRAC Transfer Cask, and the HI-STORM 100 storage overpack.

The MPC is the confinement system for the stored fuel. It is a welded, cylindrical canister with a honeycombed fuel basket, a baseplate, a lid, a closure ring, and the canister shell. All MPC components that may come into contact with spent fuel pool water or the ambient environment, with the exception of neutron absorber, aluminum seals on vent and drain port caps, and optional aluminum heat conduction elements, are constructed of stainless steel. The canister shell, baseplate, lid, vent and drain port cover plates, and closure ring are the main confinement boundary components. The honeycombed basket, which is equipped with neutron absorbers, provides criticality control. There are eight approved MPC designs; MPC-24, MPC-24E, and MPC-24EF which can contain a maximum of 24 pressurized water reactor (PWR) fuel assemblies; the MPC-32 and MPC-32F which can contain a maximum of 32 PWR fuel assemblies; and the MPC-68, MPC-68F, and MPC-68FF which can contain a maximum of 68 boiling water reactor (BWR) fuel assemblies. Vibration suppressors are considered integral non-fuel hardware consisting of zircaloy or stainless steel tubes.

The HI-STORM 100 storage overpack provides shielding and structural protection of the MPC during storage. The overpack is a heavy-walled, steel and concrete, cylindrical vessel. In addition to the HI-STORM 100 storage overpack, there are three additional approved variations including the HI-STORM 100S, HI-STORM 100A, and the HI-STORM 100SA. The HI-STORM 100S is a shorter version of the HI-STORM 100. This amendment proposes to expand the HI-STORM 100 Cask System to include options specific for IP1. The affected components are the MPC enclosure vessel, MPC-32 and MPC-32F, HI-STORM 100S Version B, and HI-TRAC 100D. The HI-STORM 100S Version B and HI-TRAC 100D were implemented by Holtec under the purview of 10 CFR 72.48 and included in Revisions 3 and 4, to the FSAR submitted by Holtec, in accordance with the requirements of 10 CFR 72.248, on April 10, 2006.

The HI-STORM 100S, Version B, and HI-TRAC 100D have been shortened for use at IP1. The variant of HI-STORM 100S, Version B, for use at IP1, called the HI-STORM 100S-185, was shortened by approximately 33 inches to a total height of approximately 185 inches. The other physical characteristics (e.g., inlet and outlet vents, inner and outer shells, and lid) of the HI-STORM 100S, Version B, remain unchanged.

The MPC basket and shell for use at IP1, were also shortened by approximately 33 inches. The neutron absorber panels and sheathing were shortened by approximately 20 inches. The neutron absorber panels in the MPC-32 for IP1 effectively cover the entire height of the basket. The primary features that define the MPC-32 (e.g., cell opening, cell pitch, basket wall thickness, neutron absorber thickness, and B10 loading) are unchanged for use at IP1, and as such the basket will still be called an MPC-32. The MPC-32 for IP1 may be used with both the HI-STORM 100S, Version B, and HI-STORM 100S-185.

The HI-TRAC transfer cask provides shielding and structural protection of the MPC during loading, unloading, and movement of the MPC from the spent fuel pool to the storage overpack. The HI-TRAC was previously reviewed and approved by the staff for the original application. The HI-TRAC 100D was also shortened by approximately 33 inches and the thickness of the outer steel shell was reduced by 1/4 inch and the lead thickness by 3/8 inch. The water jacket thickness, pool lid, and bottom flange remain unchanged. This variant of the HI-TRAC 100D is called the HI-TRAC 100D, Version IP1.

The basic sequence of operations for the HI-STORM 100 Cask System is as follows: (1) the transfer cask, with the MPC inside, is lowered into the spent fuel pool and the MPC is loaded with spent nuclear fuel; (2) the transfer cask and loaded MPC are removed from the spent fuel pool and the MPC is drained, dried, welded closed, inspected, and backfilled with an inert gas; (3) the transfer cask is placed on top of the overpack and the MPC is lowered into the overpack; and (4) if necessary the overpack, with the MPC inside, is moved to the storage pad. A loaded HI-TRAC transfer cask can be handled vertically or horizontally, with the exception of the HI-TRAC 100D, Version IP1, which can only be handled vertically. A loaded HI-STORM 100 Cask System overpack can only be moved vertically. MPC transfer between the transfer cask and overpack can be performed inside or outside a 10 CFR Part 50 controlled structure (e.g., a reactor building).

III. FINDINGS

The proposed changes were reviewed to the criteria and regulations described in Section II of this SER and a discussion of the staff review and findings in each technical discipline are described below.

IIIa. Structural Review

The staff reviewed the HI-STORM 100 Cask Storage System structural design for conformance to 10 CFR Part 72 requirements.

Structural Design Features

The structural design features of all three of the major components of the HI-STORM 100 Cask System remain the same under the amendment except that the height dimension has been reduced to accommodate the shorter fuel assemblies from the Indian Point 1 facility with the resulting decrease in the weight of this specific cask system variant. A reduction in height of the HI-STORM 100S Version B of nominally 33" has been accomplished and the new variant for Indian Point 1 has been identified as the HI-STORM 100S -185. The MPC -32/32F canister and fuel basket have also been shortened by a nominal 33". The MPC changes are identified as DRAFT Revision 15 (undated) to Drawing No. 3923 and for the Fuel Basket Assembly the changes are identified as DRAFT Revision 10 (undated) to Drawing No. 3927. The HI-STORM 100S changes are identified as DRAFT Revision 12 (undated) to Drawing N. 4116. The physical feature changes in the transfer cask, HI-TRAC 100D, also involved a reduction in height by a nominal 33", but also the transfer cask had the outer steel shell thickness reduced by 1/4" and the thickness of the lead shielding layer was reduced by 3/8". This transfer cask is now identified as the HI-TRAC 100D Version IP1 in a new drawing set known as Drawing No. 4724, Revision 0, dated May 11, 2007. The existing FSAR drawing sets for the overpack, the MPC, and the MPC basket as previously identified were modified by merely adding a note relative to the dimensional change in the height. No other features that impact the structural design aspects were modified. It is noted that the MPC -32/32F for IP1 can also be used inside the HI-STORM 100S Version B overpack.

Structural Design Criteria

While not specifically a direct structural criterion, the structural criteria must consider that the stored spent fuel should not sustain damage while in storage since there could be problems with retrievability after the storage period. For the use of the HI-STORM 100S -185, it will be required that each of the fuel assemblies from IP1 be prepackaged in a damaged fuel container (DFC). A new DFC design for use at IP1 is included in the FSAR and adds to the four existing DFC designs using the same materials and design concepts with the dimensional parameters matched with the specifics of the IP1 spent fuel. The top and bottom openings are covered with fine mesh stainless steel screens that will trap most fuel debris if that level of damage were to exist or develop. Since the MPC -32/32F for IP1 has been shortened to be consistent with the length of the fuel assemblies, fuel spacers to prevent gross movement of the fuel will not be necessary.

The design criteria for the normal, off-normal, accident, and extreme environmental loading conditions remain the same as described in the FSAR, Revision 5, with the additional controls on the handling accident. No height limit is imposed when lifting a loaded HI-TRAC 100D Version IP1 in the vertical position with devices designed in accordance with ANSI N14.6 and having redundant drop protections features and when outside the 10 CFR Part 50 licensed facility. Site specific analysis may be performed, as prescribed by CoC Appendix B, Section 5.5.a, and the FSAR, to determine a vertical lift height limit if redundant drop features are not utilized and the lifting devices have not been designed in accordance with ANSI N14.6.

Structural Loading Conditions

For the parameters used in the analyses of the HI-STORM 100S-185 storage cask such as the weights and heights of center of gravity for the different configurations of loading, all are under the bounding values of the standard HI-STORM 100S and the MPC-32. These values are tabulated in Chapter 3.II of the proposed FSAR revisions. For the HI-TRAC 100D Version IP1 new calculations were performed for the trunnions and trunnion blocks even though they are the same as are used on the HI-TRAC 125. The differences are in the structural dimensions of the shell body of the cask where the attachments are made. The calculation methodology was similar to that used previously. The summary of stress calculation results showed there were sufficient margins of safety against failure. New evaluations were performed for the flooding condition on the cask storage system as well as for tornado wind and missile impact. The effects of seismic events and explosions were also addressed. The non-mechanistic tip-over event has been analyzed for the storage cask to demonstrate that there will be no permanent deformation of the cask that would prevent the removal of the MPC after such an event. The HI-TRAC 100D Version IP1 was analyzed for intermediate and large tornado missile strikes and while the results indicate that the water jacket will remain intact following a large tornado missile strike, it has been assumed to have failed when the shielding performance was evaluated.

Structural Evaluation Findings

Based on the NRC staff's review of the information provided in the HI-STORM 100 Cask System amendment request, LAR 1014-5, the staff finds the following:

1. The FSAR amendment adequately describes all changes to structures, systems, and components (SSCs) that are important to safety and provides drawings and text in sufficient detail to allow evaluation of their structural effectiveness.

2. Any modifications to SSCs as a result of the amendment important to safety are described for the HI-STORM 100 Cask System in sufficient detail to enable an evaluation of their structural effectiveness and are designed to accommodate the combined loads under normal, off-normal, accident, and natural phenomena events.

3. The HI-STORM 100 Cask System amendment is designed to allow handling and retrieval of spent nuclear fuel for further processing or disposal. The staff concludes that no accident or natural phenomena events analyzed will result in damage of the system that will prevent retrieval of the stored spent nuclear fuel.

4. The HI-STORM 100 Cask System amendment results in a system whose design and fabrication will maintain the spent nuclear fuel in a subcritical condition under credible

conditions. The configuration of the stored spent fuel is unchanged under the design conditions. Additional criticality evaluations are discussed in Section 6 of this SER.

5. The cask and its systems important to safety modified by the amendment have been evaluated to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

6. The NRC staff concludes that the structural design of the HI-STORM 100 Cask System and the amendment are in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The structural evaluation provides reasonable assurance that the HI-STORM 100 Cask System will enable safe storage of spent nuclear fuel. This finding is based on a review of the amendment request that considered the regulation itself, appropriate regulatory guides, applicable industry codes and standards, accepted practices, and confirmatory analysis.

IIIb. Thermal Review

The staff reviewed the HI-STORM 100 Cask Storage System thermal design for conformance to 10 CFR Part 72 requirements. The most significant changes proposed by the applicant that affect the thermal performance of the HI-STORM 100 Cask System are listed as follows:

1. Include a shorter HI-STORM 100 storage overpack, MPC, and HI-TRAC, for IP1 fuel assemblies.
2. Limit the heat load to a maximum of 8 kW for IP1 fuel loaded in MPC-32/32F.
3. Add separate backfill requirements for an MPC-32 containing Array/Class 14x14E.

The HI-STORM 100S-185 is a shorter version of the generic HI-STORM 100S, Version B, overpack specifically designed to store IP1 spent fuel assemblies. A shorter version of the MPC-32 (MPC-32-IP1) and HI-TRAC 100D Version IP1, are designed for the IP1 spent fuel.

The most relevant assumptions which have a major impact in the thermal results are:

1. No credit is taken for motion of helium inside MPC.
2. The maximum heat load is limited to 8 kW for IP1 spent fuel.

Table 4.II.2 of the FSAR presents the results of the thermal analysis. All MPC and HI-STORM 100 Cask System component temperatures are lower than the allowable limits of Section 2.2 of the FSAR. Based on these results it can be concluded that fuel temperatures, confinement boundary temperatures, surface temperatures, and MPC internal pressure are bounded by the generic HI-STORM 100 Cask System design. All cask materials and fuel cladding are maintained within their temperature limits for normal, off-normal, and accident conditions.

<u>Thermal Evaluation Finding</u>

Based on the review of the license amendment request, the staff finds that the thermal design of the Cask System options proposed for use at IP1 are in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

IIIc. Shielding Review

The staff reviewed the HI-STORM 100 Cask Storage System thermal design for conformance to 10 CFR Part 72 requirements. The most significant changes proposed by the applicant that affect the shielding performance of the HI-STORM 100 Cask System are listed as follows:

1. Shortening of the storage overpack, creating the HI-STORM 100S-185.
2. Shortening of the MPC enclosure vessel, creating an MPC enclosure vessel specific for Indian Point Unit 1 fuel.
3. Modifications of the transfer cask, creating the HI-TRAC 100D, Version IP1, with shortened height, reduced thickness of the outer steel shell (~1/4 in) and lead shielding (~3/8 in), and reduced maximum weight of 75 tons.
4. Loading of the MPC-32/32F with 32 damaged and/or intact IP1 assemblies with, or without, an Antimony-Beryllium source in each assembly; all IP1 assemblies, intact and damaged, to be loaded into damaged fuel cans. Parameters such as maximum allowable burnup, minimum cooling time, and minimum enrichment were modified for the 14x14E assembly class/array (14x14E) for analysis in the MPC-32/32F to reflect the parameters of the IP1 fuel assemblies, which are the only assemblies within the 14x14E assembly class/array.

The staff reviewed these proposed changes to and their effects on the currently approved HI-STORM 100 Cask System. The staff did not review any potential shielding design changes that may have been incorporated under the change authority of 10 CFR 72.48.

For the proposed IP1 fuel assembly contents, the applicant calculated a design-basis source term based upon the bounding specifications with the same method used for previously approved HI-STORM 100 Cask System amendments. These specifications are given in Section 5.II of the amendment. The neutron and gamma sources were calculated for IP1 fuel at a burnup of 30,000 MWD/MTU, a minimum enrichment of 3.5 wt. % Uranium-235 and a minimum cooling time of 30 years. Some IP1 fuel assemblies have a minimum enrichment of 2.7 wt. % Uranium-235; however, the maximum burnup of these assemblies is only 10,000 MWD/MTU, and the minimum cooling time is 30 years. The source from this latter group of assemblies is bounded by the assemblies having the higher minimum enrichment and burnup limit.

The design-basis IP1 gamma source term is listed in Table 5.II.6 of the proposed amendment. IP1 assemblies have a stainless steel shroud around them that is perforated with uniformly spaced holes. The source term in the table includes the contribution from the activated stainless steel cladding and this shroud. The source calculation assumes a Cobalt impurity level of 2.2 g/kg in the steel based upon the IP1 assemblies' vintage and the discussion in SAR Section 5.2.1 of data that indicate that the cobalt impurity in steel in assemblies manufactured in the 1970s has been found to be as high as 2.2 g/kg.

The design-basis IP1 neutron source term is listed in Table 5.II.5 of the proposed amendment. IP1 assemblies also contain Antimony-Beryllium (Sb-Be) secondary neutron sources that replace a single fuel rod in the assembly. The neutron source term in Table 5.II.5 does not include this neutron source. The applicant performed analyses that indicate that the neutron source strength of the Sb-Be source is similar to that of a single rod in an IP1 assembly. Thus, the applicant concluded that explicit consideration of the source in the dose analyses is not necessary and modeled all rod locations as fuel rods in the analyses. Staff notes that the activated Antimony (Sb-124) in IP1's Sb-Be also contributes to the assembly's gamma source. The applicant's analyses, however, indicate that the regeneration of the Sb-124 is negligible; therefore, the contribution to the gamma source as well as to neutron generation in the Beryllium from the Sb-124 is also negligible.

The staff reviewed the applicant's analysis of the design-basis IP1 source term. The staff has reasonable assurance that the design-basis IP1 gamma and neutron source terms for the HI-STORM 100 Cask System shielding analyses are acceptable. Limits for fuel parameters such as burnup, cooling time and decay heat for the IP1 assemblies are incorporated into Appendix B of the CoC as limits for the 14x14E fuel assembly class/array.

Since the only proposed change to the overpack is a reduction of the height and the source term for the IP1 contents is bounded by the currently approved design-basis contents, the applicant did not perform any dose rate calculations for the overpack loaded with IP1 contents. The staff notes, however, that the proposed overpack and MPC heights result in the top of the MPC being at the closest position relative to the overpack outlet vents of any of the configurations analyzed for an MPC in a HI-STORM overpack. The staff finds that the dose rates from the proposed overpack and MPC configuration will be bounded by the dose rates from the current MPC and overpack configurations based upon the proposed IP1 contents, which have a significantly lower source strength (both total source strength and source strength per energy group) than the currently approved design-basis contents. However, the site user should consider the proximity of the MPC to the overpack outlet vents to ensure implementation of appropriate ALARA precautions for operations near these overpack areas.

The proposed changes to the transfer cask result in a reduction in shielding capability versus the current transfer casks. Therefore, the applicant performed dose rate calculations for the transfer cask loaded with IP1 fuel under normal and accident conditions, presenting dose rates adjacent to and at one meter from the cask under normal conditions and at one meter from the cask under accident conditions. The analysis was performed with the same material properties and similar configurations to those used for previously approved HI-STORM 100 Cask System amendments; the configurations were modified to account for differences in the proposed transfer cask's radial shielding. Section 5.II.3 of the amendment describes the shielding and source configuration used in the IP1 analysis.

The applicant proposed to include damaged IP1 fuel as allowed contents with the ability to load damaged fuel in any fuel basket location. The IP1 fuel is only to be loaded in the MPC-32 and/or MPC-32F baskets; thus, the proposal is only for allowing damaged IP1 fuel to be stored in any location in these two MPCs. Initially, the applicant's analysis did not address the dose rates from a cask filled with damaged fuel assemblies under accident conditions. Instead, the analysis assumed all the assemblies remain intact. This assumption was based upon inspections of the fuel showing no damage to the assembly. However, the lack of sufficient records and the inability to perform full inspections of the assemblies to demonstrate an intact fuel condition result in the inability to classify IP1 fuel assemblies as intact. Yet, the applicant states that due to the assembly design having a shroud and cladding of stainless steel, the assemblies are much less likely to sustain damage under accident conditions than standard PWR assemblies. The applicant also referred to the damaged fuel analyses done in SAR Section 5.4.2.2 to show that the impact on dose rates from damaged fuel (versus intact fuel) is small.

In its review, the staff found that the bases for the analysis assumption of intact fuel were not sufficient. First, the inspections appear to be visual only, by which only a small fraction of the assembly surface area can be evaluated. Thus, there is a large percentage of the assembly that has not been inspected that may be damaged. Second, while the assembly shroud may aid in reducing the damage an assembly may sustain under accident conditions, the argument should contain some quantitative support and appropriate justification as to what defines the worst case reconfiguration of the fuel. Third, the analyses in SAR Section 5.4.2.2 are for casks that limit the allowable storage locations to, at most, the outermost ring of basket storage locations (MPC-68). Although, the contribution from assemblies decreases as the location moves toward the cask center, the contribution of assemblies in locations near the outer basket locations is not insignificant. In addition to the dose rates at the cask midplane, the dose rates around the lower part and upper part of the cask are of interest due to redistribution of the source, resulting from any reconfiguration of the assemblies, such that there are higher dose rates in areas where personnel are likely to be (particularly near the lower portion of the cask).

Due to the foregoing staff finding, the applicant modified its analysis to include dose rates from a transfer cask filled with damaged fuel that has reconfigured. The analysis technique is consistent with the method used in the generic damaged assembly analysis in SAR Section 5.4.2.2. The results show increases in the radial dose rates at all locations (versus dose rates from intact assemblies), with the greatest increase near the lower areas of the cask.

A comparison of dose rates shows that the IP1 transfer cask dose rates are bounded by the dose rates from the generic HI-TRAC transfer casks containing design basis fuel. Based upon the foregoing, it is determined that the off-site dose rates and the occupational exposures from the IP1 version of the HI-STORM 100 Cask System are bounded by those calculated for the generic system. Therefore, the dose rates and occupational exposures estimated for the generic HI-STORM 100 Cask System are used to demonstrate compliance with 10 CFR Parts 20 and 72.

The staff has reviewed the applicant's analysis, as modified, of the normal conditions and accident conditions dose rates and finds the analysis method to be acceptable for the proposed IP1 version of the HI-STORM cask system and contents. The staff also finds that the generic system bounds the IP1 version of the cask system. Therefore, the staff finds there is reasonable assurance that compliance with 10 CFR Parts 20 and 72 can be achieved by the general licensee for the proposed IP1 version of the cask system and contents. The staff notes that actual doses to individuals beyond the controlled area boundary will depend upon several site specific conditions such as cask-array configurations, topography, demographics, and distances. In this regard, 10 CFR 72.104(a) requires that radiation exposure, from all uranium fuel cycle operations in the region, to any individual located beyond the ISFSI controlled area does not exceed certain specified doses. Pursuant to 10 CFR 72.212, general licensees are required to perform written evaluations, prior to cask use, that establish that the dose requirements of 10 CFR 72.104 have been met. In addition, the general licensee will also have an established radiation protection program, as required by 10 CFR Part 20, Subpart B, and will demonstrate compliance with dose limits to individual members of the public and workers, as required, by evaluation and measurements. The general licensee should implement appropriate ALARA precautions to minimize personnel doses. Additionally, TS 5.7 establishes a radiation protection program for the HI-STORM 100 Cask System. The limits for 14x14E fuel assembly array/class (IP1) decay heat, burnup, cooling time, enrichment, and other characteristics are specified in Appendix B of the CoC. The NRC may inspect the site-specific use of the HI-STORM 100 Cask System for compliance with radiological protection requirements.

Shielding Evaluation Finding

The staff finds that the design of the radiation protection system of the HI-STORM 100 Cask System can be operated in compliance with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the HI-STORM 100 Cask System will provide safe storage of spent fuel. This finding is based on a review that considered the regulation itself, the appropriate regulatory guides, applicable codes and standards, the applicant's analyses, and acceptable engineering practices.

IIId. Criticality Review

The staff reviewed the HI-STORM 100 Cask System criticality safety design for conformance to 10 CFR Part 72 requirements. The applicant desires to store Indian Point Unit 1 fuel in a shorter version of the generic HI-STORM 100S, Version B, overpack specifically designed to store IP-1 spent fuel assemblies (i.e., HI-STORM 100S-185). In addition, a shorter version of the MPC-32 (MPC-32-IP1) and HI-TRAC 100D, Version IP1, are designed specifically for the IP-1 spent fuel.

The most significant changes proposed by the applicant for this license amendment request that affect the criticality safety performance of the HI-STORM 100 Cask System are listed as follows:

1. Elimination of the soluble boron requirement for IP-1 fuel.
2. Addition of a new damaged fuel container.
3. Requirement that all IP-1 fuel assemblies are stored in up to 32 damaged fuel containers.
4. Reduction of the maximum allowed enrichment for Array/Class 14x14E fuel to 4.5 wt% ^{235}U.

The elimination of the soluble boron requirement for IP-1 fuel is operationally necessary since the spent fuel pool at IP-1 does not normally contain any soluble boron. The applicant modified their calculations for the HI-STAR overpack under fully flooded conditions to reflect 0 ppm ^{10}B in the water, assumed that potentially damaged IP-1 fuel was loaded into DFCs, and reduced the maximum enrichment to 4.5 wt% ^{235}U. These results are summarized in Table 6.II.1 of the application and bound the HI-STORM and HI-TRAC storage conditions. The applicant conservatively assumed an extended active fuel length of 150 inches versus the actual length of 102 inches, assumed that all assemblies are centered in each cell, as well as an eccentric condition where all assemblies are shifted toward the center of the basket. The outer spacers were modeled as reduced in size by 0.125 inches to conservatively place the fuel assemblies closer to each other in the calculations, and the steel channel around the rods is not modeled to limit neutron absorption in the calculations. Using all of these very conservative conditions, the applicant performed several studies on the modified class 14x14E fuel in the MPC-32 using various rod array loadings in DFCs to determine the bounding case. In all instances the bounding analyses demonstrate that the maximum k_{eff} of the cask system remains well below the regulatory limit of 0.95 including all biases and uncertainties for all credible conditions.

Criticality Evaluation Finding

The staff finds, based on verification of adequate system modeling by the applicant and that the acceptance standard of a maximum k_{eff} of 0.95 was maintained for all analyzed scenarios, the analyses supporting the changes to the criticality safety analyses identified above, allowing the storage of IP-1 fuel in the HI-STORM 100S-185, were considered acceptable. The staff finds that this change is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been met.

IIIe. Confinement Review

No changes were proposed that affect the overall confinement design and performance of the HI-STORM 100 Cask System to be used at IP1. The applicant indicated that the discussion of the confinement system and the supporting technical basis for the HI-STORM 100 Cask System provided in the FSAR remain fully applicable for the IP1 specific options of the HI-STORM 100 Cask System. The staff has reviewed the confinement system and finds that the assertions and conclusions reached by the applicant are appropriate for the modifications made to the HI-STORM 100 Cask System

IIIf. Operational Procedures Review

Staff finds there were no changes to the operational procedures that affect the cask system shielding and radiation protection designs. As discussed in Section III.g of this SER, the applicant relies upon analyses for generic HI-STORM 100 casks loaded with design-basis contents to provide occupational and off-site dose estimates for cask system operations. The staff notes that the general licensee is responsible for demonstrating site-specific compliance with the limits and requirements of 10 CFR Parts 20 and 72, as required by 10 CFR Part 20 and 10 CFR 72.212. This site-specific demonstration should account for actual operations sequences used at the site, such as the transport of a loaded MPC in the transfer cask between the Unit 1 and Unit 2 fuel buildings at Indian Point, as explained by the applicant in its basis for changing the definition of TRANSPORT OPERATIONS in Appendices A and B and the modification to TS 5.5.a.3 in Appendix A to the CoC.

IIIg. Radiation Protection Review

No changes were proposed for the radiation protection design criteria or for the ALARA objectives, procedures, practices, and policies that were approved previously for the HI-STORM 100 Cask System. Also, the applicant continues to rely upon the analyses performed for the generic HI-STORM 100 casks loaded with design-basis contents to provide estimates of occupational doses and off-site doses under normal, off-normal, and design-basis accident and natural phenomenon event conditions. The staff notes that, as required by 10 CFR Part 20 and 10 CFR 72.212, the general licensee is responsible for demonstrating site-specific compliance with the limits and requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106, which are the radiological protection design criteria for the HI-STORM 100 Cask System. In addition, TS 5.7 establishes direct radiation dose rate limits and other radiation protection criteria for the cask system.

Based upon the analysis provided in this amendment, the actual occupational dose rates for the proposed IP1 version of the cask system loaded with design-basis IP1 contents are expected to be much less than the estimates given in Section 10 of the FSAR. Additionally, the general licensee will have an established radiation protection program, as required in 10 CFR Part 20, Subpart B and will demonstrate compliance with occupational dose limits in 10 CFR Part 20, Subpart C and other site-specific 10 CFR Part 50 license requirements by means of evaluations and measurements. The actual doses to an individual beyond the controlled area boundary depend on several site-specific conditions such as cask-array configurations, topography, demographics, distances, and use of engineered features (e.g., berm). In this regard, 10 CFR 72.104(a) requires that radiation exposure, from all uranium fuel cycle operations in the region, to any individual located beyond the ISFSI controlled area does not exceed certain specified doses. Pursuant to 10 CFR 72.212, general licensees are required to perform written evaluations, prior to cask use, that establish that the dose requirements of 10 CFR 72.104 have been met. The NRC may inspect the site-specific use of the HI-STORM 100 Cask System for compliance with radiological protection requirements.

The staff has reasonable assurance that compliance with 10 CFR 72.104(a) can be achieved by the general licensee. The staff also has reasonable assurance that the effects of direct radiation from bounding design-basis accidents and natural phenomena will be below the regulatory limits in 10 CFR 72.106(b).

Radiation Protection Evaluation Findings

The staff finds that the design of the radiation protection system of the HI-STORM 100 Cask System is in compliance with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the HI-STORM 100 Cask System, as proposed in this amendment, will provide safe storage of spent fuel. This finding is based on a review that considered the regulation itself, the appropriate regulatory guides, applicable codes and standards, the applicant's analyses, and acceptable engineering practices.

IIIh. Accident Analysis Review

The staff reviewed the consequences of postulated off-normal events with respect to 10 CFR 72.104(a) dose limits, and finds them acceptable. The radiation consequences from off-normal events are essentially the same as for normal conditions of operation for the proposed contents and design. The staff has reasonable assurance that the dose to any individual beyond the controlled area will not exceed the limits in 10 CFR 72.104(a) during off-normal conditions (anticipated occurrences). Sections III.c and III.g of this SER further examine the radiological doses applicable to off-normal events, as appropriate.

The staff reviewed the design-basis accident analyses with respect to 10 CFR 72.106(b) dose limits and finds them acceptable. The staff has reasonable assurance that the dose to any individual at or beyond the controlled area boundary of 100 meters will not exceed the limits in 10 CFR 72.106(b) for the proposed contents and design. Sections III.c and III.g of this SER further examine the estimated radiological doses during accident conditions.

IIIi. Technical Specifications Review

The staff reviewed the proposed CoC and TS changes. These changes include the addition of the IP1 transfer cask description in the CoC, which was added because the differences in weight, height, and radial shielding make this a transfer cask that is significantly different from those already described in the CoC. Furthermore, this transfer cask is limited to use at Indian Point

Unit 1. Changes were also made to Appendix B Tables 2.1-1 and 2.1-2 regarding the specifications for approved contents. The staff finds that these changes are appropriate for the modifications made to the HI-STORM 100 Cask System.

IV. CONCLUSIONS

The staff has reviewed the proposed changes to the Final Safety Analysis Report (FSAR) for the HI-STORM 100 Cask System. Based on the statements and representations contained in the FSAR as amended, and the conditions given in the CoC as amended, the staff concludes that the HI-STORM 100 Cask System, specifically the HI-STORM 100S-185, MPC-32-IP1, and HI-TRAC 100D Version IP1, meets the requirements of 10 CFR Part 72.

The staff determined that, unless otherwise noted in this SER, all analytical methods used by the applicant, which provide the basis for design modifications and the addition to the list of approved cask contents for the HI-STORM 100 Cask System proposed in LAR 1014-5, are acceptable. However, for the purposes of the LAR 1014-5 review, the staff did not revisit any previously approved methodologies used in the original HI-STORM 100 Cask System application or those reviewed for LAR 1014-1, LAR 1014-2, and LAR 1014-4, or proposed in LAR1014-3, and did not make any new determination on the adequacy of those methodologies, unless the methodology was used as the basis for a proposed LAR 1014-5 change.

Issued with Certificate of Compliance No. 1014,
on January 8, 2008.