[7590-01-P]

NUCLEAR REGULATORY COMMISSION

10 CFR Part 72

RIN 3150-AG 31

List of Approved Spent Fuel Storage Casks: Holtec HI-STORM 100 Addition

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is amending its regulations to add the Holtec HI-STORM 100 cask system to the list of approved spent fuel storage casks. This amendment allows the holders of power reactor operating licenses to store spent fuel in this approved cask system under a general license.

EFFECTIVE DATE: This final rule is effective on (**insert date 30 days from the date of publication in the Federal Register**).

FOR FURTHER INFORMATION CONTACT: Merri Horn, telephone (301) 415-8126, e-mail mlh1@nrc.gov of the Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

SUPPLEMENTARY INFORMATION:

Background

Section 218(a) of the Nuclear Waste Policy Act of 1982, as amended (NWPA), requires that "[t]he Secretary [of Energy] shall establish a demonstration program, in cooperation with the private sector, for the dry storage of spent nuclear fuel at civilian nuclear reactor power sites, with the objective of establishing one or more technologies that the [Nuclear Regulatory] Commission may, by rule, approve for use at the sites of civilian nuclear power reactors without, to the maximum extent practicable, the need for additional site-specific approvals by the Commission." Section 133 of the NWPA states, in part, "[t]he Commission shall, by rule, establish procedures for the licensing of any technology approved by the Commission under Section 218(a) for use at the site of any civilian nuclear power reactor."

To implement this mandate, the NRC approved dry storage of spent nuclear fuel in NRC-approved casks under a general license, publishing a final rule in 10 CFR Part 72 entitled, "General License for Storage of Spent Fuel at Power Reactor Sites" (55 FR 29181; July 18, 1990). This rule also established a new Subpart L within 10 CFR Part 72 entitled, "Approval of Spent Fuel Storage Casks," containing procedures and criteria for obtaining NRC approval of dry storage cask designs.

Discussion

This rule will add the Holtec HI-STORM 100 cask system to the list of NRC approved casks for spent fuel storage in 10 CFR 72.214. Following the procedures specified in 10 CFR 72.230 of Subpart L, Holtec International submitted an application for NRC approval with the Safety Analysis Report (SAR) entitled "Topical Safety Analysis Report for the HI-STORM 100 Cask System." The NRC evaluated the Holtec International submittal and issued a preliminary Safety Evaluation Report (SER) and a proposed Certificate of Compliance (CoC) for the Holtec HISTORM 100 cask system. The NRC published a proposed rule in the Federal Register (64 FR 51271; September 22, 1999) to add the Holtec HI-STORM 100 cask system to the listing in 10 CFR 72.214. The comment period ended on December 6, 1999. Four comment letters were received on the proposed rule.

Based on NRC review and analysis of public comments, the NRC staff has modified, as appropriate, its proposed CoC, including its appendices, the Technical Specifications (TSs), and the Approved Contents and Design Features, for the Holtec HI-STORM 100 cask system. The NRC staff has also modified its preliminary SER. Finally, comments were received from other industry organizations suggesting changes to the TSs and the Approved Contents and Design Features. Some of these were editorial in nature, others provided clarification and consistency, and some reflected final refinements in the cask design. The NRC staff agrees with many of these suggested changes and has incorporated them into the final documents, as appropriate. The NRC staff has also modified the rule language by changing the word "Certification" to "Certificate" to clarify that it is actually the Certificate that expires.

The NRC finds that the Holtec International HI-STORM 100 cask system, as designed and when fabricated and used in accordance with the conditions specified in its CoC, meets the requirements of 10 CFR Part 72. Thus, use of the Holtec HI-STORM 100 cask system, as approved by the NRC, will provide adequate protection of public health and safety and the environment. With this final rule, the NRC is approving the use of the Holtec HI-STORM 100 cask system under the general license in 10 CFR Part 72, Subpart K, by holders of power reactor operating licenses under 10 CFR Part 50. Simultaneously, the NRC is issuing a final SER and CoC that will be effective on **[insert date the rule is effective]**. Single copies of the CoC and SER are available for public inspection and/or copying for a fee at the NRC Public Document Room, 2120 L Street, NW. (Lower Level), Washington, DC.

Summary of Public Comments on the Proposed Rule

The NRC received four comment letters on the proposed rule. The commenters included a industry users group, two members of the public, and a State. Copies of the public comments are available for review in the NRC Public Document Room, 2120 L Street, NW (Lower Level), Washington, DC 20003-1527.

Comments on the Holtec HI-STORM 100 Cask System

The comments and responses have been grouped into eleven areas: general, radiation protection, accident analysis, criticality, design, welds, structural, materials, thermal, technical specifications, and miscellaneous. Several of the commenters provided specific comments on the draft CoC, the NRC staff's preliminary SER, the TSs, and the applicant's SAR. Some of the editorial comments have been grouped. To the extent possible, all of the comments on a particular subject are grouped together. The listing of the Holtec HI-STORM 100 cask system within 10 CFR 72.214, "List of approved spent fuel storage casks" has not been changed as a result of the public comments. A review of the comments and the NRC staff's responses follow.

A. General

Comment A.1: One commenter expressed concern over the number of cask designs being certified because there would be more problems and a lack of standardization and integration in the country's total waste system. The commenter stated that this amendment would change existing environmental concerns as it would add one more design, complicating the waste system for workers at a plant. The commenter asked how many designs would be certified by the NRC and how many designs could be used at one plant. Additional designs add to more mistakes and human error because each design has different fabrication criteria and handling procedures.

Response: These comments are beyond the scope of this rule that is focused solely on whether to add a particular cask design, the Holtec HI-STORM 100 cask system, to the list of approved casks. Pursuant to the general license, each licensee must determine whether or not the reactor site parameters are encompassed by the cask design bases considered in the cask SAR and SER. Further, each general licensee must document this determination in accordance with 10 CFR 72.212.

Comment A.2: One commenter stated that the tiered environmental impact statement (EIS) is outdated for current dry cask design and should be redone, particularly looking at terrorism and sabotage at an independent spent fuel storage installation (ISFSI).

Response: The NRC disagrees with the comment. The environmental assessment (EA) and finding of no significant impact (FONSI) prepared as required by 10 CFR Part 51 conform to National Environmental Policy Act (NEPA) procedural requirements. Tiering on past EISs and EAs is a standard process under NEPA. As stated in the Council on Environmental Quality's 40 Frequently Asked Questions, the tiering process makes each EIS/EA of greater use and meaning to the public as the plan or program develops without duplication of the analysis prepared for the previous impact statement.

The NRC reviewed potential issues related to possible radiological sabotage of storage casks at reactor site ISFSIs in the 1990 rule that added Subparts K and L to 10 CFR Part 72 (55 FR 29181; July 18, 1990). The NRC still finds the results of the 1990 rule current and acceptable. In addition, each Part 72 licensee is required by 10 CFR 73.51 or 73.55 to develop a physical protection plan for the ISFSI. The licensee is also required to install systems that provide high assurance against unauthorized activities that could constitute an unreasonable risk to the public health and safety.

Comment A.3: One commenter questioned whether the NRC was including interim storage away from reactors in the EA, such as at a Federal or private storage site in Nevada or Utah. The commenter further questioned whether it was the NRC's intent to include transfer and storage at a second site in the EA. The commenter asked if the certification covered use at an interim site in Nevada or Utah.

Response: The EA supports the generic use of the Holtec HI-STORM 100 cask system under a general license. The storage could occur at any site that meets the definition of a general licensee under 10 CFR Part 72. The general licensee must evaluate the site to determine whether or not the chosen site parameters are enveloped by the design bases of the approved cask as required by 10 CFR 72.212(b)(3). The EA does not cover transportation from one site to another.

Comment A.4: One commenter questioned whether the NRC claims to have done research on the condition of spent fuel after 20 to 50 years of storage at a reactor in pools and dry casks, after being unloaded twice and being transported across the country. The commenter stated that a detailed analysis of what can happen to spent fuel before it gets to Nevada or Utah should be conducted by the NRC. The commenter asked what the spent fuel will be like and what the potential environmental impacts will be after the fuel is unloaded and transported.

Response: The NRC staff has reviewed numerous research reports regarding the long term condition of spent fuel in wet and dry storage. Additionally, the NRC has ongoing confirmatory research with spent fuel removed from dry storage after 10 to 20 years. Analysis of spent fuel has included the loads from routine shipping; and the effects, primarily due to vibration, were found to be negligible.

The HI-STORM 100 MPC is a dual-purpose canister. Once loaded in the MPC, the fuel is not intended to be unloaded and reloaded as the questioner suggests. The lid welding and testing requirements and the structural and thermal analyses in the SAR give the NRC staff reasonable assurance that cask confinement and fuel integrity will be maintained under design basis normal, off-normal, or accident events. Therefore, fuel unloading should not be necessary. Regardless of whether unloading may be necessary, each cask user is required to develop detailed site-specific unloading procedures. Proper unloading does not cause any particular degradation to occur to the fuel.

Comment A.5: One commenter stated that the no action alternative was acceptable because the NRC should not be certifying numerous designs. The commenter stated that other agencies such as NWTRB, EPA, OCRWM, and DOE should be contacted for their views on what happens to the whole waste system as more designs are certified.

Response: The NRC disagrees with the comment. The NRC found no inherent design features that would result in significant environmental impacts and that the HI-STORM 100 design meets regulatory requirements. Therefore, there is no basis for denial of the application. The NRC does not limit the number or types of casks that may be certified. The NRC is not required to contact the agencies mentioned by the commenter and we have not specifically solicited their input. The commenter may contact these other agencies if interested in their views.

Comment A.6: One commenter recommended finding a reference (reference 1 on page 3-16 of the SER) that is more recent than 1962.

Response: The NRC disagrees with this comment. This reference refers to the change of the coefficient of friction from static to dynamic condition. The rational behind this engineering principle has not changed with time.

Comment A.7: One commenter stated that the NRC should request simpler designs because of material interactions instead of approving designs with new materials that have never received long term testing for material interactions.

Response: The NRC staff disagrees with this comment. The materials used in casks are selected upon the basis of the needed properties. Casks are constructed from a limited number of materials. The materials used in the Holtec HI-STORM design have a long history of use in the nuclear industry and the performance of those materials is well known.

Comment A.8: One commenter objects to site specific changes that are made to generic designs.

Response: This comment is beyond the scope of this rule that is focused solely on whether to place the HI-STORM 100 cask system on the list of approved casks. Section 72.48 permits changes to the spent fuel storage cask as described in the FSAR and defines the conditions under which these changes may be made without prior NRC approval.

Comment A.9: One commenter stated that it appeared that Holtec split what appears to be one generic system into two separate rules and asked why the system was not certified together. Systems should be complete when they are proposed for rulemaking. The commenter further stated that vendors should apply for storage and transport at the same time and that NRC should not allow loading until the transportation portion is certified.

Response: The NRC disagrees with the comment. The HI-STAR 100 Cask System and HI-STORM 100 Cask System are two separate spent fuel storage cask systems. Each is a complete spent fuel storage cask system that satisfies the requirements of 10 CFR Part 72. Regarding the dual-purpose (storage and transportation) use of a cask system or its components, separate certifications are required for approval of a cask design (or individual components such as a canister) under the provisions of use for 10 CFR Parts 71 and 72. There is no regulatory requirement that the certification be simultaneous.

Comment A.10: One commenter asked a number of site-specific questions related to Private Fuel Storage's plans to use the Holtec HI-STAR and HI-STORM cask systems at the Utah site. These issues related to cask handling, dry transfer, sabotage scenarios, infrastructure for unloading, etc. One commenter stated that they understood that Private Fuel Storage plans to use the HI-STAR system for storage and transport with the HI-STORM as a companion concrete overpack, that the metal HI-STAR overpack would be used as a backup, and that the commenter objected to these plans.

Response: The comment is beyond the scope of this rule that is focused solely on whether to add a particular design, the Holtec HI-STORM 100 cask system, to the list of approved casks. The rule will enable licensees to use this cask system under the general license provisions of 10 CFR Part 72. The rule does not address site-specific issues related to potential users.

Comment A.11: One commenter objected to calling the cask a multi-purpose cask (MPC) because that stands for storage, transport, and disposal, and stated that the cask is not approved for these functions which can cause confusion when real MPCs are certified.

Response: The NRC disagrees with the comment. The name or model number given to the cask design is developed by the applicant. The CoC for the Holtec HI-STORM 100 is intended for the interim storage of spent fuel. The use of MPC in a dry storage cask application or an NRC SER/CoC is not a certification under 10 CFR Part 71 for the transport of radioactive materials or an approval for disposal at a high-level waste repository.

Comment A.12: One commenter stated that Holtec should not be allowed to approve its own suppliers and that the suppliers should be ASME-approved.

Response: The NRC disagrees with the comment. NRC regulations do not require an ASME stamp for a cask or the use of ASME-approved suppliers. The design and fabrication requirements for a certified dry cask storage system are described in 10 CFR Part 72 and the NRC staff's Standard Review Plan, NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems" (SRP). Applicant submittals are reviewed to the criteria in the SRP. Cask fabrication activities are audited by the licensees and inspected by the NRC staff to ensure that components are fabricated as designed. The CoC holder and licensee are responsible for verifying that fabricators are qualified. The CoC holder and licensee must have a Quality Assurance (QA) Program that has been approved by the NRC as part of the licensing or CoC issue process. This QA program must meet the requirements of 10 CFR 72.148 and 10 CFR 72.154 for the selection of fabricators. Also, the procurement documents issued to the fabricator must comply with 10 CFR 21.31. The licensee/CoC holder is required to verify that all regulations and CoC conditions applicable to the container are met. The NRC inspects the licensee/CoC holders and fabricators to verify compliance. Additionally, many storage cask fabricators are certified by the American Society of Mechanical Engineers and are N-Stamp Certificate holders.

Comment A.13: One commenter stated that issues should not be resolved in telephone conferences but in public meetings with a record in the public document room.

Response: The NRC disagrees with the comment. Telephone conferences are an important mode of communication with applicants and licensees and enable the NRC staff to conduct its official business efficiently. If, in these telephone conferences, the NRC staff receives information that would form the basis for its regulatory decision, that information is documented and made available for public inspection under 10 CFR Parts 2 and 9.

Comment A.14: One commenter stated that all details of the design should be finalized and open for public comment.

Response: The NRC disagrees that all design details need to be finalized and open for public comment before a design is approved. The NRC staff focuses its review on those design details that are significant with respect to the health and safety of the public and/or are required to make a regulatory finding. Design details that are pertinent to the NRC staff's findings are finalized and made available for public inspection and comment under 10 CFR Parts 2 and 9.

B. Radiation Protection

Comment B.1: One commenter objected to the use of less shielding for the 100-ton transfer cask and allowing the utilities to perform a cost-benefit analysis to justify the use of the 100-ton transfer cask at the expense of the worker. The workers should receive the minimum achievable dose and not the maximum allowable dose. The NRC should not allow the use of the 100-ton transfer cask because the dose is 3 times higher and workers should not be treated as guinea pigs. The commenter stated that the utilities should be required to use the 125-ton transfer cask which is safer and modify their facilities to accommodate the transfer cask or choose a cask that works for their specific site limitations because the utilities shouldn't limit the shielding for workers.

Response: NRC disagrees with this comment. Each cask user will operate the HI-STORM 100 under a 10 CFR Part 20 radiological protection program. ALARA means making every reasonable effort to maintain exposures to radiation as far below the dose limits while taking in account the state of technology, the economics of improvements in relation to the state of technology, and the economics of improvements in relation to benefits to the public health and safety. As stated in Section 2.0.3 of the SAR, the general licensee should utilize the 125-ton transfer cask provided it is capable of using it. However, licensees not capable of using the more shielded design may employ ALARA considerations when evaluating whether to modify its plant or use the 100-ton transfer cask. The NRC found this acceptable as discussed in Section 10.2 of the SER.

Comment B.2: One commenter asked why the specific dose rate criteria for the HI-TRAC was not given and indicated that the criteria should be included.

Response: The applicant did not provide explicit dose rate values as design criteria for the transfer cask designs, but stated that the radiological requirements of 10 CFR Parts 72 and 20 as the overall shielding design objectives for the cask system. The NRC found this acceptable. The TSs in Appendix A of the CoC specify dose rate limits for the transfer casks that are based on the applicant's shielding calculations.

Comment B.3: One commenter questioned the bounding analysis for cobalt impurities, asked how much cobalt is really in the fuel, and if the quantity had been tested and verified for the real thing.

Response: The applicant's analysis of cobalt impurities is discussed in Section 5.2.1 of the SER. The applicant showed that the cobalt impurity values that are assumed in its shielding analyses were appropriate based on industry data and analysis of post-irradiation cooling of older fuel. The NRC found this acceptable. The cask user is not required to measure the actual quantity of cobalt in its spent fuel. The cask user will operate the cask under a 10 CFR Part 20 radiological protection program and verify that the cask system meets the dose rate limits specified in the TSs.

Comment B.4: One commenter asked why backscattering was not considered for all cask designs.

Response: This comment is beyond the scope of this rule that is focused solely on whether to add a particular cask design, the Holtec HI-STORM 100 cask system, to the list of approved casks. Note that backscatter was considered for the Holtec HI-STORM 100 cask system.

Comment B.5: One commenter asked what are the various array configurations allowed and what are the differences between them. The commenter asked if the cask array is limited to two rows and for the applicable NRC criteria.

Response: The use of the HI-STORM design is not limited to two rows. The NRC requires the applicant to perform off-site dose calculations from a typical ISFSI array to demonstrate that radiation shielding features are sufficient to meet the radiological requirements of 10 CFR Parts 72.104 and 72.106. As discussed in Section 5.3.1 of the SER, the applicant used a two-row cask array model as part of its methodology to estimate off-site dose rates. The values obtained by this method can be applied to dose rate calculations for typical cask arrays that may consist of multiple rows. NRC found the dose estimates to be acceptable. Each general licensee will identify an ISFSI configuration and perform a site-specific dose evaluation to demonstrate compliance with Part 72 radiological requirements.

Comment B.6: One commenter asked why the dose rate for the bottom of the MPC-68 was higher than for the MPC-24 when the dose rates at the side and top were higher for the MPC-24. The commenter stated that the trunnion doses showed that extreme care needs to be taken in those areas and that the bottom doses are really high and don’t get enough attention.

Response: The applicant appropriately assumed design basis fuel loadings for each canister and estimated dose rates at various locations. The NRC notes that dose rates at the bottom of the canister depend on several factors such as the fuel hardware characteristics, irradiation and cooling history, and the relative position of each fuel type within the cask system. The NRC found that the applicant appropriately addressed these and other factors, and that the calculated dose rates at the bottom and at the trunnions of the transfer cask were acceptable. In addition, each cask user will operate the HI-STORM 100 under a 10 CFR Part 20 radiological protection program and monitor dose rates during loading and unloading.

Comment B.7: One commenter asked what the dose for the 2 x 5 cask array was at 100 meters.

Response: Figure 5.1.3 of the SAR indicates that the dose rate for a 2x5 array at 100 meters is approximately 600 to 700 mrem/yr assuming a design basis fuel loading and 100 percent occupancy. Each general licensee will identify an ISFSI configuration and perform a site-specific dose evaluation, based partly on site-specific characteristics, to demonstrate compliance with Part 72 radiological requirements.

Comment B.8: One commenter asked why other cask designs do not account for approximate atmospheric conditions. The commenter also asked the conditions of weather or location for which the air density decreases.

Response: Atmospheric density changes daily. The measure of the density is provided by local weather forecasters through the barometric pressure. When a high pressure front passes an area, the air density is greater than when a low pressure weather front passes the same location.

The comment concerning other cask designs is beyond the scope of this rule that is focused solely on whether to place the Holtec HI-STORM 100 cask system on the list of approved casks. For the HI-STORM 100, each general licensee should consider atmospheric conditions relevant to its ISFSI as indicated in Section 5.4.2 of the SER.

Comment B.9: One commenter asked how much the releases from dry storage add to the effluent from a reactor site and the duration of a release, and what happens to the cask and fuel during the release.

Response: Specific effluent releases from reactors operated by general licensees are beyond the scope of this rule. However, NRC does not expect any effluent release from the HI-STORM 100 under credible conditions. Design basis public exposures from direct radiation and hypothetical releases are discussed in SER Sections 10.4 and 10.5.

Comment B.10: One commenter approved of the condition in Appendix B of the CoC regarding the evaluation of engineering features (e.g. berm) that are used for radiological protection by the user.

Response: No response is necessary.

Comment B.11: One commenter stated that average surface dose rates in TS 3.2.1 for transfer cask dose rates should not be used, that the highest value should be used, and the limit should not be exceeded. The commenter also asked why the side dose rates are measured along the middle of the flat surface section of the neutron shield rather than on the radial steel fins where dose rates are assumed by the commenter to be higher.

Response: The NRC disagrees with the comment. The specification of surface average dose rates and the measuring locations on the side of the neutron shield are consistent with health physics methods that are used to characterize radiation fields around a cask. The measuring locations are also consistent with the dose rate calculations presented in the applicant's shielding analysis. The cask user will operate the HI-STORM 100 under a 10 CFR Part 20 radiological protection program. NRC has reasonable assurance that the general licensee's radiological protection and ALARA program will detect and mitigate exposures from the radiation fields that are expected during operation of the HI-STORM 100 system.

Comment B.12: One commenter asked why the dose rate for the bottom of the transfer cask is not provided in TS 3.2.1 and what is that dose rate.

Response: Dose rate limits for the bottom of the transfer casks are not needed because they would not provide a significant benefit in ensuring compliance with regulatory limits on occupational dose and dose to the public. The dose limits at the top and side of the transfer casks are adequate to help ensure that the cask system is safely operated in compliance with 10 CFR Part 20 and Part 72. Calculated dose rates at the bottom of the transfer casks are reported in Sections 5.1 and 5.4 of the SAR.

Comment B.13: One commenter recommended that Section 5.1.2 of the SER be revised to clarify that overpack surface dose rates are design objectives and are shown to be met by analysis, and that the TSs are equal to or more conservative than the design objectives.

Response: The NRC disagrees with this comment. The NRC staff does agree that the vent dose rates calculated by the applicant are significantly less than the applicant's proposed design criteria. However, the differences between the calculated vent dose rates and the proposed design criteria are not relevant to the bases and findings in the SER. The TSs in Appendix A of the CoC specify vent dose rate limits for the overpack that are based on the applicant's shielding calculations. Therefore, a revision to the SER to reflect the dose rate difference is not necessary.

Comment B.14: One commenter recommended that Section 5.4.11 and Table 5.4-1 of the SER be clarified to indicate that the dose rates are not peak or maximum values.

Response: The NRC agrees with the comment. The SER has been clarified to state the vent dose rates are average over the area of the vent opening. A footnote has been added to Table 5.4.1 to clarify values are average over surface detector areas.

Comment B.15: One commenter recommended that Section 10.5.1 of the SER be revised to indicate that the maximum MPC leak rate is utilized in the calculations.

Response: The NRC agrees with the comment. The SER text has been revised accordingly.

Comment B.16: One commenter indicated there was an inconsistency between the accident condition whole body and thyroid dose values referenced in Chapter 11 of the draft SER and the dose values calculated in Section 7 of the applicant's SAR.

Response: The NRC agrees with the comment. The SER has been revised to indicate the correct whole body and thyroid dose values calculated by the applicant. The accident condition whole body total effective dose equivalent (TEDE) is 44.1 mrem and the thyroid dose is 4.1 mrem.

Comment B.17: One commenter objected to the use of a 30-day duration of a radiological release during an accident. The commenter noted that this assumption is stated in Interim Staff Guidance 5 but that it is not justified in the guidance or any accompanying report. The commenter pointed out that NRC regulations for ISFSIs do not require offsite emergency planning, or planning for the ingestion pathway zone, and therefore, there is no basis for assuming that something happens within 30 days to stop the release.

Response: The NRC disagrees with the comment. As indicated in ISG-5, Rev.1, the 30-day assumption is consistent with the time period that is used to demonstrate compliance with radiological dose requirements associated with reactor facilities that operate under 10 CFR Part 50. The applicant specified corrective actions for each accident in Chapter 11 of the SAR. NRC believes that these corrective actions can be reasonably achieved within 30 days. Although NRC does not expect effluent release from the HI-STORM 100 under credible accident scenarios, the 30-day assumption in the analysis is acceptable because the NRC staff has reasonable assurance that in the 30-day timeframe adequate protective measures can and will be taken for the public in the event of a radiological emergency. These protective measures include implementation of the general licensee's Part 50 emergency plan, evacuation of the surrounding public, and mitigation of radiological ingestion pathways.

Comment B.18: One commenter objected to the assumption that a person at the fence post (500 meters) would be exposed for only 2000 hours/year which is the number of working hours in a year. The commenter stated that 8,760 hours/year should be used because a licensee can not control who would be in the area outside the fence or how long they would be there. For conservatism, the applicant should have assumed that people, such as mothers with pre-school aged children, the elderly, ranchers, and farmers are present at the fence post day-long and year-round.

Response: The NRC agrees that 8,760 hours/year should be used and notes that Section 7.2.9 of the HI-STORM SAR explicitly states that: “The individual at the site boundary is exposed for 8,760 hours [7.0.2].” The NRC staff’s independent calculations confirmed Holtec’s calculated results, as stated in the NRC staff’s SER. In addition, Section 7.2.9 also assumed in its calculations that: “The distance from the cask to the site boundary is 100 meters.” With respect to hypothetical individual exposed at the site boundary, the methods used in the dosage calculations cover children, the elderly, ranchers, farmers, etc. The overall public dose limit is protective of all individuals because the variation of sensitivity with age and gender was accounted for in the selection of the lifetime risk limit, from which the annual public dose limit was derived.

The NRC continues to believe that the existing regulations and approved methodologies adequately address public health and safety. The issue of dose rates to children was addressed in the Federal Register on May 21, 1991 (56 FR 23387).

Comment B.19: One commenter stated that the dose due to direct gamma and ingestion of radionuclides should be considered in the dose calculation because to ignore these pathways underestimates the dose. The commenter further objected to the NRC staff stating (in the Holtec HI-STAR 100 final rule) that these pathways would be addressed in the general licensee's site-specific review. The commenter stated that there is no regulatory requirement for these actions to be taken by the general licensee. The commenter stated that it is misleading for the applicant to do a calculation that provides a reassuring result, based on assumptions that have nothing to do with the real requirements of the regulations because licensees tend to rely heavily on the generic analyses that have been performed by cask manufacturers.

Response: The NRC disagrees with the comment. Although the NRC does not expect effluent release from the HI-STORM 100 under credible conditions, the applicant's method used to determine design basis dose rates from a hypothetical release are adequate to demonstrate that the confinement features are sufficient to meet the radiological requirements of 10 CFR 72.106. The NRC staff believes the methods applied by the applicant conservatively bound hypothetical dose rates to the general public. Further, 10 CFR 72.212(b)(6) requires the general licensee to review its reactor emergency plan and radiation protection program to determine its effectiveness and make changes if necessary when using a cask listed in 10 CFR Part 72, Subpart L.

Comment B.20: One commenter stated that the thyroid and whole body doses should consider chlorine-36 (Cl-36) because it will be present in the irradiated fuel and will significantly contribute to the dose. The commenter points out that the Department of Energy acknowledges that Cl-36 is one of the significant radionuclides in Appendix A, of the Yucca Mountain Draft EIS.

Response: The NRC disagrees with the comment. The NRC staff's independent analysis of the thyroid and whole body dose was based on independent calculations using the ORIGEN computer code, as referenced by the commenter. The calculated contribution of the chlorine gas was below the truncation limit used in the calculation. Cl-36 has an inconsequential contribution on the total dose to an individual.

C. Accident Analysis

Comment C.1: One commenter asked if lead could be a missile strike barrier from a tornado or from current weapons. The commenter asked if missiles could penetrate the transfer cask and canister inside, and when the missile strike is assumed to occur (i.e. when a loaded transfer cask is on top of the overpack.) The commenter stated that this needs to be updated and evaluated.

Response: The lead backed outer shell of HI-TRAC has been evaluated for the required tornado missile strike. The analysis shows that there is no penetration consequence that would lead to a radiological release. The threat of missiles from weapons is beyond the scope of this rule.

Comment C.2: One commenter expressed concern that the transfer cask is a real target on top of the storage cask and asked if it had been fully evaluated for terrorism and sabotage, particularly when it was on top of the storage cask. The commenter asked if the overpack was put in place while on the pad; the commenter felt that this would be a target for terrorists. The commenter asked if the transfer cask, with inner cannister inside, could be knocked off by a terrorist blast and fall, crash, or roll into other casks or be upended so that the fuel is upside down.

Response: The performance of the transfer cask in a sabotage or terrorist event was not evaluated. The threat of terrorism or sabotage is beyond the scope of this rule. See also the response to C.8.

Comment C.3: One commenter asked if the seismic event was based on the actual pad analysis and not the reactor building seismic analysis because the conditions between the reactor building and pad location could significantly differ.

Response: The storage pad is a site-specific issue and is beyond the scope of this cask design rule. Under 10 CFR 72.212, the cask operators are required to perform written evaluations to ensure that storage pads have been designed to adequately support the stored casks. The licensee using a particular cask design has the responsibility under the general license to evaluate the match between reactor site parameters and the range of site conditions (i.e. the envelope) reviewed by the NRC for an approved cask.

Comment C.4: One commenter asked how a full cask array would behave in a seismic event. The commenter asked what buildings or equipment are allowed on the pad that could crash into the casks during a seismic event, such as the transfer equipment. The commenter asked if a crack or “push up” of the pad could cause the cask to roll (down an incline or into water).

Response: The SAR indicates that the HI-STORM 100 overpack will neither slide nor tip over due to a seismic event with the design-basis earthquake input listed in Section 3.4.2 of the SER. The use of a general licensed cask by a utility requires that the user ensure that the site is not subject to any potential accident that has not been analyzed for the general license. This would include any potential design basis earthquakes that were not enveloped by the NRC SER for the cask or any site conditions associated with the actual pad and cask locations that could affect the cask design.

Comment C.5: One commenter asked what the design-basis earthquake on top of the surface pad was and where it occurred. The commenter questioned why the bottom surface was not evaluated because the ground can push up and crack or cause heaving in the concrete and how the condition of the bottom surface is known.

Response: The design basis earthquake is the most severe earthquake that has been historically reported for a particular site and surrounding area, with sufficient margins for the limited accuracy, quantity, and period of time in which historical data have been accumulated.

Structure, systems, and components important to safety are designed to withstand the effects of this earthquake without loss of capability to perform their safety functions. The design basis earthquake is described by an appropriate response spectrum anchored at the peak ground acceleration. The response is then amplified through the pad to obtain the input response spectrum at the top of the pad (or at the bottom of the cask) for cask seismic evaluation. Soil and storage pad interaction is a site-specific issue that will be addresses in the cask user's 10 CFR 72.212 evaluation and is beyond the scope of this rule.

Comment C.6: One commenter asked what happens if the pad is cracked and heaving up as the cask is tipping over because a tornado or seismic event will likely affect both the pad and the casks.

Response: The NRC does not consider the scenario described by the commenter to be credible. The evaluation in Section 3 of the SAR shows that tipover will not occur. However, as a defense-in-depth measure, cask tipover is also evaluated in Section 3 of the SAR and discussed in Section 3.4.2 of the SER.

Comment C.7: One commenter asked if the cask could become upside down in a tornado or seismic event and if it happened would the top of the fuel hit the underside of the MPC lid with the weight on the overpack lid studs.

Response: The HI-STORM 100 overpack is evaluated for tornado, tornado missiles, and seismic events in Section 3 of the SAR. The results indicate that the cask will not tip over. Therefore, the cask will not become upside down.

Comment C.8: One commenter stated that an airplane crash with its fuel fire should be evaluated, including crash into a full cask array, damage to the pad, and a fuel and airplane explosion after the crash. The commenter stated that an anti-missile device with an incendiary device and a truck bomb should be analyzed for the cask transfer facility (CTF).

Response: The NRC disagrees with the comment. Before using the HI-STORM 100 casks, the general licensee must evaluate the site to determine whether or not the chosen site parameters are enveloped by the design bases of the approved casks as required by 10 CFR 72.212(b)(3). The licensee's site evaluation should consider the effects of nearby transportation and military activities.

The NRC reviewed potential issues related to possible radiological sabotage of storage casks at reactor site ISFSIs in the 1990 rule that added Subparts K and L to 10 CFR Part 72 (55 FR 29181; July 18, 1990). NRC regulations in 10 CFR Part 72 establish physical protection requirements for an ISFSI located within the owner-controlled area of a licensed power reactor site. Spent fuel in the ISFSI is required to be protected against radiological sabotage using provisions and requirements as specified in 10 CFR 72.212(b)(5). Further, specific performance criteria are specified in 10 CFR Part 73. Each utility licensed to have an ISFSI at its reactor site is required to develop physical protection plans and install systems that provide high assurance against unauthorized activities that could constitute an unreasonable risk to public health and safety.

The physical protection systems at an ISFSI and its associated reactor are similar in design features to ensure the detection and assessment of unauthorized activities. Alarm annunciations at the general license ISFSI are monitored by the alarm stations at the reactor site. Response to intrusion alarms is required. Each ISFSI is periodically inspected by NRC. Also, the licensee conducts periodic patrols and surveillances to ensure that the physical protection systems are operating within their design limits. The ISFSI licensee is responsible for protecting spent fuel in the casks from sabotage not the certificate holder. Comments on the existing regulations specifying what type of sabotage events must be considered are beyond the scope of this rule.

Comment C.9: One commenter questioned why the tornado missile test simulated a pulse impact of a vehicle and stated that a sharp object such as a metal pole or other items that might be in the vicinity of a real pad would do more penetration damage.

Response: In addition to the 4,000-pound automobile impacting at a 126 mph velocity, the SAR also provided analyses for two more missiles impacting at 126 mph velocity: a 1-in diameter steel sphere and an 8-in diameter rigid cylinder. Results of the analyses show that the 4,000 pound automobile produces the highest impact force on the cask because it has the largest mass. Based on these results, the NRC staff has reasonable assurance that the 4,000 pound vehicle bounds the effect of other credible types of tornado missiles.

Comment C.10: One commenter stated that the 15-minute transporter fire is not valid. A big plane crash with its fuel should be evaluated as well as a sabotage missile penetration with an incendiary device.

Response: The NRC disagrees with this comment. The basis for the 4.8-minute fire (not a 15 minute fire, see response to comment C.18) is associated with the time it would take to burn 50 gallons of fuel, presumably carried by the transporter. The CoC, Appendix B, Section 3.4, states 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." Other modes of transport causing the fire (e.g., airplanes, trains, delivery trucks or missiles) are not considered as plausible and are beyond the scope of this rule. Before using the HI-STORM casks, the general licensee must evaluate the site to determine whether or not the chosen site parameters are enveloped by the design bases of the approved cask as required by 10 CFR 72.212(b)(3). Included in this evaluation is the verification that no credible source of an external explosion that would produce an external pressure above that analyzed in the SAR and that any cask handling equipment used to move the HI-STORM cask to the pad is limited to 50 gallons of fuel (refer to CoC, Appendix B, Section 3.4 - Site Specific Parameters and Analyses).

Comment C.11: One commenter asked why there were no calculations for the bottom plate, overpack lid, etc. in a fire because the temperatures of these plates were important to know and could affect the pad or fire fighting equipment.

Response: The applicant did calculate the temperatures for the bottom plate and the overpack lid. However, these temperatures were not reported in the SAR. Not all calculated temperatures are reported in the SAR. With respect to the impact of fire on the pad or fire fighting equipment, a postulated 50 gallon fuel source would have minimal impact on those components. The heat generated by the pool of fuel is directed upward where the fuel is in a gaseous state. The limiting temperatures will occur above the surface of the concrete pad. Because the fuel has to vaporize in order to burn, the liquid fuel on the concrete will have minimal impact on the bottom plate of the overpack lid (in a liquid state, the fuel is cool). The duration of the fire is less than 4 minutes. The impact on the fire fighting equipment would be minimal, if any. Table 4-3 of the SER was modified to indicate that the temperatures were not reported.

Comment C.12: One commenter asked how the 45,000 MWD/MTU for 5 years related to the sabotage and terrorist evaluation for radiation disposal and stated that the evaluation is outdated.

Response: The comment on the sabotage report is beyond the scope of this rule. See the discussion in the response to C.8.

Comment C.13: One commenter asked if the water jacket could be pierced with an anti-missile gun or if a terrorist could shoot the jacket full of holes, and what are the consequences if these events did occur.

Response: The specific threat of an anti-missile gun or other small arms against the HI-STORM 100 is beyond the scope of this rule. However, the resultant dose rate for an assumed complete loss of the water jacket is addressed in Section 5.1.2 of the SAR. The analysis indicates that the off-site dose at 100 meters will be below the 5 rem accident limit in 10 CFR 72.106.

Comment C.14: One commenter asked why a burial under a landslide during a seismic event is not considered.

Response: Burying a cask due to seismic event, landside, or tornado is considered a very unlikely event. Considering the unlikeliness of the event coupled with the casks being able to withstand these events make burying and any adverse consequence in the opinion of the NRC not credible.

Comment C.15: One commenter asked why a vertical drop of a loaded transfer cask is not considered a credible accident, particularly as it is perched on top of the concrete overpack to load.

Response: A vertical drop of a transfer cask is not considered credible because vertical lifting of a loaded transfer cask must be performed with structures and components designed to prevent a cask drop. The criteria for those structures and components are specified in Section 3.5 of Appendix B to CoC No. 1014. The restrictions on vertical lifting are specified in Section 5.5 of the TSs (Appendix A to the CoC).

Comment C.16: One commenter stated that defense-in-depth is needed for sabotage events which could cause a tipover.

Response: The NRC disagrees with this comment. Sabotage events are beyond the scope of this rule. They are considered in Part 73. Furthermore, the SAR demonstrates that the HI-STORM 100 overpack will not tipover due to a design basis accident. However, as an added defense-in-depth measure, cask tipover is evaluated in Section 3 of the SAR and discussed in Section 3.4.2 of the SER.

Comment C.17: One commenter asked why a postulated explosion from a truck bomb at the pad fence was not evaluated.

Response: The specific threat of a truck bomb is beyond the scope of this rule. The response to C.8 addresses radiological sabotage of storage casks at generally licensed ISFSIs.

Comment C.18: One commenter asked the basis for the 217-second fire for the overpack and the 4.8-minute fire for the transfer cask. The commenter also asked if the NRC assumed that nothing on the vehicle or in the vicinity (such as grass or trees or other structures) will burn and cause the fire to burn longer.

Response: The duration of a fire burn is based on several factors. One factor is the rate at which the fire burns, normally categorized as inches of fuel burned per minute. The burn rate (inches per minute) is the same for both the overpack and the transfer cask because the source of fuel is the same (e.g., diesel fuel). The duration of the burn comes from the postulated depth of the pool of fuel. A conservative estimate of the time of burn is to assume that the spilled fuel does not extend beyond 1 meter of the surface of the overpack or the surface of the transfer cask. (In reality, the fuel will spill significantly farther than one meter on a flat surface, just as spilling a bucket of water on the ground, and will not accumulate to any significant depth which creates a shorter fire burn time.) Because the outer diameter of the overpack and the outer diameter of the transfer cask are different, the postulated depth or height that 50 gallons of fuel is postulated to reach will differ for the two cases. The case with the higher column of fuel will burn longer. Because the surface area of the pool of fuel for the overpack is 1.3 times larger than for the transfer cask, the pool of fuel for the overpack will be lower (given the same volume of available fuel, e.g., 50 gallons). A lower pool of fuel will burn quicker. (Note that the burn rate is in inches of fuel per minute, and a smaller column of fuel will burn quicker than a higher column of fuel). Therefore, the burn time for the overpack is shorter than the burn time for the transfer cask.

With respect to other flammable sources that could catch fire, before using the HI-STORM cask, the general licensee must evaluate the site to determine whether or not the chosen site parameters are enveloped by the design bases of the approved cask as required by 10 CFR 72.212(b)(3). Included in this evaluation is the verification that the cask handling equipment used to move the concrete cask to the pad is limited to 50 gallons of fuel (refer to CoC, Appendix B, Section 3.4.5) and that the assumptions used in the SAR bound the consequences for the proposed site. Additional assessments would have to be performed if other sources are identified that could result in a more limiting fire.

Comment C.19: One commenter objected to the use of the leakage rate used by Holtec because it is based on an analysis of a transportation cask rather than a storage cask, for which the NRC and industry have different design and testing requirements. The commenter noted that the small assumed leakage rate and calculation methodology in NUREG/CR-6487 are based on ANSI standard N14.5 for transportation casks. ANSI N14.5 assumes that casks will be leak-tested periodically before shipment and after maintenance and repair. The commenter pointed out that some ISFSIs have no design provisions for testing helium leakage during storage and no provisions for repairing and maintaining casks and testing for leakage after repair and maintenance. Therefore, it is inappropriate to assume that these storage casks will have the same small leakage rate as transportation casks for which leakage potential is designed and planned to be monitored. The commenter stated that neither Holtec nor the NRC has any basis for relying on NUREG-1617 to assume a small leakage rate in a storage cask breach.

Response: The NRC disagrees with the comment. The ANSI N14.5 standard was developed to determine allowable leak rates for shipping packages that employ mechanical seals, which typically undergo repetitive use. Periodic testing is prescribed for the mechanical seal to ensure it has not degraded from repetitive use and/or seal maintenance. The analytic technique in ANSI N14.5 that is used to determine a leak rate across an assumed leak path is valid for determining an assumed leak rate across the confinement boundary of a welded canister. An off-site dose can be subsequently calculated using standard atmospheric dispersion principles and assuming a partial release of the cask constituents at the calculated leak rate. The welded closure is leak-tested to a sensitivity equal to the calculated leak rate to ensure integrity of the confinement system before storage operations. Periodic testing of the confinement boundary is not applicable because the welded confinement boundary is designed to remain intact during normal, off-normal, and accident conditions for the lifetime of the canister.

Comment C.20: One commenter stated that the methodology used in NUREG/CR-6487 may not apply for accidents that exceed the design basis accident. The allowed leak hole size can easily be exceeded in accidents involving sabotage such as an impact with a MILAN or TOW-2 hand held anti-tank device, a jet engine, or military ordnance.

Response: The NRC disagrees with the comment. Consideration of accidents that exceed design basis is not required by 10 CFR Part 72 and is beyond the scope of the NRC staff's review. The threat of accidents involving sabotage is beyond the scope of this rule. Sabotage issues are covered by 10 CFR Part 73.

Comment C.21: One commenter stated that Holtec should consider a 300 gallon fire or a 6,000 gallon fire. The commenter stated that these are credible accidents at an ISFSI and should be considered. A heavy haul tractor carries 300 gallons of fuel and is likely to be used at ISFSIs. Locomotives that carry casks to ISFSIs may carry 6,000 gallons of diesel fuel.

Response: The NRC disagrees with the comment. The analysis need only address the maximum permissible source of fuel at the storage site near the HI-STORM 100 system (10 CFR Part 72). Section 3.4 of Appendix B to the CoC limits the source of fuel near the HI-STORM 100 system to 50 gallons. Licensees are required to verify that all conditions of the CoC are met.

Comment C.22: One commenter stated that Holtec's fire analysis is deficient because the fire calculations assume that the fire takes place outside the concrete storage cask and does not consider the possibility of a fuel fire being drawn into the intake vent of the HI-STORM 100 cask.

Response: The NRC staff disagrees with the comment. The purpose of the fire analysis is to assess the consequences of a postulated fire on the HI-STORM system. The elements of interest are the impact of the fire on the peak clad temperature and the impact of the fire on the system materials. A 50-gallon fuel supply will have a very short burn duration. Applying the conservative assumptions of 10 CFR Part 71, a 50-gallon fuel supply would theoretically result in a pool size of 0.54 inches if limited to a one-meter spread around the overpack. The burn duration of the fuel in this configuration is 3.6 minutes. This burn duration will have insignificant impact on the peak clad temperature. The heat capacity of the system is too great to have an appreciable feedback on the peak clad temperature for a short duration transient. The greatest impact of a fire will be felt on the overpack. A bounding analysis was performed on the overpack by imposing a maximum burn temperature (specified in 10 CFR Part 71) on the entire outer surface of the overpack. This maximizes the impact on the steel liner and the concrete. In a less conservative calculation, the maximum temperature will be limited to the lower portion of the overpack. For additional conservatism, the applicant increased the inside temperature of the overpack to 300°F to account for heating of the air as it passes through the vents. As illustrated in SAR Figures 11.2.1 - 11.2.5, this bounding calculation illustrates that only the outer boundary of the concrete exceeds the temperature limit for concrete. At a depth of one inch into the concrete the temperature limit is not challenged. If a conservative assumption postulates the fire to occur inside the vent, similar results would occur because there is only a limited amount of energy (BTU) that can be deposited into the massive overpack structure. Exceeding the concrete temperature limit only at the concrete surface does not lead to a safety concern, and therefore, the SAR analysis is acceptable.

Comment C.23: One commenter stated that the consequences of a hit by an anti-tank missile, such as the MILAN or TOW-2 missile should be considered. The commenter noted that the regulations only require a licensee to install systems that protect against unauthorized entry; however, entry to a site is not necessary to successfully carry out sabotage using an anti-tank missile. The commenter stated that the NRC should place additional conditions in the CoC to lower the probability of a sabotage event. The commenter further pointed out that the NRC has been inconsistent and arbitrary in determining whether to treat sabotage issues as site-specific or generic.

Response: The NRC disagrees with the comment. The threat of an anti-tank missile and other sabotage events is beyond the scope of this rule. Requirements on radiological sabotage are covered in 10 CFR Part 73 and apply to both ISFSIs and spent fuel storage cask designs. Therefore, comments on a specific threat or mode of attack are beyond the scope of this Part 72 rule. See also the response to C.8 addressing radiological sabotage of storage casks at reactor site ISFSIs.

D. Criticality

Comment D.1: One commenter objected to the assumption on the continued efficacy of the boral over a 20-year storage period because it has never been tested or proven.

Response: The NRC disagrees with the comment. The NRC staff does not consider the loss of fixed neutron poisons credible after installation into the cask because the poisons are fixed in place and contained. The neutron absorber is designed to remain effective in the HI-

STORM system for a storage period greater than 20 years and there are no credible means to lose the neutron absorber. Section 6.3.2 of the HI-STORM SAR describes the neutron absorber and its environment, and evaluated boron depletion due to neutron absorption. Section 9.1.5.3 of the SAR describes the testing procedures for the neutron absorber material. The neutron absorber material will be manufactured and tested under the control and surveillance of a quality assurance and quality control program that conforms to the requirements of 10 CFR Part 72, Subpart G. The compositions and densities for the materials in the computer models were reviewed by the NRC staff and determined to be acceptable. This material is not unique and is commonly used in other spent fuel storage and transportation applications.

Comment D.2: One commenter asked if Boral had ever been used in any dry storage casks before and if it had, how long and had it been tested. The commenter asked if this was an experiment with a new application. The commenter further asked what proof was available to show the continued efficacy of a boral panel. The commenter asked what other fuel storage and transport applications utilized Boral and stated that it should be documented in the SER.

Response: As described in SAR section 1.2.1.3.1, Boral has been used in environments comparable to those in spent fuel storage casks since the 1950s, and in spent fuel shipping casks for Canadian spent fuel in the 1960s. In the United States, Boral has been used in numerous other spent fuel transportation casks since the early 1980's and in storage casks since the early 1990's. Some of the casks that use Boral are the NAC-I28 S/T, NAC-S/T, the NUHOMS-24P, NUHOMS MP-187, and BMI-1. The NRC disagrees that the HI-STORM SER should include a list of other casks that use Boral. Information on other spent fuel casks and Boral is publicly available. The response to comment D.1 discusses the efficacy of Boral and why testing other than initial fabrication testing is not necessary.

Comment D.3: One commenter stated that a test should be conducted to verify the presence and uniformity of the neutron absorber in fabrication.

Response: The presence and uniformity of the neutron absorber is verified as described in Section 9.1.5.3 of the SAR. The neutron absorber material will be manufactured and tested under the control and surveillance of a quality assurance and quality control program that conforms to the requirements of 10 CFR Part 72, Subpart G.

Comment D.4: One commenter asked if water injection in unloading reflood could result in large amounts of steam generation and two-phase flow conditions inside the MPC cavity causing over pressurization of the confinement boundary and a potential criticality.

Response: As stated in SAR section 6.4.2.1, the HI-STORM system was evaluated with various water densities inside the cask. The cask met the design criterion of k_{eff} less than or equal to 0.95 for all credible flooding conditions. The cask is most reactive when filled with full density water. As can be seen in SAR Table 6.4.1, the cask reactivity decreases when filled with low density water (i.e., steam).

In addition, Section 4.5.1.1.6 describes the cask cooldown and reflood analysis during fuel unloading operation. This section of the SAR states that before reflooding the cask with water, the helium inside the MPC is cooled to below 200 °F which is below the boiling point of water. The procedures are outlined in Section 8.3.1 of the SAR with reference to TS 3.1.3. These procedures limit steam generation and two-phase-flow interactions with the fuel to acceptable levels, thereby preventing over pressurization of the MPC.

E. Design

Comment E.1: One commenter asked if there are three MPCs that are NRC-certified for storage and transfer because the SER states that they are evaluated and approved.

Response: As stated in Condition 1.b. of the CoC and in Section 1.1 of the SER, there are three types of MPCs that can be used in the HI-STORM 100 Cask System: the MPC-24, the MPC-68, and the MPC-68F. The MPC-24 holds up to 24 PWR fuel assemblies that must be intact. The MPC-68 holds up to 68 BWR fuel assemblies that may be intact or damaged (i.e., with known or suspected cladding defects greater than hairline cracks or pinholes). The MPC-68F holds up to 68 BWR fuel assemblies that may be intact, damaged, or in the form of fuel debris (i.e., with known or suspected defects such as ruptured fuel rods, severed fuel rods, and loose fuel pellets). All three MPCs have the same external dimensions. Section 1.2.1.1 and Table 1.2.1 of the SAR has been revised to clarify that there are three types of MPCs.

Comment E.2: One commenter asked how and to what the trunnion is attached, and what it is made of.

Response: The trunnions are attached by welds to the inner and outer shell and to the HI-TRAC top flange. The trunnions are fabricated of SB-637-N07718 steel and SA-350-LF3 steel.

Comment: E.3: One commenter stated that the concrete for the overpack should be reinforced and asked why it is not reinforced.

Response: The NRC disagrees with this comment. The main function of the concrete encased between the steel shells in the HI-STORM 100 overpack is shielding. The structural strength of the HI-STORM 100 overpack is provided by the inner and outer carbon steel shells. The concrete, on the other hand, will provide an added benefit to the HI-STORM 100 overpack because it will increase the stiffness and weight of the overpack to resist external forces due to seismic, tornado, and tornado missiles.

Comment E.4: One commenter asked if the pedestal could shift in movement and touch the liner or if it could corrode to the carbon steel liner or baseplate. The commenter also asked what the baseplate was made of and if a ceramic baseplate should be used.

Response: The pedestal consists of concrete, 17 inches thick, encased in a steel shell. This shell is welded to the steel overpack baseplate, and the weld is examined according to the ASME Code Section V. Stresses in the pedestal have safety factors exceeding 16. The pedestal will not shift. The exterior and interior surfaces of the overpack are coated with an epoxy paint to prevent corrosion. The overpack baseplate is made of carbon steel that meets the design criteria.

Comment E.5: One commenter stated that jamming of parts could be a problem because unjamming the parts could cause damage (during both loading and unloading). The commenter further asked if the cask had been tested for jamming and what the situation would be after 20 to 40 years in storage.

Response: Stainless steel shims, depicted in Detail T of drawing 1495, sheet 5, prevent the MPC from contacting the overpack interior and preclude the paint from being scraped during the operational steps. The drop accident analyses cause stresses which significantly bound the stresses that could occur during normal handling operations. Therefore, damage to the MPC during loading and unloading into the overpack is not credible.

The calculation in the SAR demonstrated that there will be no jamming of the MPC in the overpack under the most severe stack-up of tolerances. The cask has not been tested for jamming; however, a dry run of all operational steps is required before use of the system.

The license life of all overpack and MPC components is 20 years. The applicant engineered the overpack, HI-TRAC, and MPC for 40 years of design life. More detailed information regarding the service life of the overpack, HI-TRAC, and MPC can be found in Sections 3.4.11 and 3.4.12 of the SAR.

Comment E.6: One commenter stated that the clearances were not adequate. The commenter asked if the wet fuel would be inserted into the overpack in the same way as the dry run and what happens if the crane does not have the MPC completely vertical when inserting it in the overpack or if the HI-TRAC or pad is not level.

Response: There is no adverse tolerance stack-up that would prevent the insertion of the MPC into the overpack. Additionally, the dry run will verify that the MPC can be inserted into the HI-TRAC and overpack. All cell plates of the MPC are constructed of stainless steel that is not effected by immersion in water; therefore, the tolerances for the dry run would not change and the wet fuel will go in the same way as in the dry run.

Comment E.7: One commenter is concerned that the manufacturer's tolerances are not clear if fabrication is the minimum margin of safety or minimum clearance allowed.

Response: The most severe "stack-up" of manufacturer's tolerances provides sufficient clearance for insertion of the MPC into the HI-TRAC. The minimum clearance allowed is thus met. The cask could be manufactured to the minimum allowed clearances, but this would not reduce the minimum margin of safety.

Comment E.8: One commenter asked if there would be a problem if the radial clearance of the HI-TRAC MPC is at a maximum and the radial clearance of the MPC overpack is at the minimum allowed. The commenter asked how much leeway is allowed in fabrication in both of these radial clearance measurements.

Response: No operational problem exists if the radial clearance of HI-TRAC/MPC is at a maximum tolerance "stack-up" and the radial clearance of the MPC overpack is at the minimum tolerance "stack-up." These tolerances have been evaluated for all manufacturer's design criteria requirements and for all design temperatures. The largest allowable radial dimension of the HI-TRAC is greater than the smallest allowable radial dimension of the overpack. Fabrication requirements, including tolerances, are stated on the drawings. These tolerances provide sufficient clearance for operations.

Comment E.9: One commenter expressed concern over the 13/16-inch difference in the maximum MPC diameter and minimum overpack internal diameter because it was a minuscule amount for fabrication. The commenter also asked what was meant by average radial clearance of about 0.4 inches and stated that it was not a lot of clearance.

Response: The 13/16 inches is the minimum clearance accounting for tolerances between the MPC diameter and the channels/shims that are attached to inner shell of overpack. The channels/shims provide guidance for MPC insertion, position MPC within the overpack, and allow the cooling air flow to circulate through the overpack. The minimum clearance between the MPC and overpack inner shell is approximately 5 inches without the channels. Both the clearance between the MPC and channels/shims and between the MPC and overpack inner shell are considered to be acceptable. The SER has been changed to clarify that 13/16 inches is the clearance between the maximum MPC diameter and the channels/shims that are attached to inner shell of overpack rather than between the MPC and overpack inner shell. The average radial clearance is diametral clearance divided by two.

Comment E.10: One commenter asked what the computed decrease (page 3-9 of the SER) was related to. The commenter expressed concern that these were very small calculation amounts (0.11 inches) to depend on computer accuracy.

Response: The computed decrease of 0.11 inches is the calculated maximum decrease in the inner diameter of the overpack shell due to a tipover accident. The 0.11 inches decrease in the inner diameter of the overpack shell is not computed by computer simulation. Rather, it is computed by using a standard text book equation for deformation calculation. The deformation due to tipover is expected to be small. This calculation has been evaluated by the NRC staff and found to be acceptable.

Comment E.11: One commenter asked if the base under the pads would be the same at all sites and asked what is under the pad. The commenter is concerned that the pad evaluation has not received adequate attention because it is a crucial part of the tipover and drop evaluation.

Response: Each user is required to meet the site parameters in CoC, Appendix B, Section 3.4 that include specific requirements for the pad. Site characteristics will be investigated by each cask user and addressed in the cask user's 10 CFR §72.212 evaluation. The pad is a site-specific issue. Site-specific issues are beyond the scope of this rule that is focused solely on whether to add the HI-STORM 100 cask system to the list of approved casks.

Comment E.12: One commenter asked why there were two different weights for the transfer cask.

Response: As discussed in Section 1.2.1.2.2 of the SAR, the 100-ton transfer cask weighs less than the 125-ton transfer cask because it has a reduced thickness in lead and water. The 100-ton transfer cask is designed for facilities not capable of handling the heavier 125-ton transfer cask.

Comment E.13: One commenter asked why the bottom pool lid supported the weight of a loaded MPC plus water.

Response: During lifting of the transfer cask from the fuel pool there is water in the MPC

and the annulus. Therefore, the structural evaluation of the bottom pool lid of the transfer cask must consider all the applicable weights supported by the pool lid, including the water.

Comment E.14: One commenter stated that the cask should be up on something to air out the area under the cask to prevent rusting. The commenter questioned if the baseplate rusted if that could cause the cask to tipover or lean.   The commenter is concerned that if the canister ended up leaning against the inner liner of the concrete shell, it would cause blockage of the venting annulus and create a hotspot in the concrete.

Response: The NRC disagrees with this comment. The baseplate is coated with an epoxy type coating to prevent corrosion. Some rusting may occur at scratches in the coating. However, even a postulated extreme case, assuming no coating present, would not result in sufficient corrosion to cause an amount of leaning that would be significant.

Comment E.15: One commenter asked if there is any leeway for the pressure in the concrete encasement between the two carbon steel outer and inner liners, if the concrete had room to move, and if the concrete could split the outer carbon steel encasing it, particularly at the welds.

Response: The coefficient of thermal expansion of steel is only slightly greater than that of concrete, and the thermal gradient through the overpack wall, experienced during the extreme temperature criteria, was calculated to be approximately 40°F. This temperature difference and thermal coefficient of expansions do not cause the inner steel to apply significant force to the concrete in the overpack. The outer steel shell expands somewhat more than the concrete; therefore, the concrete has room for expansion and exerts no force on the outer steel plates.

Comment E.16: One commenter asked what a bottom pool lid is and how it is replaced by the heavier shielded transfer lid and if it has been tested.

Response: The bottom pool lid is described in Section 1.2.1.2.2 of the SAR. The lids are interchanged with a transfer slide device as described in Section 8.1.1 of the SAR. The NRC did not require test results for lid changing operations. The NRC found the pool and transfer lid design to be acceptable for the HI-STORM 100 system.

Comment E.17: One commenter asked if the 17.000 inches of concrete for the overpack baseplate was a typo and if the number of significant figures was correct.

Response: The value in the SER has been revised to state 17.0 inches that reflects the thickness assumed in the shielding analysis.

Comment E.18: One commenter stated that the configuration discussion in Section 6.4.1 of the SER is not clear because the HI-STAR doesn't have a transfer cask.

Response: As stated in SER section 6.4.1, the HI-STORM system has a transfer cask, not the HI-STAR system. The transfer cask, the HI-STORM overpack, and the HI-STAR overpack are constructed of different materials. The effectiveness of these materials to reflect neutrons affects the criticality safety of the system; therefore, each was explicitly evaluated. The other parameters affecting the criticality safety of the HI-STORM system, including the transfer cask, are identical to the HI-STAR system.

Comment E.19: One commenter asked if the closure ring was a ring or a lid.

Response: The closure ring is a ring. In the MPC, the lid and the closure ring are two different components.

Comment E.20: One commenter asked how many rings are included in the design.

Response: There is one closure ring included in the design.

Comment E.21: One commenter asked why voids in the installation of the lead shield are only minimized instead of being disallowed completely, if the shield was composed of lead bricks or poured, and which was more prone to voids. The commenter asked if lead bricks could be used and then have lead poured into the cracks between the bricks, and how the lead shield is installed.

Response: The HI-STORM 100 must be fabricated and tested in accordance with the drawings specified in the SAR and under a quality assurance program that meets the requirements of 10 CFR Part 72, Subpart G. The proper fabrication of the lead shield, including potential voids, will be evaluated under this quality assurance program. As discussed in Section 9.1.5.2 of the SAR, effectiveness of the lead pours are verified during fabrication by performing gamma scanning on all accessible surfaces of the transfer cask in the lead-pour regions. Installation of the lead shields is discussed in Section 9.1.5.1 of the SAR. The SAR specifies the use of poured lead and does not allow the installation of lead bricks.

Comment E.22: One commenter asked what the relief valve was and what type of maintenance it received.

Response: A relief valve is a mechanical device that opens when pressure inside a system exceeds the actuation pressure of the valve (pressure that will open the valve). Relief valves are common pressure limiting devices. Relief valves are placed on water heaters in homes to ensure that the water pipes in a house will not fail due to excessive pressure. Similarly, relief valves are attached to the radiator in a car to ensure that the coolant hoses do not burst from excessive pressurization of the engine coolant system. Maintenance of the relief valves are discussed in SAR Section 9.2.4. The relief valves are calibrated annually to ensure that their pressure relief setting is correct or they are replaced with factory-set relief valves.

Comment E.23: One commenter asked if there were holes in the shield jacket to add and drain things and indicated that holes would be a potential sabotage threat for someone to drain the jacket or add something dangerous to the water.

Response: There are drain holes in the water jacket end plate. The 125-ton HI-TRAC has two 1 ½-inch drain holes and the 100-ton HI-TRAC has four 3/4-inch drain holes. The resultant dose rate for an assumed loss of the water jacket is addressed in Section 5.1.2 of the SAR. The analysis indicates that the off-site dose at 100 meters will be below the 5 rem accident limit in 10 CFR 72.106. In addition, NRC regulations in 10 CFR Part 72 have established physical protection and security requirements for an ISFSI located in the owner-controlled area of a licensed power reactor site.

Comment E.24: One commenter stated that Conditions 1a and 1b of the CoC should both state that the cask system has two transfer casks.

Response: The NRC disagrees with the comment. Condition 1b of the certificate of compliance specifies that there are two types of transfer cask options: the 125-ton HI-TRAC and the 100-ton HI-TRAC. It is not necessary to repeat that information in Condition 1a.

Comment E.25: One commenter stated that there should be a drawing of the damaged fuel container in the CoC because the structure is not explained.

Response: The NRC disagrees with this comment. A drawing of the damaged fuel container is included in Chapter 1 of the SAR and is available to the public. This level of detail is not necessary in the CoC.

Comment E.26: One commenter asked what the screens are made of, how the screens are attached, if the screens can deteriorate or come loose over time, and what happens if the screens fall out.

Response: As shown on the drawings in Chapter 1 of the SAR, the damaged fuel container, including the screen, is constructed of stainless steel. The damaged fuel container is an additional structural component that will make the MPC fuel basket even stronger. The screen is placed between two steel plates welded together with a 0.06 inch, continuously 360 degree, all around fillet weld. It is not considered credible for the screens to fall off or fail. However, if a screen failed, there would be no release of radioactive material because the MPC is sealed. Small amounts of loose debris in the MPC have been considered during unloading operations, as described in SAR Section 8.3.4.

Comment E.27: One commenter stated that damaged fuel and intact fuel should not be placed in the same cask because it can cause potential problems in unloading.

Response: The NRC disagrees with the comment. Damaged fuel can be stored safely with undamaged fuel. If damaged fuel is stored with undamaged fuel, then CoC, Appendix B, Section 2.1.1.c requires all fuel assemblies in the cask to meet the more restrictive heat generation requirements for the damaged fuel. Additionally, damaged fuel must be loaded into damaged fuel containers to enable safe handling during cask loading and unloading operations.

Comment E.28: One commenter asked what the basis is for putting the hotter fuel in the center of the cask. The commenter also asked if the doses would be accurate if the lower dose fuel is placed at the periphery positions. The commenter stated that it would be better to have a more even heat and dose distribution in the MPC and asked if dose was more important than the heat.

Response: The design of the HI-STORM cask considered both the thermal and radiological effects of the fuel. If one assumes the same enrichment and burnup (time that the fuel was left in the reactor to produce power), the fuel that is left longer to decay in the spent fuel pool (e.g., "cooler fuel") will generate less heat and high-energy radiation than the fuel that is removed sooner from the pool (e.g., "hotter fuel"). For the method used in the HI-STORM design, cooler fuel assemblies are stored on the periphery of the cask for two reasons. First, the "cooler fuel" assemblies have lower allowable peak clad temperature limits and the temperature of the assemblies on the periphery is cooler. Second, storing the "cooler fuel" on the periphery of the MPC provides some additional radiological shielding from the hotter fuel assemblies in the center.

Comment E.29: One commenter asked if BPRAs and thimble plugs could be stored in the cask. The commenter stated that they should not be because they add weight.

Response: BPRAs and thimble plugs have not been analyzed for this cask system and, therefore, are not authorized for storage in the HI-STORM 100 at this time.

Comment E.30: One commenter asked what the cask transfer station is and whether it had been designed yet. The commenter asked if it is constructed of reinforced concrete. The commenter stated that more explanation was necessary and that a drawing should be included. The commenter asked how and why an impact limiter is used. The commenter asked what the basis is and if an evaluation had been completed. The commenter was concerned with the use of terms “if” and “shall be designed” because this implies the CTF hasn’t been designed. The design should have specific criteria.

Response: The term “cask transfer station” in CoC, Appendix B, Section 3.5.1, is a typographical error and has been corrected to “cask transfer facility.” A cask transfer facility (CTF) is a facility used for transferring the MPC between the transfer cask and the overpack. The CTF does not include 10 CFR Part 50 controlled structures such as the fuel handling building or reactor building. The NRC disagrees that a drawing of the CTF or more design details are necessary. The HI-STORM 100 Cask System is approved for use under the general license provisions of 10 CFR Part 72. Therefore, the cask may be used in any nuclear power reactor site licensed under 10 CFR Part 50, provided that the site parameters are enveloped by the cask design bases. The specific design and operation of the CTF will be dictated by site-specific needs. Because of the varied needs of each reactor site, the NRC found it impractical and unnecessary to review and approve a specific CTF design, including the specific materials of construction. The NRC reviewed and approved the criteria for the design, construction, and operation of the CTF. These criteria are specified in CoC, Appendix B, Section 3.5, and SAR Section 2.3.3.1.

The impact limiter is a possible CTF design feature whose function would be a defense-in-depth measure because the lifting equipment used in the CTF must be designed to preclude a drop. As discussed in the response to comment J.14, the specific requirement for an impact limiter has been eliminated because other methods may be available to prevent a canister breach in case of a canister drop during transfer operations.

Comment E.31: One commenter stated that we should clearly state what the CTF is and make sure that every detail of the procedure is carefully analyzed because it is vague.

Response: The CTF is defined in SAR Section 2.3.3.1 and in CoC, Appendix B, Section 1.0. Detailed design and operational requirements for the CTF are also specified in SAR Section 2.3.3.1, as well as in CoC, Appendix B, Section 3.5. Under the provisions of 10 CFR 72.212 and CoC Condition 2, each licensee that elects to use a CTF must develop written procedures for operating the CTF. These procedures are subject to NRC review during inspection. As required by TS 5.2.h, the licensee must conduct a dry run training exercise, prior to first use of a CTF, to demonstrate that the procedures can be conducted safely and successfully.

Comment E.32: One commenter recommended that Design Drawing 1495, Sheets 4 and 6 and Design Drawing BM-1575, Sheet 2 be revised.

Response: The NRC agrees with the comment. These changes correct drafting errors or provide a level of flexibility that is acceptable to the NRC staff. The SAR drawings have been revised accordingly.

F. Welds

Comment F.1: One commenter asked how use of the trunnions puts stress on the weld at the water jacket.

Response: Use of the pocket trunnion does not put any stress on the water jacket. The seal weld between the pocket trunnion and water jacket shell is for retaining the water inside the water jacket. The pocket trunnion is attached to the outer transfer cask shell by full penetration welds all the way around the trunnion. When the pocket trunnion is used, the force is transferred to this weld and not to the seal weld on the water jacket. The other type of trunnion on the transfer cask is the lifting trunnion. The lifting trunnion is not connected to the water jacket and, therefore, puts no stress on the water jacket.

Comment F.2: One commenter asked how the welds are checked to be leakproof and whether water can enter the trunnion.

Response: All the structural welds have to be examined and inspected according to the applicable ASME code. The welded joint is an integral part of the structure and is leak proof. Because the trunnions are made of solid steel, water cannot leak into them.

Comment F.3: One commenter stated that the lid and closure ring of the MPC should be full penetration welds and should be ultrasonic tested (UT) as this is the basis for qualification as a redundant seal.

Response: The NRC disagrees with this comment. Full penetration welds are unnecessary from the structural and containment boundary requirements of the design. Employing unnecessarily heavy welds leads to fabrication problems such as excessive warpage. UT of heavy section, full or partial penetration, austenitic stainless steel welds to ASME Code acceptance criteria is not feasible with current technology. The redundant seal concept is based upon the use of two welds forming the leak barrier, not the inspection method. With redundant welds, one weld could leak and the second still provide leak tight integrity.

Comment F.4: One commenter stated that just because there is no known plausible, long-term degradation mechanism to cause seal welds to fail doesn't mean that the welds won't fail.

Response: The NRC disagrees with this comment. The NRC staff has examined the plausible mechanisms that would cause failure of the seal welds and has determined that those mechanisms are inoperative under normal service and design accident conditions for HI-STORM. This gives the NRC staff reasonable assurance that the welds will not fail under design basis normal, off-normal, and accident conditions.

Comment F.5: One commenter asked how lid welds are removed in unloading and stated that the procedures should be in the documents before the casks are loaded.

Response: The NRC disagrees with this comment. Welded cask lids may be removed by one of several methods. The method of removal and the detailed procedures (as opposed to the general procedures of the SAR) are the responsibility of the ISFSI licensee, subject to NRC review and inspection. The TSs require ISFSI licensees to perform lid removal method demonstrations on full-size mock-ups as part of their pre-operational testing and training exercises. The NRC staff has reviewed and inspected several methods and their associated procedures. Inclusion of such detailed procedures in the SAR is unnecessary.

Comment F.6: One commenter asked what happens if the water jacket welds leak water.

Response: The resultant dose rate for an assumed loss of the water jacket is addressed in Section 5.1.2 of the SAR. The analysis indicates that the off-site dose at 100 meters will be below the 5 rem accident limit in 10 CFR 72.106.

Comment F.7: One commenter stated that the penetrant test (PT) is unacceptable, that the criteria for layers and time are "wishy washy," and that PT tests should not be allowed.

Response: The NRC disagrees with this comment. The commenter has not specified why PT is unacceptable or why it should not be allowed. PT is a Code accepted examination method. The progressive PT technique is used and accepted in the nuclear industry when a volumetric examination by means such as UT is impractical. UT is unsuitable for heavy section austenitic stainless steel welds.

The basis for the structural lid weld examination methods is documented in the NRC's Interim Staff Guidance-4, Revision 1, that allows the use of a multi-layer (i.e., progressive) PT examination in lieu of a volumetric examination.

Comment F.8: One commenter stated that welds need to be checked carefully.

Response: The NRC agrees with the comment. Welds are important which is why they are examined and inspected according to the applicable ASME code.

Comment F.9: One commenter stated that the leak testing procedure used to demonstrate MPC closure cannot be performed as described and that performance of the test is not generally consistent with ANSI N14.5-1997, “Leakage Tests on Packages for Shipment.” Consequently, containment of the radioactive material to the stated criteria cannot be demonstrated. The principal reason provided by the commenter is that the nominal concentration of helium in air is 5 parts per million. This atmospheric concentration masks any leakage from the MPC using the specified test conditions. In addition, the commenter noted that there is no direct reference to definitions, equations, formula, methodology, or criteria of the standard in the text. The commenter further noted that when terminology from the standard is given, it is (for the most part) used incorrectly.

Response: The NRC disagrees with the comment. These welds are multipass stainless steel welds that are dye penetrant examined multiple times during the weld process. The multiple dye penetrant examinations provide reasonable assurance of high integrity welds that will retain the inert gas and prevent leakage of radioactive material into the environment. The leakage testing of these welds provides additional insight into the leak-tightness of these welds.

The NRC staff has reasonable assurance that the leakage test procedure outlined in SAR Section 8.1 can be performed as described provided that appropriate equipment is used and the leak test method is properly qualified. The leakage testing for the lid is to be performed with a sniffer type probe; however, the test method for the port and drain covers is not specified in the SAR. There are test methods discussed in Appendix A of ANSI N-14.5 that could be used to perform the port and drain cover leakage testing. Detailed procedures are developed by the user who is responsible for ensuring that the TS limits are met and therefore, confinement is adequately maintained. The leakage testing will require a demonstration that the detector can identify an appropriate calibrated leak in the presence of background helium. Although sniffer probes detect discrete leaks rather than an integrated leakage rate, the typical sniffer probe sensitivity of 10^{-8} provides reasonable assurance that the TS leakage rate limit of $5x10^{-6}$ will not be exceeded.

As stated in the SAR, the leak testing will be performed in accordance with ANSI N14.5. It is not necessary to include any more level of detail in the SAR; therefore, no change to the SAR is necessary. Appropriate detail will be included in the site procedures.

The SAR has been changed to use terminology consistent with the TS and the 1997 revision of ANSI N14.5. The terminology was changed from std cc/s to atm cc/s. SAR section 7.3.3 justifies the use of the units atm-cc/sec. Also, the SAR was revised to delete the sensitivity of the detector. The sensitivity will be addressed in the detailed site procedures.

G. Structural

Comment G.1: One commenter stated that all of the accident level events and conditions listed in the SER, particularly a transfer cask handling accident or sabotage, should be evaluated for structural analysis in a jamming condition on top of the overpack.

Response: All design basis normal, off-normal, and accident events have been evaluated in the structural analyses and are discussed in Section 3 of the SER. This includes an evaluation of the transfer cask under a 42-inch horizontal drop during transfer operations. A horizontal drop from a greater height is not considered because the horizontal lifting height limit for the transfer cask is 42 inches. The evaluation shows that the HI-TRAC meets all structural requirements and there is no adverse effect on the confinement, thermal, or subcriticality performance of the contained MPC.

As discussed in the response to C.15, vertical drop of a transfer cask is not considered credible because vertical lifting of a loaded transfer cask must be performed with structures and components designed to prevent a cask drop. Also, as discussed in the response to E.5, jamming is not considered to be credible because of the design of the HI-STORM system. The threat of sabotage is beyond the scope of this rule and is discussed in the response to C.8.

Comment G.2: One commenter stated that bending of the web and pushing of the flanges possibly accompanied by some local weld failures sounded feasible and may not result in limited deformation as assumed. The commenter asked if a full size cask had been tested in a drop or tipover.

Response: The channels attached to the inner shell of the overpack are not classified as important-to-safety and serve no structural purpose. Deformation of the channels, whether limited or complete collapse, does not affect retrievability of the MPC. On the contrary, the deformation of these channels due to a tipover accident absorbs energy which reduces the deceleration loadings to the MPC and provides a greater opening in the overpack during retrieval. NRC regulations do not require full size testing of casks. The applicant can choose the method of analysis. Computer analyses have been performed to determine the responses of a cask in drop and tipover accidents.

Comment G.3: One commenter questioned how the structural analysis conducted for the 125-ton HI-TRAC transfer cask could bound the 100-ton version and indicated that the 100-ton version needs its own analysis.

Response: All the structural analyses and evaluations of the 125-ton transfer cask were repeated for the 100-ton transfer cask. However, the analytical results of the 125-ton transfer cask are greater than that of the 100-ton transfer cask. Therefore, the structural analysis of the 125-ton transfer cask bounds the 100-ton transfer cask.

Comment G.4: One commenter asked what would be the consequences of the deformation of the outer shell and lead and water jacket from a missile, particularly if the transfer cask was on top of the concrete shell.

Response: The HI-TRAC transfer cask is always held by the handling system while in a vertical orientation completely outside of the fuel building. Therefore, considerations of instability due to a tornado missile strike are not included in the evaluation. However, a structural evaluation of the damage to the HI-TRAC transfer cask from an intermediate missile strike and a large missile strike is performed. The evaluation shows that the outer shell and the water jacket would not experience any plastic deformation and will not adversely affect the retrievability of the MPC.

H. Materials

Comment H.1: One commenter questioned why carbon steel was used for the inner and outer plate instead of stainless steel because of the concern over corrosion. The commenter also asked if the carbon steel was coated.

Response: The materials used in the fabrication of the cask are described in Chapters 1 and 3 of the SAR and discussed in Section 3.3 of the NRC SER. These materials have been found acceptable because they meet the requirements for their respective applications in the cask system. They are suitable for the expected loading and storage in wet and dry environments, including corrosion and galvanic effects. There is no requirement for designers to select materials from a given class, e.g. stainless steels.

The carbon steel used in the overpack is protected from corrosion by an industrial epoxy coating commonly used for the protection of steel.

Comment H.2: One commenter stated that one alloy should be specified for cask fabrication instead of allowing a choice because if later problems develop, there are fewer variables.

Response: The NRC disagrees with the comment. The materials used in casks are selected on the basis of the needed properties. Allowing a choice of more than one material or alloy for fabrication is acceptable provided that each of the options has the appropriate properties. The materials chosen for use in the Holtec HI-STORM 100 design have a long history of favorable performance in the nuclear industry.

Comment H.3: One commenter questioned why plain concrete is not included in NUREG-1536 and why an exemption was being given to allow plain concrete since reinforced concrete is stronger.

Response: No exemption was given to allow plain concrete to be used for structural components. The plain concrete in the HI-STORM 100 overpack is for shielding only and is not a structural component of the overpack. The reinforced concrete included in NUREG-1536 is for concrete structures (concrete components that provide structural strength) only. The HI-STORM 100 overpack is a welded steel structure, not a concrete structure.

Comment H.4: One commenter asked if the NRC has reviewed the manufacturers direction for the carboline 890 and thermaline 450 coatings. The commenter asked how the coatings are used and applied, and if they will wash or flake off in pool water, making the water cloudy.

Response: The NRC staff has reviewed the manufacturer’s technical information for the coatings mentioned. Both coatings are standard coatings employed in industry for immersion service and are applied using common industry tools and techniques. No performance problems would be expected during intended service.

Comment: H.5: One commenter asked if the carbon steel caused reactions that could create loading or unloading problems such as reaction products clogging venting or draining equipment with crud or flakes or making the pool water cloudy.

Response: Carbon steel exposed to the cask loading environment produces very fine particulates that do not clog equipment. Turbidity that may arise from corrosion of uncoated carbon steel can be controlled with appropriate water treatment equipment.

Comment H.6: One commenter asked if temperature or coatings on the channel could affect the fit. The commenter also asked if flaking of the coating could clog a channel slide or if corrosion in the channels could cause problems in unloading.

Response: The effects of temperature on the channels have been calculated and do not affect the fit. Each coat of the epoxy paint applied to the exposed surfaces of the inner components of the overpack is, at maximum, 0.008 inches thick. Two coats result in a maximum diametral reduction in inside diameter of 0.032 inches. This reduction will not affect the fit. Both the interior and the exterior of the channels are coated to prevent corrosion.

Comment H.7: One commenter asked if aging was factored into the analysis of the pad and stated that the specific site should be evaluated for a full cask array.

Response: Concrete is resistant to environmental conditions, including air pollution and moisture. Therefore, the NRC staff expects no significant degradation of the pad during the licensed lifetime of the ISFSI facility. Each proposed site is subject to a specific evaluation to ensure that the design parameters satisfy site-specific conditions. In addition, cask users are responsible for inspecting and maintaining the pad, and for ensuring that significant degradation is not occurring over time.

Comment H.8: One commenter asked what the condition of the concrete is right under the shell and expressed concern that the concrete could crack where nobody would see damage needing repair.

Response: As discussed in the response to E.3, the main function of the concrete encased between the steel shells in the HI-STORM 100 overpack is shielding. The structural strength of the HI-STORM 100 overpack is provided by the inner and outer carbon steel shells. Cracking of the concrete would not have a significant impact on the cask's ability to meet the regulatory dose limits. There is no credible mechanism for the concrete to undergo any significant damage. Thus, inspection of the concrete is not necessary.

Comment H.9: One commenter asked if concrete expanded or released water or gas when it is superheated.

Response: Concrete contains some traces of free water. If the water is heated, it will evaporate. Concrete will expand upon heating and contract upon cooling. The amount is governed by the temperature. These expansions/contractions are reversible and not permanent. There are no significant effects of expansions/contractions that would occur even if the temperature went considerably beyond the design temperature parameters.

Comment H.10: One commenter asked how the bottom face affects the supporting surfaces (heat, radiation, weight, stress, pressure etc.).

Response: As listed in Table 4.4.9 of the SAR, the temperature of the bottom lid plate at normal conditions is 183°F. That temperature will not have an adverse effect on the concrete. Radiation will have minimal impact on the concrete pad due to the shielding provided by the pedestal. The weight, stress, and pressure from the cask bottom have no adverse effect upon the pedestal or slab because they are specifically designed to support all the loads due to the casks.

Comment H.11: One commenter asked how the gas and liquid media that escapes from the damaged fuel container interacts with other materials in the MPC and if they can cause problems.

Response: The materials of the cask have been selected to be compatible with any constituent or reaction product of the fuel.

<u>I. Thermal</u>

Comment I.1: One commenter asked if hot spots in the cladding could cause lead to sag in the transfer cask if the inner canister is in place and the temperature is close to the boiling point of the water pack.

Response: Hot spots in the cladding would not result in sagging or melting of the lead. The bounding calculation performed by Holtec assumed all the fuel assemblies were at the design basis limit (hottest assemblies). The bounding rod cladding temperature occurs at the center of the MPC and does not have a direct impact on the lead. The assemblies on the periphery of the MPC are significantly cooler because they are located near the cooler surface of the MPC. Table 11.2.8 in the SAR provides the results from a calculation that assumes no water in the water jacket. These results bound the impact of boiling in the water pack. Based on those results, it can be concluded that the lead temperature remains well below the melting temperature.

Comment I.2: One commenter asked what happens if the water in the transfer pack boils and the steam pressure builds up, and stated that this situation should be evaluated.

Response: As the pressure builds up, the pressure is relieved through a safety valve. As water is removed through the safety valve, the temperature of the water remains at the saturation temperature. The case of water boiling in the HI-TRAC water jacket is bounded by the event that assumed no water in the water jacket. This event leads to a temperature in the water jacket that is higher than the saturation temperature of the water. The impact of loss of water in the water jacket is summarized in Table 11.2.8 of the SAR.

Comment I.3: One commenter asked how, during normal conditions, the temperature of the outer shell could be higher than the temperature of the concrete because the carbon steel would breathe less than the concrete, causing the heat to be retained in the concrete. The commenter also asked how the temperature of the concrete could be measured since it is encased in the carbon steel.

Response: The question raised by the commenter is not clear. The temperature of the concrete is higher than the temperature of the outer shell under normal conditions. Reviewing Table 4.4.9 in the SAR, the temperature at the overpack outer shell is not higher than the concrete cross sectional average temperature. The temperature distribution through the overpack under normal conditions is listed in Table 4.4.9 of the SAR (e.g., 149 ^{o}F for the concrete and 131 ^{o}F for the outer shell).

With regard to the question of measuring the temperature of the concrete, the applicant does not measure the temperature of the concrete. Bounding calculations are used to assure that the concrete temperature limits will not be exceeded.

Comment I.4: One commenter asked how the pad reacts to the bottom plate of a cask from a temperature differential standpoint. The commenter asked if the pad would crack and sink under each cask and form a concave area that could then collect moisture. The commenter further asked if the collected moisture could boil and if the moisture could cause the bottom plate to rust.

Response: The heat transfer between the bottom plate of the overpack and the concrete pad is modeled in the thermal computer code for the HI-STORM cask system. As listed in Table 4.4.9 of the SAR, the temperature of the bottom lid plate at normal condition is 183 °F. That energy is transmitted to the concrete pad down to the ground, which is at the normal soil annual average temperature of 77 °F. Therefore, the concrete will not experience temperatures above boiling (no superheating will occur). In the winter, the concrete will not reach freezing temperatures below the cask because it generates heat. If the concrete reaches or exceeds boiling or freezing temperatures, there is no detrimental effect on the strength or condition of the concrete. The pad is specifically designed to support the weight of the casks without any cracking or sinking of the pad. The bottom plate of the cask is stainless steel and will not rust.

Comment I.5: One commenter questioned the basis and validity of simulating the heat effect of adjacent casks radiating heat back to an interior cask and if an analysis of the real situation had been conducted.

Response: The impact of radiation heat transfer from neighboring casks was calculated in the HI-STORM thermal evaluations. The method used by the applicant was to assume that all of the radiated heat is reflected back to the cask. This modeling assumption is equivalent to assuming that the cask was totally encircled by other casks. In reality, less heat will be radiated back to the cask; therefore, the calculations bounded the effects of neighboring casks. The impact of neighboring casks was shown to be minimal. The NRC staff does not require validation of the analytic method with actual experimental data.

Comment I.6: One commenter asked why the analysis assumed that the soil below the overpack was at a constant temperature because the casks could cause hot spots.

Response: The analyses did model the hot spots below the overpack. The computer simulation of the overpack modeled the concrete pad that the overpack is placed on and the temperature of the soil below the concrete pad. The soil is one of several paths for heat to leave the cask. The most significant path for heat dissipation is through the air passage between the MPC and the overpack. The applicant used the highest annual average soil temperature found in the USA. The purpose for using the highest average temperature for the soil and air in the thermal analyses is to demonstrate fuel retrievability and that the cladding is protected during storage against degradation that leads to gross ruptures(10 CFR Part 72.122). One acceptable method for demonstrating that the cladding will not undergo gross rupture is to place a limit on the allowable cladding temperature such that reasonable assurance exists that the cladding will not significantly degrade. A report by the Pacific Northwest National Laboratory, PNL-6189, dated May 1987, provides one acceptable approach for establishing a temperature limit. The PNL method is conservative when compared to the maximum allowable degradation permitted in Part 72 of the regulations. This method, in conjunction with the maximum annual average temperature, solar heating (e.g., insulation), analytic assumptions, etc., provide reasonable assurance that the requirements of Part 72 will be met.

Comment I.7: One commenter asked why an exception was allowed for exceeding the short term temperature limit for the fire accident scenario and stated that an exception should not be allowed.

Response: The American Concrete Institute (ACI) establishes temperature criteria for concrete. One, but not the only, acceptable demonstration that the concrete overpack will maintain its intended function is to meet the temperature criteria in ACI 349. However, as stated in the NRC staff's Standard Review Plan (NUREG-1536), "a small amount of exterior concrete spalling may result from a fire, the application of fire suppression water, rain on heated surfaces or other high-temperature condition. The damage from these events is readily detectable, and appropriate recovery or corrective measures may be presumed. Therefore, the loss of such a small amount of shielding material is not expected to cause a storage system to exceed the regulatory requirements in 10 CFR 72.106 and, therefore, need not be estimated or evaluated in the SAR. The NRC accepts that concrete temperatures may exceed the temperature criteria of ACI 349 for accidents if the temperatures result from a fire." The Holtec analysis demonstrated that the amount of concrete that exceeds the ACI temperature limit is very limited and would not pose a significant safety hazard.

Comment I.8: One commenter asked for the basis of using an average temperature of the gas in the gap and plenum of the limiting rod and questioned the validity of the assumption.

Response: The purpose for evaluating the average of the gas temperature in the fuel rod is to calculate the pressure within the fuel rod. The computer code used in the analysis calculates the temperature profile of the fuel rod, but does not calculate the corresponding pressure for that rod. To calculate the pressure, the average temperature of the gas is calculated and from the ideal gas law, the corresponding pressure is established.

Comment I.9: One commenter asked what is in the water used for forced water circulation under wet transfer of the fuel from the spent fuel pool to the location for vacuum drying and if the water could chemically affect other materials in the cavity. The commenter asked how fast the water flows, if steam could be formed, and if the water could physically affect other materials in the cavity, movement of rods, flaking of paint, etc.

Response: The licensee can either use demineralized water or water from the spent fuel pool. Neither demineralized nor spent fuel pool water would adversely interact with the system. The flow rate of the water is based on the heat output of the fuel assemblies and is a site-specific issue.

Comment I.10: One commenter asked what the water chiller is used for and what material is used as the chilling medium.

Response: The water chiller is used as the heat sink for cooling the helium inside the MPC to below 200°F. The type of water chiller used is a site-specific issue and not part of this rulemaking activity.

Comment I.11: One commenter asked for specific criteria that defines clearance around the cask for cooling purposes instead of stating a reasonable amount. The commenter also asked how close other heat sources may be located and what is considered to be a significant heat source.

Response: The actions identified by the commenter are only valid when a breakdown occurs in the helium coolers (LCO 3.1.3). Section B3.1.3 of the technical specification bases states that “if the TRANSFR CASK is located in a relatively open area such as a typical refuel floor, no additional actions are necessary." However, a licensee may elect to perform the cooling with the cask located in a pit or vault. This is a site-specific activity. The bases identify three acceptable options for ensuring adequate heat transfer for the TRANSFER CASK. The user may develop other alternatives on a site-specific basis, considering actual fuel loading and decay heat generation within the cask. One of the options is to fill the annulus between the MPC and the TRANSFER CASK with water. The second option is to remove the TRANSFER CASK from the pit or vault and place it in an open area such as the refueling floor with a reasonable amount of clearance around the cask and not near a significant source of heat. The third option is to supply nominally 1000 SCFM of ambient air to the space inside the confined space (e.g., pit or vault). With respect to defining an acceptable distance, the licensee could use the analyzed event of 15 feet center-to-center storage spacing that corresponds to a four foot clearance. Smaller clearances would also be acceptable, given the heat load rating of the cask and ambient conditions. With regard to defining a significant heat source, this is a site specific consideration. For example, if the plant is using a bank of radiant heaters near the cask, then an evaluation needs to be performed to ensure that those heaters pose no adverse impact on the cask. These options are only guidelines to an LCO that a user would have to consider.

Comment 1.12: One commenter asked how cool air was provided to the space inside the vault at the bottom of the overpack. The commenter stated that this needs to be planned out ahead of time for ALARA considerations and equipment availability.

Response: This is a site-specific issue and not part of this rulemaking activity. The NRC staff agrees with the comment that the user needs to plan this activity considering ALARA and equipment availability.

Comment I.13: One commenter stated that fuel should be adequately cooled before it goes into the transfer cask.

Response: The NRC agrees with the comment that adequate planning is needed when performing cask cooldown and reflooding. The purpose of the analyses performed in the SAR is to maintain the integrity of the fuel. The requirements on the burnup and minimum cooling time serve that purpose.

Comment I.14: One commenter asked if the temperature of the helium accurately reflects the internal temperature of the MPC and stated that this should be tested.

Response: The exit temperature of the helium reflects the conditions of the fuel rods. After the helium temperature is reduced below 200°F, the bulk of the fuel will be at low temperatures, minimizing the potential for excessive steaming. Reflooding of a canister has been demonstrated without pre-cooling the helium. No additional tests are needed.

Comment I.15: One commenter objected to the addition of ethylene glycol solution to the demineralized water in the water jacket to prevent freezing and asked where this had been tested. The commenter also asked why the antifreeze was used, how the solution would mix, how the NRC knows it will work, what types of effects it could have on the inside of the water jacket and the channel walls, if it would add weight, and how it is added to the water if the jacket is welded shut.

Response: Ethylene Glycol is the chemical name for ordinary antifreeze. Adding antifreeze to the water jacket, located on the outside of the HI-TRAC transfer cask, is an option if the user elects to move a loaded MPC in cold weather, down to 0°F. The use of antifreeze in the water jacket does not add appreciable weight to the HI-TRAC cask. Although the water jacket is a welded system, openings are designed to add and remove water from the water jacket. Antifreeze has been used in many applications to keep water from freezing. The NRC staff believes that the industry has ample experience with antifreeze that additional testing and validity is not necessary. Mixing of the antifreeze is a site-specific issue that will ensure that the proper amount of antifreeze is added to prevent the water from freezing at temperatures down to 0°F.

Comment I.16: One commenter recommended the addition of a note to Tables 4.4.20 and 4.4.21 of the SAR to provide clarification for the heat loads.

Response: The NRC disagrees with the comment. SAR Tables 4.4.20 and 4.4.21 refer to loading the MPC with uniformly aged fuel assemblies emitting heat at the design basis maximum rate. Section 4.4.2 identifies these assemblies as the limiting design basis fuel assemblies.

Comment I.17: One commenter stated that Holtec's use of a two-by-four block array to be equivalent to an infinite array assumed a center-to-center distance between casks of 18.6 feet. The commenter stated that this equivalency determination between an infinite array and a two-by-four array is invalid where the differences in cask spacing do not meet the 18.6-feet center-to-center assumption underlying the analysis. The commenter noted that the PSF facility design uses a 15-foot center-to-center distance. The commenter stated that any CoC issued for this cask system must address this shortcoming.

Response: The NRC disagrees with the comment. First, the PFS facility is outside the scope of this rulemaking activity. Second, as identified in Table 1.4.1 of the SAR, the analysis of a 2 by N array was performed using a pitch of 13.5 feet, not 18.6 feet. The 18.6-feet pitch is used for a square array. When calculating an equivalent hydraulic diameter for the square array and taking into account that the center-to-center spacing of neighboring casks between the pads is 38 feet, as described in Figure 1.4.1 of the SAR, the hydraulic diameters for the two cases (square array versus 2 by N array) is the same.

Comment I.18: One commenter stated that thermal interaction of casks through radiative heat transfer should be considered. The commenter also stated that the assumption that individual casks will not interfere with cooling air supply of each other may not be correct.

Response: The NRC agrees that neighboring casks can have an influence on each other. However, these influences have a second order impact on the results. The analyses performed by Holtec did credit radiation heat transfer between the neighboring casks. A bounding calculation was performed with an ambient temperature of 125 °F. That calculation accounted for heat reflected by the hot concrete pad and heat generated by neighboring casks. Although not required by the NRC staff's review of the SAR submittal, Holtec, in response to other inquiries, performed a sensitivity study to quantify the impact of neighboring casks and the impact of the sun heating the concrete pad.

The impact of increasing the spacing between casks by a factor of five in the radial direction resulted in a decrease in the peak cask surface temperature of 16 °F for an ambient temperature of 100 °F and 17 °F for an ambient temperature of 125 °F. The impact on peak clad temperature resulted in a decrease of 6 °F for an ambient temperature of 100 °F and a decrease of 8 °F for an ambient temperature of 125 °F. Because the peak clad temperature is on the order of 760+ °F, the impact of neighboring casks is minimal.

Comment I.19: One commenter stated that the temperature of the reflecting boundary should be taken as the temperature of the cask in interaction with the other casks and not the temperature of an isolated cask.

Response: The NRC agrees that one method for calculating the impact of neighboring casks is to model the neighboring casks in the array. Another acceptable method, that was used by the applicant, is to model the limiting (highest temperature) cask and assume that all the radiation it emits is reflected back. This analysis bounds the amount of radiation that neighboring casks can impose on the center cask. This bounding analysis is acceptable. As noted in the response to comment I.18, above, the impact of neighboring casks is minimal, given the significant margins between the allowable temperatures and the bounding calculated temperatures.

Comment I.20: One commenter stated that the Holtec model does not appear to take into account that the heating of the concrete pad is likely to diminish the “chimney effect” of the intake and outlet vents. The commenter stated that if Holtec had taken this effect into account, the calculated temperature would be higher in Revision 9 of the SAR.

Response: In a response to other inquires, Holtec performed calculations to quantify the effect of concrete pad heating on the cask performance. For the bounding 125 °F ambient temperature event, neglecting the heat reflected by the pad resulted in a reduction of cask surface temperature of 10 °F and a reduction in peak clad temperature of 6 °F. These temperature differences illustrate that the concrete pad has negligible impact on the cask.

Comment I.21: One commenter stated that ambient temperature should be defined due to the importance of the term. The commenter noted that ambient temperature is an important assumption in the thermal calculations and an important design element in the CoC. The commenter stated that the gross oversimplification of the concept of ambient temperature renders the Holtec thermal analysis completely useless. The commenter noted that Holtec assumes that the ambient temperature at the intake and outlet vents is the same; however, the temperature at ground level will be significantly higher than it will be some distance above due to the ground absorbing solar energy. The commenter stated that a desert may have a surface temperature of 180°F, much higher than the 80°F assumed by Holtec as an intake temperature. This would reduce the effective buoyancy and air velocity through the cooling ducts and result in a higher fuel cladding temperature.

Response: The NRC disagrees with the comment. The thermal response of a cask is very slow. This is due to the large mass of the system. An analogy can be reached by observing the buildings constructed in the desert. Massive concrete is used to maintain the indoor temperatures at reasonable conditions where air conditioners do not exist. The temperature in those regions fluctuates over each day. For these structures, an estimate of the average conditions can be assessed by assuming a bounding average daily temperature. Holtec used such a method. In addition, the method assumed the maximum solar heating specified in 10 CFR Part 71 averaged over a 24-hour period. Holtec used bounding assumptions approved in the NRC staff's SER.

Comment I.22: One commenter stated that NRC should have reviewed the inputs and outputs of the FLUENT calculation. The commenter also stated that the NRC should have conducted an independent analysis and validation of the thermal model employed by Holtec. The commenter stated that the HI-STAR analysis cannot be extrapolated to the HI-STORM cask because the casks are constructed of different materials, with different methods of heat dispersion. The commenter stated that the NRC performed a superficial review and had abdicated its role as independent regulator and should not issue a CoC for the HI-STORM 100 cask system because there is no lawful basis.

Response: The NRC disagrees with the comment. The NRC staff's review of the HI-STORM system was not superficial. This is clearly demonstrated by the NRC staff's requests for additional information, the applicant's many revisions to the SAR to address NRC staff concerns, and commitments made by the applicant as outlined in Appendix 12B of the SAR. The NRC staff does not perform independent confirmatory calculations for every analysis submitted in an application, nor does the NRC staff routinely review the inputs and outputs of the computer calculations without cause. Independent analyses that duplicate the extensive computer calculations performed by an applicant may, at times, be performed for various reasons. Some reasons include, but are not limited to, concern that a major error exists in the calculations; allegations that calculations were improperly performed; use of new modeling techniques not previously reviewed by the NRC staff; crediting heat transfer mechanisms not previously reviewed by the NRC staff; concern that the margin in a complex analysis is small; and concern that little conservatism exists in the modeling approach.

For the HI-STORM application, the NRC staff reviewed the basic assumptions used in the calculations, as identified in the SAR and in the NRC's requests for additional information. A detailed review of every number is not warranted. As for performing independent analysis and validation, the NRC staff was able to reach its safety findings without the need for such calculations. The need for these calculations is case specific, as addressed above. For HI-STORM, the applicant used computer codes that are employed by the NRC and have been found acceptable. The applicant demonstrated its knowledge of the code by benchmarking its methodology with a full-scale spent fuel cask instrumented with thermocouples, validating its thermal model and providing reasonable assurance that its analysts have good working knowledge of the code to perform the required calculations. The NRC staff's review of the applicant's methods and assumptions indicate ample margin and conservatism in the analyses.

The HI-STORM application review process was conducted under NRC policy and guidance, and as required by the regulations in 10 CFR Part 72. Regarding the reference to the HI-STAR analysis in Section 4.5.4 of the preliminary SER, the NRC staff intended to indicate that it was aware that Holtec's use of the FLUENT code had been previously found acceptable for the HI-STAR application. This reference was not intended to imply that the NRC staff relied on the HI-STAR calculations or the prior evaluation in its evaluation of the HI-STORM cask. Section 4.5.4 of the SER has been modified to clarify the description of the NRC staff's review. Also, to better illustrate the NRC staff's review of the applicant's submittal, Section 4 of the SER was supplemented with additional information.

Comment I.23: One comment indicated that the SER states that the ambient temperature under normal conditions must be less than 80°F. In addition, the commenter believed Holtec assumed that the ambient temperature at the inlet and outlet vents is the same and did not consider warming of the air by heat generated by neighboring casks and the concrete pad. The commenter stated that calculations indicated that a desert may have a surface temperature of 180°F and that the temperature 0.5 m above the ground would be 130°F.

Response: The NRC disagrees with the comment. The SER does not state that the ambient temperature under normal conditions must be less than 80°F. The applicant evaluated the cask conditions with an <u>annual</u> average ambient temperature of 80°F. The use of an annual average ambient temperature is used in conjunction with the method described in a

Pacific Northwest National Laboratory report PNL-6189. The method provides one acceptable means for obtaining reasonable assurance that the requirements in 10 CFR Part 72 will be met. These requirements include protecting the cladding from degradation that leads to gross ruptures and designing the storage system to allow ready retrieval of the spent fuel or high-level radioactive waste. With respect to the 180°F surface temperature in the desert, the SAR assumptions used in the 125°F ambient temperature calculation credits solar heating (also referred to as solar insolation) and heat generated by the casks. Holtec calculated a concrete pad surface temperature of 206°F (surrounding the concrete overpack), an ambient temperature just above the inlet vent of the overpack of 136°F, and a concrete temperature at the outlet vent of the overpack of 182°F. The NRC staff finds that the Holtec calculation adequately models the thermal responses of the cask and its environment.

J. Technical Specifications

Comment J.1: One commenter asked for clarification on the conditions for use and the TSs, and if they could be changed without an amendment.

Response: The conditions for cask use are specified in the CoC, and includes Appendix A (TSs) and Appendix B (Approved Contents and Design Features). These conditions cannot be changed without an amendment to the certificate.

Comment J.2: One commenter stated that the Use and Application section of the TSs is confusing and allows too much flexibility for completion times and frequencies, and that the TSs should be simple to understand and done on time.

Response: The NRC disagrees with the comment. The Section 1.0, “Use and Application” of the HI-STORM 100 TSs are modeled on the Improved Standard Technical Specifications (ISTS) for power reactors. The ISTS were developed as the result of extensive technical meetings and discussions between the NRC staff and the nuclear power industry in the early 1990s in an effort to improve clarity and consistency of the power reactor TSs and to make them easier for operators to use. The most likely users of the HI-STORM 100 Cask System TSs are power reactor licensees familiar with the format of the ISTS. The NRC staff believes that the format of the HI-STORM 100 TSs will make them easier for operators to use and will help to achieve consistency between power reactor and spent fuel dry cask storage TSs. The NRC staff disagrees that there is too much flexibility for completion times and frequency. The NRC staff believes that the specific wording of the TSs clearly specifies the allowable time to complete a required action and the frequency of any surveillance requirements.

Comment J.3: One commenter objected to the use of the term “TRANSPORT” in TS 3.2.2 and indicated that movement to the pad should be used because this CoC is for storage only.

Response: The NRC disagrees with the comment. The term TRANSPORT OPERATIONS is specifically defined in Section 1.1 of the Technical Specifications and includes all activities involved in moving a loaded overpack or transfer cask to and from the ISFSI pad.

Further clarification of the term is not warranted.

Comment J.4: One commenter stated that “Each” should be in large letters in LCO 3.2.2. The commenter also asked why all the removable contamination is not removed instead of setting a limit.

Response: The NRC disagrees with the comment. The capitalization of “each” is consistent with the format of the TSs. As discussed in the TS Bases, Section B.3.2.2, the contamination limits for the transfer cask are established from guidance in NRC IE Circular 87-01. The limits are based on minimum level of activity that can be routinely detected under a surface contamination control program using direct survey methods. These limits are consistent with levels that prevent the spread of contamination to clean areas and are significantly less than the levels associated with significant occupational exposure.

Comment J.5: One commenter stated that the dry run should be conducted in sequence and not an alternate step sequence as permitted by TS 5.2.

Response: The NRC disagrees with the comment. The dry runs are performed in discrete functional areas to demonstrate the ability to perform certain activities as anticipated. The order of performance of the functional areas, for the purpose of a dry run, is not directly pertinent to a demonstration of a user's capability. The operating procedures and technical specifications already control, as necessary, functional areas that must be performed sequentially for safe storage. The NRC staff considers it important to allow the cask user the necessary flexibility to allocate the appropriate resources and oversight to the performance of dry runs thatmay involve performing and concentrating on certain activities that would be out of sequence with a cask loading.

Comment J.6: One commenter asked why no lifting height limit was established for the vertical orientation of the transfer cask in TS 5.5 and stated that there should be a limit established.

Response: In the SAR, the design basis drop event analysis is based on the horizontal lifting height of 42 inches. Therefore, TS 5.5 only specifies the lifting height of the horizontal lifting limit. TS 5.5.c permits vertical lifting of loaded transfer cask to any height necessary to perform cask handling operations, including the MPC transfer. However, the lifts must be made with structures and components designed to prevent a drop and in accordance with the criteria specified in CoC, Appendix B, Section 3.5 and SAR Section 2.3.3.1. Therefore, a vertical lift height limit was not established.

Comment J.7: One commenter asked if the diamond-shaped water rod mentioned in note 10 of TS Table 2.1-3 had been completely analyzed.

Response: The shape (geometry) of water rods that are part of the fuel assembly, is considered in the evaluation.

Comment J.8: One commenter recommended deleting the words “For OVERPACKS with installed temperature monitoring equipment” at the beginning of the second option under SR 3.1.2.1 because users should have the option of using temporary equipment.

Response: The NRC agrees with the comment. Temperature monitoring, a surveillance option permitted in SR 3.1.2.1, could be conducted with either temporary or permanently installed equipment. The term "installed" could be interpreted as a requirement that the temperature monitoring equipment be permanently fixed. Therefore, the beginning of the second option under SR 3.1.2.1 has been reworded as follows: "For OVERPACKS with temperature monitoring equipment" (i.e., the word "installed" has been deleted).

Comment J.9: One commenter recommended several miscellaneous editorial changes to the appendices to the CoC.

Response: The NRC agrees with the comment. The appendices to the CoC have been revised to correct typographical errors and incorporate minor editorial changes.

Comment J.10: One commenter recommended that Items 5.2.f and 5.2.j in Section 5 of the TSs be revised to insert the phrase “(for which a mock-up may be used)” at the end of the items for consistency with SAR Section 12.2.2.

Response: The NRC agrees with the comment. Items 5.2.f and 5.2.j of the TSs have been revised to indicate that a mock-up may be used for those specific dry-run evolutions.

Comment J.11: One commenter recommended that item 5.5.c in Section 5 of the TSs be revised to replace the words “and MPC” with “or OVERPACK” because some utilities plan to implement an MPC transfer scheme that requires temporary lifting of the loaded OVERPACK above its lift height limit.

Response: The NRC disagrees with the comment. There are no evaluations, equipment design criteria, or other information in the SAR that support lifting a loaded overpack above its lift height limit.

Comment J.12: One commenter recommended revising the definitions of DAMAGED FUEL ASSEMBLY and PLANAR-AVERAGE INITIAL ENRICHMENT in Section 1 of Appendix B to the CoC to reflect the evolution of these terms and for consistency with those in the HI-STAR 100 CoC.

Response: The NRC agrees with this comment. The CoC, Appendix B, Section 1 has been revised to reflect the new definitions.

Comment J.13: One commenter recommended revising CoC, Appendix B, Section 3.4.6.c to replace the specified yield strength with the equivalent ASTM Grade specification.

Response: The NRC agrees with the comment in part. Storage pad design is a site-specific issue that needs to be addressed in the cask user’s 10 CFR 72.212 evaluation. CoC, Appendix B, Section 3.4.6.c lists the design parameters for the storage pads. It is not a list of components for fabrication. By using the specific ASTM Grade specification as recommended by the commenter, namely, ASTM A615, Grade 60, the designer of the pad will not have the flexibility to choose other reinforcing steels that could also be used (e.g. ASTM A616 or A617, Grade 60, etc.). To allow flexibility for the design and still ensure adequate reinforcement in the pad CoC, Appendix B, Section 3.4.6.c has been changed to state that reinforcement shall be 60 ksi yield strength ASTM material.

Comment J.14: One commenter recommended eliminating the requirement for impact limiters at the cask transfer facility contained in CoC, Appendix B, Item 3.5.2.2.

Response: The NRC staff assumes that the commenter's reference to Section 3.5.2.2 is a typographical error because the requirement for an impact limiter is in CoC, Appendix B, Item 3.5.2.1. The NRC agrees in part with the comment. The specific requirement for an impact limiter has been eliminated from CoC, Appendix B, Section 3.5.2.1.4. The NRC determined that this requirement is too restrictive because other methods may be available to prevent a canister breach in the event of a canister drop during transfer operations. Instead, Item 3.5.2.1.4 has been revised to require that the CTF be designed, constructed, and evaluated to ensure that if the MPC is dropped during an inter-cask transfer operation, its confinement boundary would not be breached.

However, the NRC disagrees with the underlying reason for the comment which is: Because a single failure proof crane (or equivalent) is required in the CTF, the design features to mitigate the consequence of a drop should not be necessary. The NRC staff acknowledges that the use of a single-failure proof crane precludes the possibility of a heavy load drop event. The requirement for a mitigating feature in the CTF design is a defense-in-depth measure that is consistent with the overall philosophy and approach of NUREG-0612. This philosophy encompasses an intent to prevent as well as to mitigate the consequences of postulated accidental load drops. The NRC staff notes that, even with a single-failure proof crane, NUREG-0612 still imposes a requirement for a safe load travel path “ to minimize the potential for heavy loads, if dropped, to impact irradiated fuel in the reactor vessel and in the spent fuel pool, or to impact safe shutdown equipment.” The NRC staff views the mitigating feature in the CTF as a defense-in-depth measure equivalent to the safe load path. Its function is to protect the MPC confinement boundary and the integrity of the spent fuel in the MPC in case of a postulated drop.

Comment J.15: One commenter recommended that CoC, Appendix B, Item 3.5.2.1.4 be clarified to indicate that the acceptance criterion for the impact limiter also applies to the use of mobile cranes.

Response: The NRC agrees with the comment. As discussed in the response to J.14, CoC, Appendix B, Item 3.5.2.1.4 has been revised to require that the CTF be designed and evaluated to ensure that if the MPC is dropped during an inter-cask transfer operation, its confinement boundary would not be breached. Section 3.5.2.1.4 has also been revised to specify that this requirement and acceptance criterion apply to both stationary and mobile cranes.

Comment J.16: One commenter recommended that CoC, Appendix B, Item 3.5.2.1.4 be revised to clarify the scope of drops that require evaluation in designing the impact limiter.

Response: The NRC agrees with the comment. CoC, Appendix B, Item 3.5.2.1.4 has been revised to clarify that the potential drops that require evaluation are those that may occur during inter-cask transfer operations.

Comment J.17: One commenter recommended that the TSs be removed from Appendix 12.A of the SAR because they are included in the appendices to the CoC.

Response: The NRC agrees with the comment. The TSs have been removed from the SAR.

K. Miscellaneous

Comment K.1: One commenter expressed approval that movement could be conducted at 0°F and above.

Response: No response is necessary.

Comment K.2: One commenter stated that the HI-TRAC transfer cask must be as safe as the HI-STORM overpack if it is to be used outside the reactor security fence.

Response: The NRC staff reviewed the HI-TRAC transfer cask and determined that, like the HI-STORM overpack, it will perform its intended safety functions under the design basis normal, off-normal, and accident events. It should be noted that the CoC authorizes use of the HI-TRAC only within the owner-controlled areas of a licensed power reactor.

Comment K.3: One commenter asked if the inner canister could be dropped through, if water could spill out of the overpack, and if the water helped to disperse the fuel particles.

Response: During a canister transfer operation, the transfer cask is placed on top of the storage overpack. The canister is then lowered through the bottom of the transfer cask into the overpack. It is unlikely that a canister drop would occur during this operation because the canister must be lifted with equipment (i.e., a single failure proof crane or equivalent) that are designed to prevent a drop. In addition, the overpack contains only traces of water that is part of the concrete material and the canister is dry during cask transfer operations.

Comment K.4: One commenter questioned the assumption that the HI-TRAC remains static because there are a number of man-made or natural causes that could put it in motion, drop, tipover, roll, etc.

Response: The HI-TRAC is required to be independently secured on top of the overpack during the transfer of the MPC.

Comment K.5: One commenter asked when the measuring equipment (for checking tolerances) is calibrated.

Response: The timing of calibration at the fabricator's facility is beyond the scope of this rule. However, the implemented QA program at the fabricator's facility provides reasonable assurance that the measuring equipment for checking tolerances of fabrication will be appropriately calibrated.

Comment K.6: One commenter asked if the restraint of 11 inches in vertical height for overpack handling would actually preclude a corner drop situation. The commenter asked how a corner drop could be initiated, such as a defective trunnion or lifting lug, etc.

Response: The 11-inch restriction on lifting height for the overpack was calculated to ensure that deceleration loading to the loaded MPC would not exceed the design criteria for the confinement boundary of the basket. A tipover of the overpack cannot occur if the baseplate is limited to 11 inches above a receiving surface.

Comment K.7: One commenter asked what happens to the inside of the cask during a horizontal drop of 50 inches.

Response: The effect of a 50-inch horizontal drop of a cask was not evaluated because the horizontal lifting height limit for the transfer cask is 42 inches. The 50-inch carry height specified in the SER was a typographical error and has been corrected to 42 inches. There is no effect on the confinement function of the MPC as a result of a horizontal drop of 42 inches. The structural evaluation shows that all stresses are within allowable values and that the confinement boundary integrity of the MPC is not impaired.

Comment K.8: One commenter requested that the SER define what is meant by cladding oxide thickness on page 4-1 of the SER.

Response: The NRC disagrees with the comment. Cladding oxide thickness is a measure of corrosion at the clad surface. As water interacts with the zirconium clad, the zirconium can interact with the oxygen molecules to create zirconium oxide (ZrO_2). The terminology is commonly used in the spent fuel storage arena and a definition in the SER is not necessary.

Comment K.9: One commenter asked why the internal rod pressure is assumed to remain the same. The commenter asked how the gas behaves in a dry cask and if it can leak from pinhole leaks and hairline cracks over the storage period. The commenter further asked how the lower pressure in the rods affects the analysis and heat transfer.

Response: The internal rod pressure is derived from the initial gas inserted during fabrication plus the fission product gases that develop during power production within the reactor core. In a closed system (e.g., the fuel pin), the pressure is a function of the gases in the fuel rod and the average temperature of the gas. As the decay heat decreases with time, so does the temperature and the pressure. Therefore, the rod temperature does not remain the same. This is similar to inflating a balloon with hot air and placing the balloon in the refrigerator. As the gases cool, the pressure decreases, as is implied by the smaller diameter of the balloon. The lower pressure reduces the stress on the cladding and permits a higher allowable temperature limit. If the rod experiences a pinhole leak or a hairline crack, the gases inside the rod will mix with the helium gas in the cask and reduce the internal pressure within the rod.

Reduction of the internal fuel rod pressure results in added assurance that the cladding will remain stable because the internal pressure will have equilibrated with that of the cask. The gases from the fuel pin mix with the gases in the cask and decreases the thermal conductivity of the helium, while at the same time increasing the density of the gas. The analyses for accident conditions incorporate the impact of reduced conductivity of the helium gases. This impact is reduced when crediting cooling that results from natural circulation of the gases inside the cask. The use of a maximum allowable temperature limit provides assurance that the fuel pins will remain intact throughout the storage period. For conservatism, the applicant assumed that 1 percent of the cladding experiences a leak under normal conditions, a 10-percent leak under off-normal conditions, and a 100-percent leak under accident conditions.

Comment K.10: One commenter asked what cask design was tested at INEEL (page 4-3 of the SER).

Response: Several full scale cask designs were tested at the Idaho National Engineering and Environmental Laboratory. The cask used by Holtec to validate the FLUENT computer code was the TN-24P. The heat output of the cask was 23 kW. The NRC staff found the FLUENT computer code acceptable for calculating the thermal response of a spent fuel cask.

Comment K.11: One commenter expressed concern over water and debris going into cracks on the pad and then freezing and thawing causing concrete upheaval and subsequent cask tipover.

Response: Issues related to cask storage pad will be addressed in the cask user's evaluation under 10 CFR 72.212 and is beyond the scope of this rule.

Comment K.12: One commenter asked how moisture and pollution in the air could affect the casks and pad over time and if the pad would ever need to be replaced.

Response: The cask can withstand the ambient environmental conditions over its 20-year license period with no significant degradation. The adequacy of the pad must be addressed by the cask users in their 10 CFR 72.212 evaluation and is beyond the scope of this rule. Cask users are responsible for inspecting and maintaining the pad. With appropriate maintenance, air pollution or moisture would not cause significant degradation to the pad.

Comment K.13: One commenter asked if both the helium and fission gases created the pressure inside the rods and for an explanation of the fission gases. The commenter also asked why only 30 percent of the fission product gas was assumed to be released instead of 100 percent because over time 100 percent would likely leak out.

Response: Fission gases are byproducts of uranium splitting in a reactor. These include gases such as hydrogen, krypton, and iodine. The gases are contained inside the fuel rod. Data have shown that a conservative estimate of 30 percent of the gases generated inside the fuel pellet can escape to the gap that exists between the fuel pellet and the cladding. This is a conservatively large number used for calculating dosage. Experimental data has shown this number to be significantly less. The rest of the gases are trapped inside the fuel pellet. Therefore, assuming that 100 percent of the gases are released from the fuel pellet is not realistic. Helium gas is added to the MPC to keep the environment inside the cask inert so it does not promote corrosion and to help cool the fuel by transferring heat from the fuel rods to the wall of the cask. The impact of helium gas on the pressure within a fuel rod is not as significant as the temperature of the gas within the fuel rod.

Comment K.14: One commenter asked what is in the water of the water jacket and if the water could affect the carbon steel channels or get into the pool through a weld crack or leak and affect the pool. The commenter also asked how hot the water and the lead get, and if the water could cause pressure buildup in the channels.

Response: The water used in the water jacket is demineralized water as is used in the loading pool, but without boron addition because the boron is unnecessary for loading/unloading operations. Carbon steel corrodes very slowly in demineralized water; thus, its effect may be ignored for the durations experienced in loading operations. If the cask is to be loaded in cold weather, antifreeze may be added to the jacket water. Antifreeze contains an inhibitor to prevent corrosion. There would be no significant effect if the jacket water or water with antifreeze leaked into the pool. With regard to the water and lead temperatures and pressure buildup in the water jackets, see the response to comment I.2.

Comment K.15: One commenter asked if there was a recent study on cladding degradation from creep cavitation.

Response: Studies on cladding degradation were performed several years ago. These studies led to the development of analytic methods to calculate the maximum allowable peak clad temperature limits. A report developed by the Pacific Northwest National Laboratory (PNL) in May 1987, PNL-6189, “Recommended Temperature Limits for Dry Storage of Spent Light Water Reactor Zircaloy-Clad Fuel Rods in Inert Gas” provides an acceptable method for assessing cladding temperature limits.

Comment K.16: One commenter stated that the 100-ton transfer cask should not be included in the certification because it is site-specific and not made the same as the 125-ton cask.

Response: NRC disagrees with the comment. The 100-ton and 125-ton transfer cask designs have been evaluated and found to meet the regulatory requirements of 10 CFR Part 72. The 100-ton transfer cask design is not considered site-specific and is approved under this rule for use by any general licensee as part of the HI-STROM 100 system as described in the SAR. Section 2.0.3 of the SAR provides guidance regarding site-specific ALARA objectives that should be considered by each user when using either transfer cask design.

Comment K.17: One commenter asked what does reasonable assurance mean in Section 5.1.2 of the SER regarding acceptability of the shielding design criteria.

Response: The finding in Section 5.1.2 is intended to mean that the NRC staff believes that the dose rate criteria presented in the SAR are acceptable values and that a cask system operating at these values can meet the applicable radiological requirements of 10 CFR Parts 20 and 72. The SAR subsequently demonstrates that the dose rates calculated for the HI-STORM system meet the regulatory requirements.

Comment K.18: One commenter asked if the MOX (mixed oxide) fuel was covered by the sabotage report. The commenter asked if MOX fuel had been tested and verified to be safe for this design. The commenter further questioned how the NRC could include MOX fuel in the

SER evaluation and stated that storage of MOX fuel should not be allowed by the certification. The commenter also asked how we know that storage of MOX fuel will work as expected because it has not yet been tested in Canada.

Response: The sabotage report is beyond the scope of this rule. However, the design and physical characteristics of a MOX fuel assembly are very similar to those of a uranium fuel assembly. The primary difference is the fuel pellet constituents and its effects on the radiological source term. Testing of MOX fuel is also beyond the scope of this rule.

The HI-STORM design was evaluated for storage of the MOX fuel assemblies listed in the Appendix B to the CoC using computer codes and models. In lieu of testing, the NRC finds analytic conclusions that are based on sound engineering methods and practices to be acceptable. Testing is only required if the analytic methods have not been validated or assured to be appropriate and/or conservative. The NRC staff reviewed the applicant's analyses and found them acceptable. The basis of the safety review and findings are identified in the SER and the CoC.

Comment K.19: One commenter asked if all the analysis was based on the 100-ton transfer cask or did HI-STORM 100 refer to something else.

Response: The shielding analysis presented in the SAR evaluated both the 100-ton and 125-ton transfer cask designs as part of the HI-STORM 100 cask system.

Comment K.20: One commenter asked how the NRC could base its evaluation on historical statements when reference documents indicate Inconel impurity may be higher than 1000 ppm. The commenter further asked what the historical statements were and how we know if the statements are valid.

Response: The applicant's analysis of cobalt impurities are discussed in Section 5.2.1 and 5.2.3 of the SER. The applicant showed that the cobalt impurity value of 1000 ppm assumed in the shielding analyses was appropriate based on industry data and analyses of post-irradiation cooling of older fuel types that may have had higher cobalt impurities for the HI-STORM 100 cask system. As discussed in Section 5.2.1 of the SAR, historical statements included industry data gathered by the applicant from utilities and vendors.

Cobalt impurities were not necessarily controlled for older fuel designs. However, the applicant showed that the post-irradiation cooling time that is inherent to these older fuel types significantly reduces the HI-STORM 100 dose rates. Therefore, the effects of higher impurities are mitigated. Based on historical knowledge of recent cobalt reduction programs, the decay effects on older fuel, and its own independent evaluations, NRC has reasonable assurance that the historical statements referenced in the application are used appropriately for the HI-STORM 100. Furthermore, each cask user will operate the HI-STORM 100 under a 10 CFR Part 20 radiological protection program and will be required to verify dose rates that are specified in the

TSs. This defense-in-depth approach will mitigate potential hardware activation anomalies and ensure compliance with radiological requirements.

Comment K.21: One commenter asked if the steel transport overpacks could be reused, how contaminated the overpacks would be after use, the number of times an overpack could be reused, and if they would be checked after each use.

Response: This comment that concerns the HI-STAR steel transport overpack, is beyond the scope of this rule on the Holtec HI-STORM 100 cask system.

Comment K.22: One commenter was pleased that the NRC had evaluated uneven flooding.

Response: No response is necessary.

Comment K.23: One commenter asked about the chance of one of the screens being damaged or loosened in unloading and the debris floating out with the cooling water into the pool.

Response: The damaged fuel container that is placed in the MPC is stainless steel and is designed to retain damaged fuel and debris in a safe configuration under all normal, off-normal, and accident conditions. The damaged fuel container also provides a means to safely handle the damaged fuel and debris during loading and unloading. It is not considered credible that the screens will fall off or fail. However if a screen failed, there would be no release of radioactive material during storage since the MPC is sealed. Consideration of loose debris during unloading is addressed in SAR Section 8.3 which outlines the MPC unloading operations in a spent fuel pool and specifically considers loose debris in the MPC. Additionally, the spent fuel pool filtration system would capture any debris that remained in the pool.

Comment K.24: One commenter asked why the volume of water removed from the cask is recorded and why this is not done for other cask designs.

Response: The purpose of recording the volume of water removed from the canister is to identify the open volume in the canister. This open volume is used to calculate the amount of helium to be added to the cask following vacuum drying. The procedure and equation used for this procedure is discussed on page 8.1-21 in the HI-STORM SAR. The comment concerning other cask designs is beyond the scope of this rule.

Comment K.25: One commenter stated that a detailed procedure on mitigating the possibility of fuel crud particulates dispersal should be included in the documents and that the procedure should not be site-specific.

Response: NRC disagrees with the comment. The generic unloading procedures for the HI-STORM 100 system are designed to mitigate crud dispersal. However, each cask user will need to develop detailed unloading procedures that incorporate the ALARA objectives of its site-specific radiation protection program. NRC expects the cask user to consider the specific characteristics of its fuel, including crud phenomena, when developing these procedures.

Comment K.26: One commenter asked how the utilities are required to document that they will not lift the overpack any higher than 11 inches and that the receiving surface hardness does not exceed that analyzed in the SAR. The commenter stated that the criteria should be clarified and which surface should be indicated.

Response: The receiving surface is the top of the storage pad as clearly stated in Sections 3.4.2 and 11.2.3.2 of the SER and described in Section 3.4.10 of the SAR. Users of the HI-STORM 100 system are required to meet Appendices A and B of the CoC that list the design parameters for surface hardness and the restriction for lifting height. Furthermore, the cask users are required to develop detailed written operating procedures. The restriction on lifting height must be incorporated into the operating procedures subject to NRC inspection.

Comment K.27: One commenter stated that Condition 8 should remain in the CoC.

Response: The NRC agrees with the comment. Condition 8 has not been removed from the CoC. Under Condition 8, Certificate holders who wish to make changes to the CoC, including Appendices A and B, must submit an application for amendment of the Certificate.

Comment K.28: One commenter asked how upending/downending of the transfer cask affected the water in the neutron shield, how the licensee knows the shield is full, what happens to the contents of the cask when the position changes, what are the stresses and pressures , and if the debris in damaged fuel containers goes through the screen.

Response: The structural, shielding, and confinement functions of the transfer cask are not affected during movement of the cask. The neutron shield will normally be filled through the drain valve at the bottom of the water jacket and is considered full as water exits the vent port at the top of the water jacket. The vent plug is then installed to retain the water in the jacket. During the upending and downending of the transfer cask, water remains within the neutron shield and fuel debris remains within the confinement boundary of the MPC. The structural evaluation in the SAR showed all the stresses and pressures to remain within allowable values.

Comment K.29: One commenter stated that exceptions to the codes should not be allowed and that the NRC should demand full code requirements.

Response: The NRC disagrees with this comment. Exceptions (alternatives) to the ASME Code specifications may be granted by the NRC staff on a case-by-case basis. During the NRC staff review of a proposed alternative, the applicant must demonstrate that the proposed alternative to the Code satisfies one of the following criteria: (1) the alternative provides an acceptable level of quality and safety, or, (2) compliance with a specific Code requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality or safety.

Comment K.30: One commenter stated that videos should not be used as a permanent record. The commenter stated that black and white photos and negatives should be used and that the negatives should be kept in museum qualified storage. The commenter asked what method is best to document weld integrity and how the records are stored. The NRC should have specific criteria for record keeping requirements.

Response: The NRC disagrees with the comment. The NRC's regulations do not explicitly require specific criteria for record keeping to document weld integrity by the applicant. A permanent record of completed welds will be made using video, photographic, or other means that can provide a retrievable record of weld integrity. As per accepted industry practice, the record is typically in color format, in order to capture the red dye typically used for PT examinations. The general licensee's QA program will specify the types of records and how the records are to be stored.

Comment K.31: One commenter stated that even if the overpack baseplates, shell, pedestal shell, and radial plates have large margins of safety in the design, they should still be examined to code.

Response: Holtec has committed to inspect the welds of the overpack baseplate to the shell, pedestal shell, and radial plates under ASME Code Section V, Article 9. Weld inspection acceptance criteria meet the requirements in ASME Section III, Subsection NF-5360.

Comment K.32: One commenter asked why a mobile lifting device is used and why it is not required to meet the requirements of NUREG-0612, Section 5.1.6(2) for new cranes. If a new crane is necessary to meet the requirements, the utilities should get one and not be allowed to lower requirements.

Response: A mobile lifting device is an alternative option to a stationary lifting device that may be used in a CTF. The decision to use either a mobile or stationary lifting device would be made by the cask users and would be based on their plant's site-specific needs. NUREG-0612, Section 5.1.6(2) specifies that new cranes should be designed to meet NUREG-

0554, “Single-Failure-Proof Cranes for Nuclear Power Plants.” These requirements are not applicable to mobile lifting devices which are not single-failure-proof; therefore, mobile lifting devices are exempted from this particular requirement in NUREG-0612. To ensure that the mobile lifting device has the equivalent level of safety as a single-failure-proof crane, additional conditions in CoC, Appendix B, Sections 3.5.2.2.1, 3.5.2.2.2, and 3.5.2.2.4 were imposed.

Comment K.33: One commenter stated that a discussion on the cask transfer facility should be included in the SER, and that the public should not have to read the SAR to understand the generic design. The commenter requested that this part of the cask transfer facility be resubmitted with a complete clear design with specific criteria.

Response: The NRC disagrees with the comment. SER Section 1.1 discusses the CTF in a level of detail appropriate for an SER. The detailed design and operating criteria for the CTF are given in SAR Section 2.3.3.1. This satisfies 10 CFR 72.24, which requires that the SAR contain information on structures, systems, and components important to safety in sufficient detail for the NRC staff to make its regulatory finding. Repeating this information in the SER is not necessary. The NRC disagrees that cask transfer facility should be resubmitted with a complete clear design with specific criteria. The specific criteria for the CTF are already given in CoC, Appendix B, Section 3.4, and SAR Section 2.3.3.1. As discussed in the response to E.30, NRC found it unnecessary to approve a specific CTF design.

Comment K.34: One commenter recommended that Section 3.5.7 of the SER be revised to reflect that transport of the HI-TRAC transfer cask in the vertical orientation is permitted. The comment also recommended that “50 inches” be changed to “42 inches” to be consistent with TS Table 5-1.

Response: The NRC agrees with the comment. The SER has been modified to reflect that transport of the HI-TRAC transfer cask in the vertical orientation is permitted. The horizontal lifting height per TS Table 5-1 will be corrected to 42 inches to correct the typographical error.

Comment K.35: One commenter recommended that Section 9.1.2.2.b of the SER be revised to delete "(either to the fuel pool or the site licensee's off-gas system)" because users may or may not have these systems at their plants.

Response: The NRC agrees with the comment. It is up to the cask users to develop the specific procedures for venting the MPC and to determine the appropriate location under their plant's waste gas handling system design and radiation protection program. Section 9.1.2.2.b of the SER has been modified as recommended.

Summary of Final Revisions

As a result of the NRC staff's response to public comments, or to rectify issues identified during the comment period, TSs 5.2.f and 5.2.j have been modified (see comment J.10). The NRC staff has also updated the CoC, including Appendix B, and has removed the bases section from the TSs attached to the CoC to ensure consistency with NRC's format and content. The NRC staff has also modified its SER. In addition, the NRC staff has modified the rule language by changing the word "Certification" to "Certificate" to clarify that it is actually the Certificate that expires.

Agreement State Compatibility

Under the “Policy Statement on Adequacy and Compatibility of Agreement State Programs” approved by the Commission on June 30, 1997, and published in the Federal Register on September 3, 1997 (62 FR 46517), this rule is classified as compatibility Category “NRC.” Compatibility is not required for Category “NRC” regulations. The NRC program elements in this category are those that relate directly to areas of regulation reserved to the NRC by the Atomic Energy Act of 1954, as amended (AEA), or the provisions of Title 10 of the Code of Federal Regulations. Although an Agreement State may not adopt program elements reserved to NRC, it may wish to inform its licensees of certain requirements via a mechanism that is consistent with the particular State’s administrative procedure laws, but does not confer regulatory authority on the State.

Finding of No Significant Environmental Impact: Availability

Under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in Subpart A of 10 CFR Part 51, the NRC has determined that this rule is not a major Federal action significantly affecting the quality of the human environment and therefore, an environmental impact statement is not required. This final rule adds an additional cask to the list of approved spent fuel storage casks that power reactor licensees can use to store spent fuel at reactor sites without additional site-specific approvals from the Commission. The environmental assessment and finding of no significant impact on which this determination is based are available for inspection at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC. Single copies of the environmental assessment and finding of no significant impact are available from Merri Horn, Office of Nuclear Material

Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 415-8126, e-mail mlh1@nrc.gov.

Paperwork Reduction Act Statement

This final rule does not contain a new or amended information collection requirement subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). Existing requirements were approved by the Office of Management and Budget, approval number 3150-0132.

Public Protection Notification

If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

Voluntary Consensus Standards

The National Technology Transfer Act of 1995 (Pub. L. 104-113) requires that Federal agencies use technical standards that are developed or adopted by voluntary consensus standards bodies unless the use of such a standard is inconsistent with applicable law or otherwise impractical. In this final rule, the NRC is adding the Holtec International HI-STORM 100 cask system to the list of NRC-approved cask systems for spent fuel storage in 10 CFR 72.214. This action does not constitute the establishment of a standard that establishes generally-applicable requirements.

Regulatory Analysis

On July 18, 1990 (55 FR 29181), the Commission issued an amendment to 10 CFR Part 72. The amendment provided for the storage of spent nuclear fuel in cask systems with designs approved by the NRC under a general license. Any nuclear power reactor licensee can use cask systems with designs approved by the NRC to store spent nuclear fuel if it notifies the NRC in advance, the spent fuel is stored under the conditions specified in the cask's CoC, and the conditions of the general license are met. In that rule, four spent fuel storage casks were approved for use at reactor sites and were listed in 10 CFR 72.214. That rule envisioned that storage casks certified in the future could be routinely added to the listing in 10 CFR 72.214 through the rulemaking process. Procedures and criteria for obtaining NRC approval of new spent fuel storage cask designs were provided in 10 CFR Part 72, Subpart L.

The alternative to this action is to withhold approval of this new design and issue a site-specific license to each utility that proposes to use the casks. This alternative would cost both the NRC and utilities more time and money for each site-specific license. Conducting site-specific reviews would ignore the procedures and criteria currently in place for the addition of new cask designs that can be used under a general license, and would be in conflict with NWPA direction to the Commission to approve technologies for the use of spent fuel storage at the sites of civilian nuclear power reactors without, to the maximum extent practicable, the need for additional site reviews. This alternative also would tend to exclude new vendors from the business market without cause and would arbitrarily limit the choice of cask designs available to power reactor licensees. This final rule will eliminate the above problems and is consistent with previous Commission actions. Further, the rule will have no adverse effect on public health and safety.

The benefit of this rule to nuclear power reactor licensees is to make available a greater choice of spent fuel storage cask designs that can be used under a general license. The new cask vendors with casks to be listed in 10 CFR 72.214 benefit by having to obtain NRC certificates only once for a design that can then be used by more than one power reactor licensee. The NRC also benefits because it will need to certify a cask design only once for use by multiple licensees. Casks approved through rulemaking are to be suitable for use under a range of environmental conditions sufficiently broad to encompass multiple nuclear power plants in the United States without the need for further site-specific approval by NRC. Vendors with cask designs already listed may be adversely impacted because power reactor licensees may choose a newly listed design over an existing one. However, the NRC is required by its regulations and NWPA direction to certify and list approved casks. This rule has no significant identifiable impact or benefit on other Government agencies.

Based on the above discussion of the benefits and impacts of the alternatives, the NRC concludes that the requirements of the final rule are commensurate with the Commission's responsibilities for public health and safety and the common defense and security. No other available alternative is believed to be as satisfactory, and thus, this action is recommended.

Small Business Regulatory Enforcement Fairness Act

In accordance with the Small Business Regulatory Enforcement Fairness Act of 1996, the NRC has determined that this action is not a major rule and has verified this determination with the Office of Information and Regulatory Affairs, Office of Management and Budget.

Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act of 1980 (5 U.S.C. 605(b)), the Commission certifies that this rule will not, if promulgated, have a significant economic impact on a substantial number of small entities. This rule affects only the licensing and operation of nuclear power plants, independent spent fuel storage facilities, and Holtec International. The companies that own these plants do not fall within the scope of the definition of “small entities" set forth in the Regulatory Flexibility Act or the Small Business Size Standards set out in regulations issued by the Small Business Administration at 13 CFR Part 121.

Backfit Analysis

The NRC has determined that the backfit rule (10 CFR 50.109 or 10 CFR 72.62) does not apply to this rule because this amendment does not involve any provisions that would impose backfits as defined in the backfit rule. Therefore, a backfit analysis is not required.

List of Subjects in 10 CFR Part 72

Criminal penalties, Manpower training programs, Nuclear materials, Occupational safety and health, Reporting and recordkeeping requirements, Security measures, Spent fuel.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended; the Energy Reorganization Act of 1974, as amended; and 5 U.S.C. 553; the NRC is proposing to adopt the following amendments to 10 CFR part 72.

PART 72--LICENSING REQUIREMENTS FOR THE INDEPENDENT STORAGE OF SPENT NUCLEAR FUEL AND HIGH-LEVEL RADIOACTIVE WASTE

1. The authority citation for Part 72 continues to read as follows:

AUTHORITY: Secs. 51, 53, 57, 62, 63, 65, 69, 81, 161, 182, 183, 184, 186, 187, 189, 68 Stat. 929, 930, 932, 933, 934, 935, 948, 953, 954, 955, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2071, 2073, 2077, 2092, 2093, 2095, 2099, 2111, 2201, 2232, 2233, 2234, 2236, 2237, 2238, 2282); sec. 274, Pub. L. 86-373, 73 Stat. 688, as amended (42 U.S.C. 2021); sec. 201, as amended, 202, 206, 88 Stat. 1242, as amended, 1244, 1246 (42 U.S.C. 5841, 5842, 5846); Pub. L. 95-601, sec. 10, 92 Stat. 2951 as amended by Pub. L. 10d - 48b, sec. 7902, 10b Stat. 31b3 (42 U.S.C. 5851); sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332); secs. 131, 132, 133, 135, 137, 141, Pub. L. 97-425, 96 Stat. 2229, 2230, 2232, 2241, sec. 148, Pub. L. 100-203, 101 Stat. 1330-235 (42 U.S.C. 10151, 10152, 10153, 10155, 10157, 10161, 10168).

Section 72.44(g) also issued under secs. 142(b) and 148(c), (d), Pub. L. 100-203, 101

Stat. 1330-232, 1330-236 (42 U.S.C. 10162(b), 10168(c),(d)). Section 72.46 also issued under sec. 189, 68 Stat. 955 (42 U.S.C. 2239); sec. 134, Pub. L. 97-425, 96 Stat. 2230 (42 U.S.C. 10154). Section 72.96(d) also issued under sec. 145(g), Pub. L. 100-203, 101 Stat. 1330-235 (42 U.S.C. 10165(g)). Subpart J also issued under secs. 2(2), 2(15), 2(19), 117(a), 141(h), Pub. L. 97-425, 96 Stat. 2202, 2203, 2204, 2222, 2244, (42 U.S.C. 10101, 10137(a), 10161(h)). Subparts K and L are also issued under sec. 133, 98 Stat. 2230 (42 U.S.C. 10153) and sec. 218(a), 96 Stat. 2252 (42 U.S.C. 10198).

2. In Section 72.214, Certificate of Compliance 1014 is added to read as follows:

§ 72.214 List of approved spent fuel storage casks.

* * * * *

Certificate Number: 1014

SAR Submitted by: Holtec International

SAR Title: Final Safety Analysis Report for the HI-STORM 100 Cask System

Docket Number: 72-1014

Certificate Expiration Date: (**Insert date 20 years after final rule effective date**)

Model Number: HI-STORM 100

Dated at Rockville, Maryland, this 12th day of April ,2000.

For the Nuclear Regulatory Commission.

/RA/

Frank J. Miraglia, Jr.,
Acting Executive Director for Operations.

Certificate Expiration Date: (**Insert date 20 years after final rule effective date**)

Model Number: HI-STORM 100

Dated at Rockville, Maryland, this 12th day of April ,2000.

For the Nuclear Regulatory Commission.

/RA/

Frank J. Miraglia, Jr.,
Acting Executive Director for Operations.

DOC NAME: ADAMS Accession No. ML003685148 *see previous concurrence
"C" = copy without attachment/enclosure, "B" = copy with attachment/enclosure, "N" = No copy

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