ATTACHMENT 1

Federal Register Notice

of Final Rulemaking

[7590-01-P]

NUCLEAR REGULATORY COMMISSION

10 CFR Part 72

RIN 3150-AG17

List of Approved Spent Fuel Storage Casks: (HI-STAR 100) Addition

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is amending its regulations to add the Holtec International HI-STAR 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 (30 days from the date of publication in the Federal Register).

FOR FURTHER INFORMATION CONTACT: Stan Turel, telephone (301) 415-6234, e-mail spt@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 International HI-STAR 100 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 together with the Safety Analysis Report (SAR) entitled "HI-STAR 100 Cask System Topical Safety Analysis Report (SAR), Revision 8." 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 International HI-STAR 100 cask system. The NRC published a proposed rule in the Federal Register (64 FR 1542; January 11, 1999) to add the HI-STAR 100 cask system to the listing in 10 CFR 72.214. The comment period ended on March 29, 1999. Nine comment letters were received on the proposed rule.

Based on NRC review and analysis of public comments, the 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 International HI-STAR 100 cask system. The staff has also modified its preliminary SER and has revised the title of the SAR in the listing of this cask design in 10 CFR 72.214.

The title of the SAR has been revised to delete the revision number so that in the final rule the title of the SAR is "HI-STAR 100 Cask System Topical Safety Analysis Report." This revision conforms the title to the requirements of new 10 CFR 72.248, recently approved by the Commission.

The proposed CoC has been revised to clarify the requirements for making changes to the CoC by specifying that the CoC holder must submit an application for an amendment to the certificate if a change to the CoC, including its appendices, is desired. This revision conforms the change process to that specified in 10 CFR 72.48, as recently approved by the Commission. The CoC has also been revised to delete the proposed exemption from the requirements of 10 CFR 72.124(b) because a recent amendment of this regulation makes the exemption unnecessary (64 FR 33178; June 22, 1999). In addition, other minor, nontechnical, changes have been made to CoC 1008 to ensure consistency with NRC's new standard format

Comments on Direct Final Rule

As part of the proposed rule, the NRC staff requested public comment on the use of a direct final rulemaking process for future amendments to the list of approved spent fuel storage casks in 10 CFR 72.214. The direct final rulemaking process is used by Federal agencies, including the Environmental Protection Agency (EPA) and the NRC, to expedite rulemaking where the agency believes that the rule is noncontroversial and significant adverse comments will not be received. Use of this technique in appropriate circumstances has been endorsed by the Administrative Conference of the United States (60 FR 43110; August 18, 1995). Under the direct final rulemaking procedure, the NRC would publish the proposed amendment to the 10 CFR 72.214 list as both a proposed and a final rule in the Federal Register simultaneously. A direct final rule normally becomes effective 75 days after publication in the Federal Register unless the NRC receives significant adverse comments on the direct final rule within 30 days after publication. If significant adverse comments are received, the NRC publishes a document that withdraws the direct final rule. The NRC then addresses the comments received as comments on the proposed rule and subsequently issues a final rule.

One commenter supported use of the direct final rule process for future revisions to the listing in 10 CFR 72.214, stating that it was imperative that the regulatory process be streamlined when there is no adverse safety concern. Two commenters were opposed to use of a direct final rule process stating that a direct final rule would diminish the public role in commenting on the approval of spent nuclear fuel casks and thereby the public's ability to affect the outcome of rulemaking procedures. One of these commenters believed that, given past problems with the casks, future approval should be subject to adequate and rigorous public scrutiny. Those opposed also believed that 30 days (as would be allowed in a direct final rule process) is not sufficient time to prepare comments that may be significantly adverse so as to cause the NRC to withdraw the published final rule. The two commenters did not believe that an addition to or revision of the listing is likely to be either noncontroversial or routine as evidenced by the number of comments they had on the Holtec HI-STAR 100 proposed rule.

A number of significant adverse comments were received on the NRC's proposed listing of the Holtec International HI-STAR 100 cask system which are described in subsequent sections of this notice. Therefore, it does not appear that the direct final rule approach can be implemented at this time for additions to the cask listing. The NRC will reassess this issue in the future after experience with more new listings to 10 CFR 72.214 has been gained. However, with respect to amendments to existing CoCs, the NRC anticipates that, except in unusual cases, the direct final rulemaking process can be used because the cask design and analysis will have gone through the public comment process for the initial CoC listing and the revision will be limited to the subject of the amendment. Unless the NRC has reason to believe that a particular amendment will be controversial, the NRC plans to use a direct final rule for amendments to the cask systems in the 10 CFR 72.214 listing. The NRC disagrees that use of the direct final rulemaking procedure will limit the public's ability to affect the outcome of the rulemaking. Receipt of a significant adverse comment will cause the direct final rule to be withdrawn and the comment to be considered as though received in response to a proposed rule. Further, the NRC believes that 30 days is a sufficient amount of time in which to submit a comment on an amendment to the CoC for a listed cask since most issues related to the cask design will have been resolved in the rulemaking conducted to place the design on the 10 CFR 72.214 list.

Comments on the Holtec International HI-STAR 100 Cask System

The comments and responses have been grouped into five areas: general comments, cladding integrity, health impacts, sabotage events, thermal requirements, and miscellaneous items. Several of the commenters provided specific comments on the draft CoC, the NRC staff's preliminary SER, the TSs, and the applicant's Topical SAR. Some of the editorial comments have been grouped as well as some of the comments on the drawings in the SAR. To the extent possible, all of the comments on a particular subject are grouped together. The listing of the Holtec International HI-STAR 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:

General Comments

Comment No. 1: One commenter asked a number of questions about the process for review and approval of spent fuel storage cask designs, and suggested changes to the process.

Response: The NRC finds these comments to be beyond the scope of the current rulemaking which is focused solely on whether to place a particular cask design, the Holtec International HI-STAR 100 cask system, on the 10 CFR 72.214 list.

Comment No. 2: One commenter stated that the cask should be built and tested before use at reactors, including the loading and unloading procedures. The commenter objected to the use of computer modeling and analysis.

Response: The NRC disagrees with the comment. The HI-STAR 100 Storage Cask System Design has been reviewed by the NRC. The basis of the safety review and findings are clearly identified in the SER and CoC. Testing is normally required when the analytic methods have not been validated or assured to be appropriate and/or conservative. In place of testing, the NRC staff finds acceptable analytic conclusions that are based on sound engineering methods and practices. NRC accepts the use of computer modeling codes to analyze cask performance. The appropriateness of the computer codes and models used by Holtec are addressed in the SER and Topical SAR. The NRC staff has reviewed the analyses performed by HOLTEC and found them acceptable. No changes to the CoC, TSs, SER, or Topical SAR are recommended. These models are based on sound engineering sciences and processes.

Comment No. 3: One commenter requested that a troubleshooting manual be prepared that includes information on how many of what type cask are loaded, where and how long they have been loaded, and on problems that have occurred, and the solutions. The commenter is seeking basic information that is periodically updated.

Response: This comment is beyond the scope of this rulemaking.

Cladding Integrity

Comment No. 4: One commenter noted that Holtec's conclusion that fuel rod integrity will be maintained under all accident conditions is based on the fact that the HI-STAR 100 system is designed to withstand a maximum deceleration of 60 g, while a Lawrence Livermore National Laboratory Report (UCID-21246, Dynamic Impact Effects on Spent Fuel Assemblies, Chum, Witt, Schwartz (October 20, 1987)) (LLNL Report) shows that the most vulnerable fuel can withstand a deceleration of 63 g in the most adverse orientation (side drop). The commenter believes that Holtec and the NRC staff have not demonstrated a reasonable assurance that the cladding will maintain its integrity because Holtec's analysis does not take into account the possible increase in rate of oxidation of cladding of high burnup fuel, and oxidation may cause the cladding to become effectively thinner, decreasing its structural integrity and lowering the 'g' impact force at which fuel cladding will shatter. With respect to a possible increase in rate of oxidation of cladding, Holtec has not factored the information in Information Notice (IN) 98-29, "Predicted Increase in Fuel Rod Cladding Oxidation" (August 3, 1998) into its calculations. The clear implication of IN 98-29, in the commenter's view, is that the lift height of the HI-STAR 100 cask must be reduced to lower the 'g' impact forces on the cladding. Also, the commenter provided a table, "Effects of Changing Variables in Dynamic Impact Effects on Spent Fuel Assemblies," which the commenter believes shows that the maximum 'g' impact force, that high burnup fuel with oxidized cladding can withstand, approaches 45 g.

Response: The NRC disagrees with the comment. Information Notice 98-29 states that high burn-up conditions may increase fuel rod cladding oxidation. The increased rate of oxidation is a function of the fuel burn-up and will only affect cladding in high burn-up fuel applications. In general, fuel with a burn-up exceeding 45,000 MWD/MTU is considered to be a high burn-up fuel. However, the Holtec HI-STAR 100 Storage Cask System is not authorized to contain fuel with a burn-up exceeding 45,000 MWD/MTU. Fuel cooling and the average burn-up approved for the HI-STAR 100 Storage Cask System is: (a) for MPC-24 PWR assemblies, the fuel burn-up is limited to 42,100 MWD/MTU; and (b) for MPC-68 BWR assemblies, the fuel burn-up is limited to 37,600 MWD/MTU. Therefore, the potential for significant amounts of oxidized cladding is not a concern for the HI-STAR 100 Storage Cask System, and the table provided by the commenter regarding the consequences of significantly oxidized fuel cladding is not relevant to the approved contents of this cask design.

Comment No. 5: The same commenter stated that Holtec's SAR for the HI-STAR 100 storage cask relies upon the LLNL report for its estimate of 'g' impact force that will damage fuel cladding but that the LLNL report fails to take into account the increased brittleness of irradiated fuel assemblies. Because the irradiated fuel assemblies may have been embrittled, they would also be less resistant to impact. During the course of a fuel assembly's life, subatomic particle bombardment, including neutron flux, significantly decreases the assembly's ductility and increases the assembly's yield stress, thereby embrittling the fuel assembly.

The HI-STAR 100 design cannot rely on LLNL's analysis, in the commenter's view, because the LLNL analysis does not account for irradiation and embrittlement, which lower the impact resistance of the fuel assemblies. These facts are significant when coupled with the increased oxidation rate reported in IN 98-29 because increased oxidation could tangentially cause an increase in cladding embrittlement. Thus, IN 98-29 compounds the LLNL's error in disregarding the brittle characteristics of irradiated fuel cladding.

Response: The NRC disagrees with the comment. The LLNL Report, as referred to, considers the effects of irradiation on cladding. Table 3 of the report delineates irradiated cladding longitudinal tensile tests on coupon specimens. These test specimens were machined from the cladding. The effects of irradiation will increase the Young's modulus and yield stress but decrease the ductility of the cladding. Figure 5 of the report shows that the total elongation values for zircaloy do not change significantly with strain rate and that the ductility appears to be independent of the level of the g-loading. Further, Figure 5 of the report shows that the yield strength is consistently lower than the tensile strength which suggests that significant margin exists between yielding of the cladding and gross rupture. The allowable 'g' impact force calculation in the report is based on the yield stress. Thus, the approach that is used in the LLNL Report and reflected in the SAR is conservative and acceptable.

Comment No. 6: The same commenter stated that Holtec's calculations rely upon the LLNL report's erroneous assumption that the fuel within the cladding behaves as a rigid rod. Thus, Holtec merely used a static calculation for impact analysis versus a dynamic calculation. This assumption is incorrect, in the view of the commenter. Instead of a homogenous, rigid rod, the fuel rod consists of fuel pellets stacked like coins within thin tubing. In any impact scenario, the fuel assembly acts as a dynamic system with the fuel impacting the inside of the cladding and creating a greater likelihood of cladding rupture. Holtec has not shown that the assumption of a rigid rod is conservative. The thinner cladding due to the increased oxidation serves to compound this effect because a smaller 'g' force would be required to rupture the assembly.

Response: The NRC disagrees with the comment. The assertion that the fuel rod consists of fuel pellets stacked like coins within thin tubing is incorrect for irradiated fuels. The fuel pellets are densely packed inside the fuel tubing, and the effects of irradiation will bond the pellets to each other and to the fuel cladding. Samples of irradiated fuel rods have shown that it is indeed nearly impossible to separate the fuel pellets and the cladding.

It is incorrect to assume the fuel rod acts as a dynamic system with the fuel pellets impacting the inside of the fuel rod cladding during an accident drop event. The fuel pellets are densely packed inside the fuel tube and, for irradiated fuels, the fuel pellets are bonded together and to the cladding. The LLNL Report discussed above has conservatively neglected the contributions of the fuel pellets to fuel rod rigidity. Rather, the report only considers the cladding for calculating the allowable g-load. It is true that the LLNL Report used static calculations to derive the allowable g-load equivalent to the dynamic impact loading. During an accident drop event, the fuel assembly is subjected to dynamic impact loading and the equivalent static g-load is determined by a dynamic analysis. The equivalent static g-load is then shown to be lower than the allowable g-load to ensure the fuel cladding integrity is maintained. The approach is well established and acceptable. Therefore, the NRC staff has found Holtec's accident analysis to be conservative as reflected in SER Chapter 11 and is therefore acceptable.

Comment No. 7: One commenter stated that the calculated health impacts under hypothetical accident conditions discussed in Chapter 7 of Holtec's HI-STAR 100 SAR are not 100 percent conservative. Holtec's original hypothetical design basis accident condition assumed that 100 percent of the fuel rods are nonmechanically ruptured and that the gases and particulates in the fuel rod gap between the cladding and fuel pellet are released to the multi-purpose canister (MPC) cavity and then to the external environment. The accident analysis in the final version increased the amount of radioactivity to the MPC cavity by 5 orders of magnitude in accordance with NUREG-1536, and would have placed doses at 100 m over the EPA's limit of 5 rem. An assumed small leakage rate by the applicant reduced the amount released from the cask cavity to the environment by more than 5 orders of magnitude. This design basis accident no longer represents a loss-of-confinement-barrier accident as originally described.

Response: The NRC disagrees with the comment. The hypothetical accident dose calculation is appropriate. As discussed in Interim Staff Guidance (ISG)-5, Rev. 1, "Normal, Off-Normal, and Hypothetical Accident Dose Estimate Calculations for the Whole Body, Thyroid, and Skin," the hypothetical accident assumes 100 percent fuel rod failure within the MPC cavity and release of radioactivity based on factors from NUREG/CR-6487. The applicant demonstrated that the HI-STAR 100 confinement boundary (MPC) remains intact from all credible accidents. Therefore, there is not a credible loss-of-confinement-barrier accident for the HI-STAR 100. The hypothetical accident leakage is conservatively assumed to be equal to that assumed for normal condition leakage with corrections for accident pressures and temperatures. The normal condition leak rate is specified in TS 2.1.1.

The NRC believes that there is reasonable assurance that the confinement design is adequately rigorous and will remain intact under the normal and accident conditions identified by the applicant. Therefore, the design basis change has been found to be conservative and meets applicable regulations.

Comment No. 8: One commenter requested the criteria for an intact fuel assembly, the number of pinhole leaks, blisters, hairline cracks, and crud. The commenter asked if a visual inspection is required and stated that just performing visual exam was inadequate.

Response: As proof that the fuel to be loaded is undamaged, the NRC will accept, as a minimum, a review of the records to verify that the fuel is undamaged, followed by an external visual examination of the fuel assembly before loading to identify any obvious damage. For fuel assemblies where reactor records are not available, the level of proof will be evaluated on a case-by-case basis. The purpose of this demonstration is to provide reasonable assurance that the fuel is undamaged or that damaged fuel loaded in a storage or transportation cask is confined (canned). The criteria for intact assembly are defined in TS Section 1.1 as being fuel assemblies without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means. Partial fuel assemblies (fuel assemblies from which fuel rods are missing) shall not be classified as intact fuel assemblies unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the original fuel rods.

Radiation Protection

Comment No. 9: One commenter stated that Holtec calculated the radiation dose to an adult 100 meters from the accident due solely to inhalation of the passing cloud without considering other relevant pathways, such as direct radiation from cesium and cobalt-60 deposited on the ground, resuspension of deposited radionuclides, ingestion of contaminated food and water, and incidental soil ingestion, and does not reflect 10 CFR 72.24(m).

Response: The NRC agrees that Holtec calculated the radiation dose to an adult 100 meters from the accident due solely to inhalation of the passing cloud and did not consider direct radiation and ingestion. The NRC staff considers inhalation to be the principal pathway for radiation dose to the public, and Holtec has followed NRC staff guidance in making conservative assumptions regarding the source term and duration of the release. In SER Chapter 10, the NRC staff found that the radiation shielding and confinement features of the cask design are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106. Section 72.106 addresses postaccident dose limits.

When a general licensee uses the cask design, it will review its emergency plan for effectiveness in accordance with 10 CFR 72.212. This review will consider interdiction and remedial actions to monitor releases and pathways based on the chosen site conditions and the location. Therefore, the pathways identified by the commenter will be addressed in the general licensee's site specific review.

Comment No. 10: One commenter stated that Holtec has not specifically calculated potential radiation dose to children, and this does not meet NRC regulations. Further, the commenter stated that NRC's methodology for calculating the potential dose to children is deficient.

Response: The NRC disagrees with the comments. While Holtec did not specifically calculate potential radiation dose to children, the international community and the Federal agencies (including EPA and the NRC) agree that the overall annual public dose limit, from all sources, should be 1 mSv (100 mrem) which is protective of all individuals. The purpose of the public dose limit is to limit the lifetime risk from radiation to a member of the general public. Variation of the sensitivity to radiation with age and gender is built into the standards which are based on a lifetime exposure. A lifetime exposure includes all stages of life, from birth to old age. For ease of implementation, the radiation standards, that are developed from the lifetime risk, limit the annual exposure that an individual may receive. Consequently, the unrestricted release limit of 0.25 mSv (25 mrem), a small fraction of the annual public dose limit, is protective of children as well as other age groups 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 May 21, 1991, Federal Register notice (56 FR 23387).

Comment No. 11: One commenter asked if the streaming dose rates have been measured and if not, will they be measured on the first cask loading?

Response: There is no NRC regulatory requirement to measure streaming dose rates at the first cask loading. Further, the applicant did not provide measured dose rates from cask streaming in its application because it was not required. The applicant did provide calculated streaming dose rates in the SAR shielding analysis. The HI-STAR 100 system is designed to eliminate significant streaming paths, and each user is required to operate the HI-STAR 100 under a 10 CFR Part 20 radiological program. NRC has reasonable assurance that the general licensee's radiological protection and ALARA program will detect and mitigate exposures from any significant or unexpected radiation fields for each cask loading.

Comment No. 12: One commenter stated that the applicant should have performed a specific analysis for off-normal conditions for confinement analysis and should have included an " ^{85}K" (Kr-85) dose calculation to the skin.

Response: The NRC agrees. The applicant should have done an off-normal condition confinement analysis; however, the off-normal case dose is approximately a factor of 10 greater than normal dose. The Holtec normal condition results show acceptable doses when the factor of 10 is applied for off-normal conditions and have been found acceptable as reflected in the SER. No additional action is necessary to meet applicable NRC regulations.

Comment No. 13: One commenter stated that the licensees' report on specific site doses to the public should be included in the PDR.

Response: The dose for a site-specific location is beyond the scope of this rulemaking. Licensees are required to meet the dose restriction in 10 CFR Part 20.

Comment No. 14: One commenter asked for a definition of inflatable annulus seal. The commenter further questioned the checks and criteria for surface contamination.

Response: The inflatable annulus seal, which is discussed in Sections 1.2.2.1, 8.1, and 10.1.4 of the SAR, is designed to prevent radionuclide contamination of the exterior MPC while the cask is submerged in a contaminated spent fuel pool. The space between the MPC and overpack is filled with clean water and is sealed at the top of the MPC with the inflatable annulus seal. After the seal is removed, the upper accessible portion of the MPC is examined for contamination to verify that the seal remained intact during underwater loading. NRC found the seal description and operation to be acceptable. Each general licensee will develop site-specific operating procedures that address the use of the inflatable annulus seal. Each general licensee will also operate the HI-STAR 100 under a 10 CFR Part 20 radiological protection program.

Comment No. 15: One commenter suggested that there should be criteria for the distance of dose measuring mechanism from the cask and personnel during loading and unloading.

Response: NRC disagrees with this suggestion because NRC regulations do not specifically require these criteria for dose measurement. Each general licensee is required to operate the HI-STAR 100 under a 10 CFR Part 20 radiological program and must develop site-specific operating procedures that include radiological protection dose surveys that must be conducted during loading and unloading operations.

Sabotage Events

Comment No. 16: One commenter stated that the current sabotage design basis is not a bounding accident and that the NRC should consider the effect of a sabotage event with an anti-tank missile. There is a lack of a comprehensive assessment of the risks of sabotage and terrorism against nuclear waste facilities and shipments. The NRC staff could impose additional conditions on dry storage casks and Independent Spent Fuel Storage Installations (ISFSIs), e.g., the CoC could require that an ISFSI be designed with an earthen berm to remove the line-of-sight.

The commenter stated that since the early 1980s, the NRC has relied on and poorly interpreted an outdated set of experiments carried out by Sandia National Laboratory and Battelle Columbus Laboratories that measured the release of radioactive materials as a result of cask sabotage. The NRC has never estimated the economic and safety implications of a sabotage event at a fixed storage facility. Following the publication of these Sandia study results, the NRC proposed elimination of a number of safety requirements for shipments of spent fuel. At least 32 parties submitted more than 100 pages of comments in response to the notice, to which the NRC never publicly responded. The NRC suspended action on the rulemaking but inappropriately continues to use the unrevised conclusions in the proposed rule as a basis for its policies on terrorism and sabotage of nuclear shipments.

Response: The NRC disagrees with the comment. The NRC reviewed potential issues related to possible radiological sabotage of storage casks at reactor site ISFSIs in the 1990 rulemaking 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 the 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, and the licensee conducts periodic patrols and surveillances to ensure that the physical protection systems are operating within their design limits. It is the ISFSI licensee who is responsible for protecting spent fuel in the casks from sabotage rather than the certificate holder. Comments on the specific transportation aspects of the cask system and existing regulations specifying what type of sabotage events must be considered are beyond the scope of this rulemaking.

Comment No. 17: One commenter asked whether an evaluation for a truck bomb sabotage event has been conducted.

Response: The staff has evaluated the effects of a truck bomb located adjacent to storage casks. 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). Each utility licensed to have an ISFSI at its reactor site is required to develop physical protection plans and install a physical protection system that provides high assurance against unauthorized activities that could constitute an unreasonable risk to the public health and safety. The physical protection systems at an ISFSI and its associated reactor are similar in design to ensure the detection and assessment of unauthorized activities. Response to intrusion alarms is required. Each ISFSI is periodically inspected by NRC, and the licensee conducts periodic patrols and surveillances to ensure that security systems are operating within their design limits. The NRC believes that the inherent nature of the spent fuel and the spent fuel storage cask provides adequate protection against a vehicle bomb, and has concluded that there are no safety concerns outside the controlled area.

Thermal Requirements

Comment No. 18: One commenter stated that the CoC temperature limits for the storage cask are deficient because they do not take into account a minimum pitch or center-to-center distance between casks to be stored in the ISFSI. Further, Holtec has not performed rigorous calculations to support the assigned pitch of 12-foot or 4-foot spacing between casks based on the amount of detail in its nonproprietary version of its analyses.

Response: The NRC disagrees with the comment. In Section 4.4.1.1.7 of the SAR, Holtec addressed the heat transfer interaction between the overpacks for a cask array at an ISFSI site. No forced convection was assumed (e.g. stagnant ambient conditions which would maximize the interaction heat effect). The applicant further adjusted the heat transfer in accordance with ANSYS methodology and applied it in the calculations. Further, in SER Section 4.5.2.1, the NRC staff noted that the applicant considered in its temperature calculations that multi-purpose cask baskets were loaded at design basis maximum heat loads, and systems were considered to be arranged in an ISFSI array and subjected to design basis normal ambient conditions with insulation. The NRC staff concluded in the SER that it has reasonable assurance that the spent fuel cladding will be protected against degradation by maintaining the clad temperature below maximum allowable limits.

Miscellaneous Items

Comment No. 19: One commenter asked why a coating without zinc was not required for the VSC-24 cask design. The commenter further questioned why NRC allowed coatings to be applied to casks because it will create problems for future DOE waste disposal.

Response: NRC regulations do not prohibit the use of coatings in a cask design. An applicant must provide information in its safety analysis report to support use of coatings. The applicant should describe the near and long term effects of the coatings on systems important to safety including the benefits and potential impacts of coating use. Based on the applicant's analysis, the NRC reviews and assesses the use and adequacy of the coatings. Specific comments relating directly to VSC-24 are beyond the scope of this rulemaking.

Comment No. 20: One commenter asked why the current HI-STAR 100 is not an ASME stamped component.

Response: NRC regulations do not require an ASME stamp for a cask. 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." Applicant submittals are reviewed to the criteria in the Standard Review Plan. Cask fabrication activities are inspected by the licensees and the NRC staff to ensure that components are fabricated as designed.

Comment No. 21: One commenter asked a number of questions related to the Boral and NS-4-FR concerning (1) whether it has been used "over time" in a cask, (2) the amount of "creep or slump" that has occurred over time, (3) how the testing is conducted, and (4) how the Boral content is tested in the panels. The commenter further asked if fabrication is inspected and why no surveillance or monitoring program is required to check the Boral content.

Response: The questions and comments on the Boral neutron absorber are addressed in Sections 6.4.2 and 9.1.4 of the SER and Sections 1.2.1.3.1, 6.3.2, and 9.1.5.3 of the SAR. The NRC routinely accepts the use of Boral as a neutron absorber for storage cask applications, and it has been used in casks. NRC has approved both storage and transportation cask designs that use Boral. Section 1.2.1.3.1 of the SAR describes the historical applications and service experience of Boral. This information indicates that Boral has been used since the 1950's and used in baskets since the1960's. Several utilities have also used Boral for nuclear applications such as spent fuel storage racks. Based on industry experience, no credible mechanism for "creep or slump" of Boral in the cask has been identified.

Sections 1.2.1.3.1 and 9.1.5.3 of the SAR describe the testing procedures for Boral. Boral 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. A statistical sample of each manufactured lot of Boral is tested by the manufacturer using wet chemistry procedures and/or neutron attenuation techniques.

The Boral is designed to remain effective in the HI-STAR 100 system for a storage period greater than 20 years and there are no credible means to lose the Boral. Further, the NRC accepts the use of NS-4-FR as a neutron absorber for storage cask applications, and it has been used in other casks. Therefore, surveillance and monitoring are not needed.

Comment No. 22: One commenter provided a discussion on the VSC-24 design. The issues included materials, the use of coatings, the use of March Metalfab as a fabricator, calculations being performed when problems are being solved, testing of soils and pads, and cask handling temperatures.

Response: These comments are beyond the scope of the current rulemaking.

Comment No. 23: One commenter asked how the prepossession or anodization of aluminum surfaces is checked and what the criteria were for the inspection.

Response: The NRC disagrees that an inspection is necessary. The only aluminum used in the MPC-24 or MPC-68 is for the Boral neutron absorbers. Aluminum forms a very thin, adherent film of aluminum oxide whenever a fresh cut surface is exposed to air or water, becoming thicker with increasing temperatures and in the presence of water (Source: "Corrosion Resistance of Aluminum and Aluminum Alloys," Metals Handbook, Desk Edition, American Society for Metals, 1985). Thus, no inspection or acceptance criteria are necessary.

Comment No. 24: One commenter requested clarification on whether the helium will be pure and not mixed with krypton or xenon that would have an effect on internal pressure or temperature. The commenter also asked whether the helium had to be dry.

Response: Only pure helium will be used to backfill the cask; no krypton or xenon gasses will be added during backfill. Technical Specification Table 2-1, Footnote 1, specifies that helium used for backfill of MPC shall have a purity of ≥99.995%. Acceptable helium purity for dry spent fuel storage was defined by R. W. Knoll et al. at Pacific Northwest Laboratory (PNL) in "Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel," PNL-6365, November 1987. Helium purity is addressed in SAR Section 8.1.4, MPC Fuel Loading, Step 28, and SER Section 8.1.3.

Comment No. 25: One commenter asked whether leakage of gases, volatiles, fuel fines, and crud was considered credible and whether the analysis addressed this concern.

Response: The applicant has calculated the postulated annual dose at 100 meters assuming a realistic leakage rate consistent with ANSI N14.5 Standard "Leakage Tests on Packages for Shipment for Radioactive Materials" (1997) and has reflected the results in SAR Chapter 7. The applicant's analysis addresses the commenter's concern, and the calculated dose had been found to be within regulatory guidelines (limits) and acceptable to the NRC staff.

Comment No. 26: One commenter was concerned that the cask could drop or tip over in the loading area of the plant and whether this has been evaluated. The commenter was also concerned about a drop or tip over during transfer from the pad or during transport and that all of the analysis seemed to be for the pad.

Response: The tipover, end drops, and horizontal drop analyses form part of the structural design basis for the HI-STAR 100 cask design. Holtec described drops and tipover analyses in SAR Section 3.4.9. The NRC's evaluation of the vendor's analyses is described in

SER Sections 3.2.3.1 and 3.2.3.2. The NRC found the results of these analyses to be satisfactory in that the calculated stresses were within the allowable criteria of the American Society of Mechanical Engineers (ASME) Code. Before using the HI-STAR 100 casks, the general licensee must evaluate the foundation materials to ensure that the site characteristics are encompassed by the design bases of the approved cask. The events listed in the comment are among the site-specific considerations that must be evaluated by the licensee using the cask.

Comment No. 27: One commenter asked whether the design has been evaluated for a seismic event during loading and unloading.

Response: The HI-STAR 100 casks can only be wet loaded and unloaded inside the fuel handling facility. Generally, these activities take place in a segregated under-water cask loading pit which would limit cask movement during a seismic event. The cask will be supported for a seismic event during loading and unloading. General procedure descriptions for these operations are summarized in Sections 8.1 and 8.3 of the SAR. Detailed loading and unloading procedures are developed and evaluated on a site-specific basis by the licensee using the cask.

Comment No. 28: One commenter questioned whether the method for cooling has been tested with a real cask.

Response: The NRC regulations and guidance in the Standard Review Plan require the review and approval of the design criteria. No testing is required for approval of the design under this current rule. The cask user is required to perform preoperational testing to determine the effectiveness of the cooling methods.

Comment No. 29: One commenter questioned whether the manufacturer's literature for the "high emissivity" paint on the overpack had been evaluated and tested, how the testing was done, and what the results were. The commenter also questioned whether/how the painted components were safely stored. The commenter further stated that the paint on the surfaces of the overpack should be a specified paint, not just a requirement of "an emissivity of no less than 0.85."

Response: The manufacture and application of high-emissivity paints is not a new technology. Several manufacturers provide paints with specified emissivity ratings. Thermal tests are required to confirm the heat transfer capabilities of the inner and intermediate shells and radial channels. Annual cask inspection will check the exterior surface conditions at which time the paint will be examined and touched up in local areas as necessary. The NRC does not believe that identifying a specific brand name of paint is required. There are several suppliers who manufacture paints with the specified emissivity. The NRC has reviewed the applicant's analysis and found that paints with an emissivity greater than 0.85 are acceptable.

Comment No. 30: One commenter questioned the drain down time and asked how frequently the water is checked. The commenter requested information on what happens if the MPC can't be vacuum dried successfully and when the fuel needs to be put back in the spent fuel pool.

Response: The drain down time is not specified in the TSs but is part of the vacuum drying procedure. The TSs state that the vacuum drying must be completed within 7 days.

There is not a specific procedure in the application to monitor the water content; however, that will be addressed by the cask user on a site-specific basis and is beyond the scope of this rulemaking. If the drying process is unsuccessful and the TS requirements cannot be met within 30 days, the fuel assemblies must be moved from the cask and be placed in the spent fuel pool.

Comment No. 31: One commenter requested information on the cask storage array on the pad and the radiation affect from other casks in a full cask array. The commenter further requested information on how the applicant/certificate holder/licensee will examine and/or test the HI STAR 100 and who was actually responsible for the test. The commenter questioned whether a domed cask cover would be better for runoff and sky shine concerns.

Response: The applicant performed a shielding analysis that included a three-by-three cask array (square) model to simulate the average dose contribution from the center cask, which is partially shielded by the surrounding periphery casks. This value is applied in an offsite dose formula used to estimate offsite doses from every cask in the array. The center-to-center cask pitch was assumed to be 12 feet in the shielding analyses. Testing of the actual as-installed configuration will be performed by the cask user and will be evaluated at that time. Offsite dose estimates for a typical ISFSI array, including the affects of multiple casks and skyshine, are discussed in Sections 5.4.3 and 10.4.1 of the SAR. NRC found the dose estimates to be acceptable. As required in 10 CFR 72.212, each general licensee will perform a site-specific dose evaluation to demonstrate compliance with Part 72 radiological requirements. The general licensee will identify an ISFSI configuration and may elect to use additional engineered features of its choosing, such as shield walls, a domed cover, or berms, to ensure compliance with radiological requirements. Section 1.4.7 of Appendix B to the CoC requires that any such engineered feature be considered important to safety and evaluated to determine the applicable quality assurance category.

Comment No. 32: One commenter questioned what the criteria were for the polyester resin "poured" into radial channels, how they were tested, handled and inspected, and whether they had been tested in a real cask. The commenter questioned whether a "poured" neutron shield was really safe and whether uncontrolled voids caused a problem with occupational dose requirements. The commenter stated that poured neutron shields should not be used.

Response: The NRC has reviewed Holtec's application that described the neutron shielding to be used to meet the requirements of 10 CFR 72.104 and 72.106. The NRC found the Holtec approach acceptable. The methods for testing, handling, and inspecting installation of the shielding are beyond the scope of this rulemaking. However, poured neutron shielding has been successfully used in other cask designs.

Comment No. 33: One commenter stated that appropriate limits for burnup should be specified in the CoC. The commenter is concerned that the SAR analysis assumed significantly higher burnups than allowed and significantly higher initial uranium loading than specified in the table.

Response: Burnup, cooling time, initial uranium loading, and initial enrichment are parameters that affect the total source term (radioactivity) of spent fuel. The applicant's source term analysis assumed higher uranium loadings and higher burnups than those specified in TSs of the CoC. Therefore, the radiological source term is conservative relative to the allowed burnups and uranium loadings.

As discussed in Section 5.2.1 of the preliminary SER, for the same level of burnup, neutron source terms typically increase as initial enrichment decreases. Therefore, the source term analysis employed lower-than-average enrichment values. Based on the SAR analyses, conditions of the CoC, and other requirements in Parts 20 and 72, the NRC has determined that minimum enrichment is not warranted as an additional operating control for the HI-STAR 100. Specific reasons for this determination include the following: (1) the enrichments bound a significant portion of spent fuel, and the source terms are calculated for burnups significantly higher than those allowed in the CoC; (2) the radiological source terms are adequately controlled in the CoC by limits on maximum burnup, minimum cooling time, maximum initial uranium loading, and maximum decay heat; (3) dose rates are controlled in the CoC by specific dose limits for the top and side of the cask that are based on values calculated in the shielding analysis; (4) each general licensee will perform a site-specific dose evaluation to demonstrate compliance with Part 72 radiological requirements; and (5) each general licensee will operate the ISFSI under a Part 20 radiological protection program.

NRC agrees with the comment that the preliminary SER term of "low probability" may not provide definite criteria for general license cask users regarding limitations on minimum enrichment. Therefore, Chapter 5 of the SER has been revised to clarify that minimum enrichment is not an operating control for the HI-STAR 100.

Comment No. 34: One commenter asked what has been considered as credible ways to lose the fixed neutron poisons.

Response: The NRC staff does not consider the loss of fixed neutron poisons to be credible after they are installed into the cask because the poisons are fixed in place and contained.

Comment No. 35: A commenter questioned how the welds of the MPC lid and closure ring are tested and asked for the acceptance criteria.

Response: Information on the welds is contained in SAR Tables 9.1.1, 9.1.2, and 9.1.3.

Comment No. 36: One commenter asked whether shims are used and stated that shims or gaps were not acceptable.

Response: There are no shims used in the closure weld of the HI-STAR 100 casks. The only shims used are located between the canister and the overpack at basket support locations to provide additional support for the basket supports. The actual thickness of the shim will depend on the gaps between the cask and the inside cavity of the overpack at the basket support locations. Gaps between separate components such as the cask and the overpack are unavoidable and are necessary to ensure that there will be no physical interferences and to allow free thermal expansions.

Comment No. 37: One commenter stated that all welds should be monitored unless they have been tested.

Response: NRC accepts welded closure of casks. The regulations do not require monitoring or testing of welds because there are no expected degradation mechanisms identified during the cask usage life. However, both the fabricator and cask user will examine and inspect all welds as appropriate.

Comment No. 38: One commenter stated that the detailed loading and unloading procedures developed by each cask user should be put in the PDR.

Response: Loading and unloading procedures are site-specific issues not required for design approval and are beyond the scope of this rulemaking.

Comment No. 39: One commenter asked how long before an ultrasonic testing examination is conducted should the equipment be calibrated.

Response: Comments on the site-specific examination techniques and associated calibration are beyond the scope of rulemaking for the HI-STAR 100 system.

Comment No. 40: One commenter was concerned over the possibility that the bolts could rust and crack over time or become brittle and crack because water, ice, and frost could get into the bolt holes over the years.

Response: The NRC disagrees with this concern over the integrity of the bolting material. The 54, 1 5/8-inch-diameter, closure plate bolts are made from ASME SB-637-N07718 material per SAR BM-1476. N07718, a nickel-chromium alloy, does not become brittle at colder temperatures. N07718 is a high strength, corrosion resistant material used in applications with a temperature range from -423 °F (-253°C) to 1300°F (704°C) (Source: Inconel Alloy 718, Inco Alloys International, fourth edition, 1985). This material will not rust, unlike carbon steels in corrosive environments. In addition, the material retains significant ductility down to -320°F (-196°C) as shown by impact test results (Source: Inconel Alloy 718, Table 27). Therefore, the NRC has no concerns about the bolting material.

Comment No. 41: One commenter asked what type of radiographic exam is applicable and where it would be conducted.

Response: SAR Tables 9.1.1, 9.1.2, and 9.1.3 describe which radiographic exams are to be performed and when they are required to be performed.

Comment No. 42: One commenter disagreed with allowing the use of a penetrant test in lieu of volumetric examination on austenitic stainless steels because flaws in these are "not expected" to exceed the thickness of the weld head. The commenter believes that volumetric welds should be required because if you don't know for sure the real size of the actual weld, how can you accept a certain flaw size? The commenter asked how the permanent record is kept and stated that black and white photographs should be used as a permanent record.

Response: NRC disagrees with this comment. The NRC position on inspection of closure welds is contained in ISG-4, "Cask Closure Weld Inspections." Actual cask welds are examined in accordance with site-specific procedures that are beyond the scope of rulemaking for the HI-STAR 100 system. Nondestructive Examination (NDE) methods are specified in accordance with Section III "Rules for Construction of Nuclear Power Plant Components," and Section V "Nondestructive Examination," of the ASME Code and are already described in SAR Tables 9.1.1, 9.1.2, and 9.1.3. 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.

Comment No. 43: One commenter believed that the marking material for the casks should be designated and that the mark needed to be permanent.

Response: NRC agrees with the comment. The storage marking nameplate is made from a 4-inch by 10-inch, 14-gauge Type 304 stainless steel sheet and welded to the outside of the HI-STAR 100 Overpack. Lettering will be etched or stamped on the plate. Details are shown in SAR Drawing 1397, Sheet 4 of 7, and described in SER Section 9.1.6. The nameplate will provide appropriate cask identification that will last well beyond the design life of the HI-STAR 100 system. No nonpermanent marking will be used.

Comment No. 44: One commenter requested information on "rupture disc replacements," how they are tested for replacement, what the time criteria are, and what is considered a rupture.

Response: The rupture disc is located in the neutron shield tank of the HI-STAR 100 casks. The purpose of the rupture disc is to limit pressure build-ups to a precalculated level within the neutron shield tank during the fire accident condition. When the pressure build-up exceeds the precalculated design pressure, the disc will rupture to relieve the pressure. The rupture disc is tested and certified by the manufacturer. There is no regulatory requirement for the replacement of rupture discs. The SAR has arbitrarily set a replacement schedule for every 5 years to assure functionality.

Comment No. 45: One commenter asked if the casks are checked in winter for ice and snow loads or ice around the base and if the pads will be kept clean.

Response: Casks are designed for the worst ice and snow loads possible. Ice build-ups around the cask base are not allowed, and the pad will be kept clean. Site-specific procedures will address these items.

Comment No. 46: One commenter questioned if there was an evaluation for a plane crash, with a fuel fire, into a cask or full cask array conducted and whether there is a stipulation as to putting a pad in an area where planes regularly fly.

Response: Before using the HI-STAR 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 cask as required by 10 CFR 72.212(b)(3). The licensee's site evaluation should consider the effects of nearby transportation and military activities. Generally, a cask's inherent design will withstand tornado missiles and collision forces imposed by light general aviation aircraft (i.e., 1500-2000 pounds) that constitute the majority of aircraft in operation today. The events listed in the comment are among the site-specific considerations that must be evaluated and are beyond the scope of this rulemaking.

Comment No. 47: One commenter questioned why Holtec stated that the HI-STAR 100 could be part of the final geologic disposal system.

Response: The NRC is not reviewing this design for use in a final geologic disposal system, but only for interim storage under Part 72.

Comment No. 48: One commenter asked where the MPC shell weld is located and if the pocket trunnions at the bottom of the overpack have been analyzed specifically for tipovers and falls.

Response: The MPC shell has multiple welds located both longitudinally on the side of the MPC and circumferentially on the top and bottom of the MPC. The pocket trunnions at the bottom overpack have been analyzed by the applicant for tipovers and falls. The NRC reviewed the design for normal, off-normal, and accident conditions, and found it acceptable.

Comment No. 49: One commenter stated that the lifting and pocket trunnions should be checked over the years for cracking or brittleness and for debris accumulation and should be kept ready for use over the years.

Response: The NRC agrees with this comment. As shown in SAR Table 9.2.1, lifting trunnion and pocket trunnion recesses are visually inspected before the next handling operation after HI-STAR 100 casks are placed on the ISFSI pad. The trunnion material has been evaluated for brittle fracture and found to be satisfactory for the operating temperature range. In addition, the trunnions are load tested in accordance with ANSI N14.6, "American National Standard for Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10000 Pounds (4500 kg) or More." Thus, there is no credible reason to suspect undetected cracking or brittleness. The pocket trunnion recess is closed by a pocket trunnion plug during storage. There is no possibility of animal and bird access and nesting in the recess.

Comment No. 50: One commenter requested information on the criteria for the critical flaw size.

Response: The criteria for critical flaw size are included in ISG No. 4, "Cask Closure Weld Inspections." The NRC review determined that Holtec's proposed methodology is consistent with this ISG.

Comment No. 51: One commenter asked how subcontractors are to be audited and inspected.

Response: This comment is beyond the scope of this rulemaking.

Comment No. 52: One commenter believed that the first cask for each utility should be tested at a full heat load and asked what is meant by the "First System In Place" requirement.

Response: The heat transfer characteristics of the cask system will be recorded by temperature measurements for the first HI-STAR 100 systems (MPC-24 and MPC-68) placed into service with a heatload greater than or equal to 10 kW. An analysis shall be performed by the cask user that demonstrates that the temperature measurements validate the analytical methods and the predicted thermal behavior described in Chapter 4 of the SAR.

The cask user will perform validation tests for each subsequent cask system that has a heat load that exceeds a previously validated heat load by more than 2 kW (e.g., if the initial test was conducted at 10 kW, then no additional testing is needed until the heat load exceeds 12 kW). No additional testing is required for a system after it has been tested at a heat load greater than or equal to 16 kW.

The cask user will provide a letter report to the NRC in accordance with 10 CFR 72.4 summarizing the results of each of these validation tests. Cask users may also satisfy these testing and reporting requirements by referencing validation test reports submitted to the NRC by other cask users with identical designs and heat loads.

Comment No. 53: One commenter asked how much water is to be drained under the MPC lid before welding and how the temperature enters into the calculations.

Response: Chapter 8 of the SAR directs the operators to pump approximately 120 gallons of water from the MPC before commencing welding operations. The water level is lowered to keep moisture away from the weld region. Under these conditions, ample water remains inside the MCP to maintain cladding temperatures well below their short term limits. This operating condition has been evaluated by the NRC. The resulting temperature increase is much less than any previously analyzed accident condition might produce.

Comment No. 54: One commenter asked how lifting height should be verified and stated that the height should be recorded.

Response: The maximum lifting height maintains the operating conditions of the Spent Fuel Storage Cask (SFSC) within the design and analysis basis. It is the general licensee's responsibility to limit the SFSC lifting height to allowable values. The lift height requirements are specified in TS LCO 2.1.7 for the vertical and horizontal orientations. Surveillance requirements require verification that SFSC lifting requirements are met after the SFSC is either suspended or secured in the transporter and prior to moving the SFSC within the ISFSI.

Comment No. 55: One commenter questioned how the MPC closure ring, lid, vent, and drain covers are removed during unloading and what precautions are taken.

Response: The specific procedures for removal of the closure ring, lid, vent, and drain covers are to be developed by the cask user. These procedures will be evaluated by the licensee and by the NRC during inspections to address adequacy and implementation and, therefore, are beyond the scope of this rulemaking.

Comment No. 56: One commenter questioned that if the MPC gas temperature is not met, what additional actions are required and have they been evaluated (TS B3.1.8-3)?

Response: The NRC staff has evaluated this condition. The TSs require that if the MPC gas temperature is exceeded during unloading, no additional operational actions may be conducted until the temperature is restored to below the TS limit.

Comment No. 57: One commenter asked if "dry" unloading operations are considered.

Response: A dry unloading operation was not requested or explicitly described in the SAR and thus is not currently allowed for the HI-STAR 100 system and is beyond the scope of this rulemaking.

Comment No. 58: One commenter questioned if crud disposal is a problem and how it can be mitigated.

Response: Dispersal of crud is beyond the scope of this rulemaking and is a site-specific issue. Experience with wet unloading of some fuel types after transportation has involved handling significant amounts of crud. However, the NRC notes that the HI-STAR generic unloading procedures mitigate crud dispersal. As discussed in Section 8.3.1 of the SAR, these procedures include gas sampling of the MPC internal atmosphere and specific cool-down steps. Each cask user will develop additional site-specific unloading procedures based on its radiological protection program to further address and mitigate crud dispersal.

Comment No. 59: The applicant made comments relevant to the helium backfill pressure of the cask. After discussions with the NRC staff, Holtec withdrew this comment during a telephone conversation on 5/7/99.

Response: Not applicable.

Comments on Proposed TSs:

Upon review of the public comments received on the proposed TSs for the HI-STAR-100 Storage Cask, particularly comments received from EXCEL Corporation and the Holtec Users Group, the NRC staff has determined that several structural changes to the TSs were in order. These changes result in a clearer set of TSs and move the TSs for the new generation of dual-purpose cask systems toward a standardized format.

Comment No. 60: It was suggested that controlling the bases for the TSs as part of the CoC would result in administrative burdens to all involved. These bases are not controlled as part of power reactor licenses.

Response: The NRC staff agrees. Therefore, the bases have been relocated to an appendix to the SAR.

Comment No. 61: A number of commenters also raised concerns with the inclusion of the extensive fuel specifications (formerly Section 2.0) and a very lengthy design specification section (formerly Section 4.0).

Response: The NRC staff agrees that placement of much of this information in the TSs is unwarranted. Therefore, much of the information regarding fuel specifications and some of the design and codes information were moved from the TSs to a separate appendix to the CoC. However, the NRC staff did maintain some of the information regarding requirements for bases controls by adding it to a revised Section 3.0, "Administrative Controls and Programs," of the TSs.

Upon consideration of public comments and further consideration within the NRC, the NRC staff has determined that the structure of TS Section 2.1, "SFSC INTEGRITY," did not provide appropriately clear guidance. Therefore, the NRC staff has revised this section of the TSs to reflect a more logical and focused approach. The number of limiting conditions for operations (LCOs) in this section has been reduced to four. The NRC staff believes that this will enhance the usefulness of the TSs.

Comment No. 62: One commenter stated that if surface contamination exceeds 2200 dpm/100 cm2 from gamma and beta emitting sources, and smearable contamination limits cannot be reduced to acceptable levels, the TSs require actions up to and including removal of the MPC from the HI-STAR 100 overpack after removing the spent fuel from the MPC. The commenter stated that the proposed Skull Valley ISFSI in Utah does not have facilities for decontaminating casks and, therefore, these TSs could not be met.

Response: The NRC agrees in part. The revised version of the TSs (TS 2.2.2) requires verification that removable contamination is within limits during loading operations and provides up to 7 days to restore the contamination within limits. The specifications no longer list MPC or spent fuel removal actions. Further, comments on the proposed site-specific Skull Valley ISFSI currently under review are beyond the scope of this rulemaking. Decontamination requirements will be reviewed as part of the site-specific licensing provisions under Part 72 Subpart B for the Skull Valley ISFSI.

Comment No. 63: One commenter stated that the definition of "TRANSPORT OPERATIONS" needs to be revised to reflect that the drop analysis is not limited to drops from the transporter, and that lifting of a cask with other devices is not prohibited. The commenter recommended similar changes to the definition of "LOADING OPERATIONS" and "UNLOADING OPERATIONS."

Response: The NRC disagrees. The definitions of the three terms in question do not prohibit lifting of a cask with other devices (the revised note in TS 2.1.3 clarifies this issue), nor do the definitions affect the lifting requirements contained in TS 2.1.3.

Comment No. 64: One commenter stated that it would increase the standardization of the TSs by relocating the explanatory information of the defined terms in TS Section 1.0 to the TS Bases.

Response: The NRC disagrees with the comment. The terms defined in TS Section 1.0 are important in the understanding of the TS requirements. These definitions need to be contained within the TSs. This practice is consistent with the standard TSs developed for the

U.S. nuclear power reactors.

Comment No. 65: One commenter stated that in Examples 1.3-2 and 1.3-3, the word "action" should be capitalized.

Response: The NRC agrees. The word "action" has been capitalized.

Comment No. 66: One commenter recommended the removal of portions of Table 2.1-1 and all of Table 2.1-2 and Table 2.1-3 from the TSs.

Response: The NRC agrees, in part, that this information should be moved. This design information is crucial to the conclusions reached by the NRC staff in its SER; therefore, the design information contained in these tables has been relocated (and renumbered) to a separate appendix to the CoC, along with other critical design information.

Comment No. 67: One commenter recommended a change to the format of the Titles of Tables 2.1-1, 2.1-2, 2.1-3, and 2.1-4.

Response: The NRC agrees with the comment. The format has been changed.

Comment No. 68: One commenter recommended a wording change in TS Section 3.0 from "not applicable to an SFSC" to "not applicable."

Response: The NRC agrees with this comment and has made the indicated change.

Comment No. 69: One commenter stated that there is no need to create two specifications for TS 3.1.1, MPC Cavity Vacuum Drying Pressure, and TS 3.1.2, OVERPACK Annulus Vacuum Drying Pressure. In addition, the commenter indicated there is no need to create two specifications for TS 3.1.5, MPC Helium Leak Rate, and TS 3.1.6, OVERPACK Helium Leak Rate.

Response: The NRC agrees with the comment. Section 2.1 of the TSs has been revised based on these and similar comments received to combine these TSs.

Comment No. 70: One commenter stated that the frequency of SR 3.1.7.1 should be revised because, as written, the frequency would apply only when a cask is being moved to or from the ISFSI and would not apply at other times, such as when moving casks within the ISFSI. However, the drop analysis applies any time the cask is suspended. The frequency should be revised similar to "Prior to movement of an SFSC."

Response: The NRC agrees with the comment. The frequency of SR 3.1.7.1 has been revised.

Comment No. 71: One commenter recommended that TS Sections 4.1 and 4.2 be eliminated because they contain no unique information.

Response: NRC agrees with the comment. Sections 4.1 and 4.2 have been eliminated.

Comment No. 72: One commenter recommended relocating the information contained in

TS Sections 4.3 and 4.5 to the SAR, and recommended eliminating TS Section 4.4, stating that this section is a duplication of existing regulatory requirements.

Response: The NRC agrees in part. The NRC staff agrees that these sections do not belong in the TSs. This design information has been relocated to Appendix B to the CoC. The NRC staff disagrees with the commenter's proposal to eliminate or relocate these sections to the SAR. The NRC has relocated these sections to Appendix B to the CoC due to the importance of the design information contained in these sections. The NRC staff also disagrees with the comment that TS Section 4.4 is a duplicate of existing regulations, since this section contains the acceptance criteria for the site-specific design parameters.

Comment No. 73: A commenter recommended relocating the information contained in TS Sections 4.6 and 4.8 to an Administrative Controls chapter due to their content and relocating Section 4.7 to the SAR because it is a one-time administrative task.

Response: The NRC agrees in part. The NRC staff agrees that these sections belong in the administrative section of the TSs and has placed this information in a new TS Chapter 3.0, "Administrative Controls and Programs." The NRC staff disagrees with the commenter on the proper location of Section 4.7 (now TS Section 3.2), because it is established NRC staff practice to place important administrative requirements, even one-time requirements, in the TSs.

Comment No. 74: A commenter stated that TS 3.1.8 contains conflicts because the APPLICABILITY statement, and the COMPLETION TIME when the condition is not met, are the same statement. The commenter further recommended that because of its complexity and rarity of its use, this specification be eliminated and the information specified in the SAR.

Response: The NRC agrees in part. The NRC agrees with the first point. TS 2.1.4 has been rewritten to remove this conflict. The NRC staff disagrees with the second point and considers this information important to the proper operation of the cask system. Further, the changes made to this section resolve concerns regarding its complexity.

Comment No. 75: One commenter recommended relocating the figure attached to TS 3.2.1 to the TS Bases, because the purpose of the figure is to show where dose measurements should be taken.

Response: The NRC disagrees with this comment. This figure, now attached to TS 2.2.1, is an integral part of the proper implementation of this TS and assures that the dose measurements will be taken at the proper locations.

Comment No. 76: The commenter stated that the TSs do not comply with 10 CFR 72.44(d) that requires TSs on radioactive effluents.

Response: The NRC agrees with this comment. TS Section 3.0 has been revised to incorporate the requirements of 10 CFR 72.44(b).

Comment No. 77: One commenter recommended that within TS Section 1.1, the definition for "Intact Fuel Assembly" should be revised to state "..an amount of water greater than or equal to...," adding the term "greater than or" to allow greater flexibility with respect to dummy rod sizing.

Response: The NRC agrees with the comment and has revised the definition.

Comment No. 78: One commenter recommended that within TS Table 2.1-1, Item II.B should be reworded for clarification because the current wording could be misinterpreted by users that intact fuel assemblies are required to be loaded into damaged fuel containers.

Response: The NRC agrees with the comment. The table, which has been relocated to Appendix B, has been revised.

Comment No. 79: One commenter requested clarification of TS Section 4. As written, the text does not require a written report of the results of the first measurements, only "each cask subsequently loaded with a higher heat load." NRC's intent to require a written report for the first temperature measurements is not clear. The commenter further stated that it is not clear what "calculation" is being referred to in the last two sentences, whether it is the original design calculation or a new calculation generated from the test. The commenter further recommended the addition of "decay heat" after "lesser" and before "loads" in the last line.

Response: The NRC agrees with these comments, except for the recommendation to add the phrase "decay heat," which the NRC considers unnecessary. TS Section 3.3 has been revised to clarify the reporting requirements and the calculational comparison required by this TS condition.

Comment No. 80: One commenter recommended some editorial changes to revise TS Bases 2.2.2 and 2.2.3 to clarify that 10 CFR 72.75 has additional reporting requirements that may need to be met independent of these TS requirements.

Response: The NRC agrees with the comment. A reference to 10 CFR 72.75 has been added to Appendix B to the CoC.

Comment No. 81: One commenter recommended adding a new definition for fuel building to the TSs.

Response: The NRC agrees with the comment. A definition for fuel building has been added to the TSs.

Comment No. 82: One commenter recommended editorially revising TS LCO 3.1.7, "SFSC Lifting Requirements" and the related bases to clarify the applicability. The revision is necessary because the LCO is not intended to be applicable while the transport vehicle is in the fuel building or when the cask is secured on a railcar or heavy haul trailer because the cask is not being lifted.

Response: The NRC agrees with the comment. TS 2.1.3 has been revised accordingly.

Comment No. 83: One commenter recommended a revision to TS Tables 2.1-2 and 2.1-3, Note 1, for the purposes of clarification and to allow for manufacturer tolerances.

Response: The NRC agrees with the comment. The recommended changes to the tables have been made. The table has been relocated to Appendix B of the CoC.

Comment No. 84: One commenter recommended the revision of TS Table 3-1, Item 1.c, to change the lower helium tolerance to 10 percent because the smaller tolerances were associated with convection heat transfer, for which no credit is taken in the application.

Response: The NRC agrees with the comment and has revised renumbered TS Table 2-1.

Comment No. 85: One commenter recommended that TS 4.3.1 be revised to allow for changes to codes and standards because it would provide both the vendor and the NRC the flexibility to add exceptions/alternatives to the code without amending the certificate.

Response: The NRC agrees with the comment. Section 1.3.2 of Appendix B has been revised accordingly.

Comment No. 86: The applicant recommended in TS Section 4.4.6, the revision of the soil effective modulus of elasticity from "≤6,000psi" to "≤28,000 psi." In addition, the commenter recommended an acceptable method for licensees to comply with the soil modulus limit.

Response: The NRC agrees with the comment. The information has been added to Appendix B to the CoC.

Comment No. 87: One commenter recommended the addition of a third option to TS LCO 3.1.7 and Bases B3.1.7 (or elsewhere in the TSs) that allows general licensees to calculate site-specific lifting requirements based on the site-specific pad design and associated drop/tipover analyses.

Response: The NRC agrees with the comment. TS LCO 2.1.3 has been revised to add this option.

Comment No. 88: One commenter believed that the 48-hour time limit within TSs 3.1.1 through 3.1.6 is overly restrictive.

Response: The NRC agrees with this comment in part. Accordingly, the NRC has reviewed the time limit in each applicable TS. Some of the time limits have been extended to provide for a controlled, deliberate response to the LCO condition.

Comment No. 89: One commenter recommended the deletion of the Design Features, Section 4.6, Training Module, and Section 4.7, Pre-Operational Testing and Training Exercise because the review of the training program is required by 10 CFR 72.212(b)(6) and the TS duplicates the requirement in the regulation.

Response: The NRC agrees in part. The NRC agrees that there is duplication in the TSs and the regulatory requirements. Accordingly, TS 3.1 (previously Section 4.6) has been modified to reference the general licensee's systematic approach to training. However, the NRC staff believes that listing the training exercises as a specific requirement for proper cask operation is appropriate to be included in the TSs, and it has been maintained.

Comment No. 90: One commenter recommended adding "diesel" before "fuel" in TS Section 4.4.5 and in SER Sections 3.1.2.1.8, 4.3.4, and 4.4.3.4 for clarification.

Response: The NRC agrees conceptually with the comment. TS Section 4.4.5 (now 1.4.5 of Appendix B) and SER Sections 3.1.2.1.8, 4.3.4, and 4.4.3.4 have been revised to refer to combustible transporter fuel.

Comments on the Draft CoC

Comment No. 91: Two commenters recommended that CoC Condition 10 be revised to be consistent with 10 CFR 72.48 for the cask design and operating procedures. Another commenter stated that Condition 10 was not clear.

Response: The NRC agrees with the comments. The applicable CoC condition has been revised to delete the prescriptive controls for making changes to the cask design and operating procedures. The condition now reflects 10 CFR 72.48 as recently approved by the Commission.

Comment No. 92: Two commenters recommended that a Bases Control Program be added to the TSs or CoC.

Response: The NRC disagrees with the comment. The proposed TS bases are part of the SAR. Because 10 CFR 72.48 provides a change process for the SAR for control of the bases, there is no need to incorporate this program into the CoC or TSs.

Comment No. 93: One commenter requested information on the status of a petition for rulemaking on the change process in 10 CFR 72.48.

Response: This comment is beyond the scope of this rulemaking.

Comment No. 94: One commenter stated that the description of the attachment to the CoC was in error.

Response: The NRC agrees with this comment. The description has been corrected.

Comments on the NRC Staff's SER

Comment No. 95: One commenter asked a question about what is meant by the statement included in the NRC SER in Section 9.3 related to the examination and/or testing of the HI-STAR 100 by the applicant/certification holder/licensee.

Response: The SER refers to Section 9.1 of the applicant's SAR. This section summarizes the scope and acceptance criteria for the HI-STAR 100 test program. It includes fabrication and nondestructive examinations, weld inspecting, structural and pressure tests, leakage tests, component tests, and shielding and integrity testing and controls. The SAR or SER does not specify which entity must perform each test. This is because some tests are performed during fabrication, while others can only be performed after installation. The quality assurance programs implemented by the fabricator, certificate holder, or applicant with appropriate oversight will ensure that these SAR specified tests are completed and are effective. Further, the NRC inspection program also verifies on a sampling basis that tests and surveillances are conducted as required.

Comment No. 96: One commenter recommended revising the last sentence of the first paragraph of SER Section 3.1.2.1.6 to read: "The design-basis earthquake accelerations are assumed to be applied at the top of the ISFSI concrete pad with the resulting inertia forces applied at the HI-STAR 100 mass center."

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

Comment No. 97: One commenter recommended in SER Section 3.1.4.4, in the first paragraph, the replacement of "...the fabricator is an accredited facility by the ASME for nuclear fabrication work holding "N" and "NPT" stamps,...." with "...the HI-STAR 100 System is designed in accordance with the ASME Code, as clarified by the exceptions to the Code listed in TS Table 4-1."

Response: The NRC agrees with the comment. The SER has been revised. Note that the table is now in Appendix B.

Comment No. 98: One commenter recommended that in SER Section 6.3, the word "minimum" be replaced with "maximum" in the third sentence of the first full paragraph to match the analysis.

Response: The NRC agrees with the comment. The SER has been revised to correct the error.

Comment No. 99: One commenter stated that SER Section 8.1.4, which discusses the evaluation of welding and sealing procedures, should be revised to recognize the option of performing manual welding of the MPC lid closure weld in accordance with a user's as low as reasonably achievable (ALARA) practices.

Response: The NRC disagrees with the comment. As discussed in Sections 8.1 and 10.1 of the SAR, the use of the Automated Weld System provides justification that the HI-STAR 100 is designed in accordance with Part 72 radiological requirements and ALARA objectives consistent with Part 20. However, the intent of the proposed SER revision is already implied in Section 8.1.2 of the SER that states: "Each cask user will need to develop detailed loading procedures that incorporate the ALARA objectives of their site-specific radiation protection program." Therefore, each user can develop site-specific operating procedures based on ALARA objectives that would include the use of manual welding and make changes to the SAR in accordance with 10 CFR 72.48.

Comment No. 100: One commenter recommended that SER Section 8.3.1, which discusses the evaluation of cooling, venting, and reflooding during cask unloading operations, should be revised to allow the option of a once-through purge in lieu of the closed-loop cooling system.

Response: The NRC disagrees with this comment. An amendment application with a specific design and supporting analysis for a once-through helium cooling system would be required for NRC review and is beyond the scope of this rulemaking.

Comment No. 101: One commenter noted that a more appropriate method to implement the thermal test for the overpack had been accepted by the NRC for the HI-STAR-100 transportation cask and recommended this method be used for this cask design. Appropriate changes were recommended to be made to the SER and SAR.

Response: The NRC agrees that this method should be included in the SAR for the HI-STAR 100 storage cask. Appropriate changes have been made to Section 9.1.6 of the SAR and Chapter 9 of the SER.

Comment No. 102: The applicant submitted numerous editorial comments on the SAR, SER, and CoC. Comments were intended as clarification, restoration of deleted information, grammatical corrections, corrections to text, to maintain consistency between documents, typographical corrections, format changes, and to correct terminology. These editorial changes do not change the design of the cask or supporting analysis.

Response: The NRC agrees with many of the editorial comments suggested by Holtec International. The SAR, SER, and CoC have been revised to address the comments as appropriate.

<u>Comments on the Applicant's Topical SAR</u>

Note: In response to comments received, a number of changes to the SAR were made by Holtec International, as discussed below.

Comment No. 103: One commenter proposed a revision to the language in Section 8.0 of the SAR to clarify that users will have some flexibility to use procedures and equipment suitable for site-specific needs and capabilities.

Response: The NRC agrees with the suggested editorial changes. The changes to the SAR have been made.

Comment No. 104: One commenter recommended some editorial changes within SAR Section 4.4, because the wording in Subsection 4.1.1.15 may be erroneously interpreted to mean that the chilled helium delivered to the MPC cavity to cool the internals prior to flooding the cavity with water must be at 100°F. The commenter stated that the text of the SAR requires clarification to permit each cask user's cooldown system to be engineered with the flexibility to cool MPCs containing fuel with varying levels of decay heat production.

Response: The NRC agrees with the comment. The SAR has been revised.

Comment No. 105: In SAR Section 1.5, Drawings 1399, Sheet 3, and BM-1476, and in Drawing Section "N-N," one commenter recommended the addition of four threaded holes spaced 90 degrees apart as a personnel dose reduction enhancement. The new holes would allow the personnel attaching the shield to work in an area of lesser exposure to radiation within the same time frame. The effect of the shield attachment will remain the same.

Response: The NRC agrees with the comment. Drawings 1399 and BM-1476 have been revised to reflect the change.

Comment No. 106: One commenter suggested that in SAR Revision 10, the drawings in Chapter 1 be revised to match those approved by the NRC in the transportation SAR.

Response: The NRC agrees with the comment. Seven drawings in SAR Section 1 have been revised to match those in the transportation SAR. Although four drawings have not been revised to match the transportation SAR, this is acceptable to the NRC staff because they reflect storage design features.

Comment No. 107: In the SAR, one commenter (the applicant) recommended changing Section 6.1 by replacing "(20° C - 100°)" with "(i.e., water density of 1.000 g/cc)" and delete "(20° C assumed)" to more accurately describe the assumption made in the analyses.

Response: The NRC agrees. The SAR has been revised as suggested by the commenter.

Comment No. 108: The applicant suggested a number of changes to the drawings for the HI-STAR 100 Storage Cask. These changes did not require a change to the supporting design analyses.

Response: The NRC agrees that the changes to the drawings were appropriate and do not result in any changes to the supporting design analyses. The SAR drawings have been revised in accordance with the suggested changes.

Comment No. 109: The applicant suggested using Magnetic Particle Examination in lieu of Liquid Penetrant Examination for the overpack weld examination and recommended changes to the associated drawing notes.

Response: The NRC agrees with this suggested change. The NRC agrees that resolution of this comment will involve a change to the drawings which will mean that drawings referencing this examination shall be different for the storage and transportation certificates. These differences are not significant because the staff finds Magnetic Particle Examination to be equally acceptable to Liquid Penetrant Examination. Appropriate changes to the drawings have been made.

Comment No. 110: The applicant suggested a clarification for the sequence for the hydrostatic testing and helium leakage testing during fabrication of the overpack.

Response: The NRC agrees with the suggested change. The SAR has been revised accordingly.

Comment No. 111: As it relates to the Radiography and Heat Treatment requirements for the containment boundary of the HI-STAR overpack, the applicant requested that post weld heat treatment (PWHT), after completing nondestructive examination, be used for all overpack containment boundary welds which require an exception from the ASME code.

Response: The NRC agrees. The SAR and Appendix B to the CoC have been modified appropriately.

Comment No. 112: The applicant suggested a revision to the drawings in the SAR to reflect the localized thinning tolerance in the containment shell.

Response: The NRC staff agrees with the suggested revision. However, the applicant did not provide the suggested changes in its final revisions to the SAR. The initial drawings remain acceptable.

Comment No. 113: One commenter (the applicant) recommended that changes to Technical Specification Table 4-1, MPC Enclosure Vessel and Lid, should be made to replace "and sufficient intermediate layers to detect critical wild flaws" with "and at least one intermediate PT after approximately 3/8 inch weld depth." The commenter also recommended the deletion of "Flaws in austenitic stainless are not expected to exceed the bead". The commenter further recommended several changes to the SER as follows: SER Section 8.1.4 should be changed to add "(or optional multi-layer PT examination)," after "ultrasonic examination (UT)"; the SER should recognize that users may choose to perform the MPC void-to-shell weld manually; and SER Section 11.4.1.3.1 should be reworded to read "examined using UT or multi-layer PT techniques," instead of "volumetrically examined using UT."

Response: The NRC agrees and notes that the applicant's comments with respect to TS Table 4-1 have been superseded by its latest revision to the SAR. Changes have been made to Table 1-3 to Appendix B. The SER has been revised as recommended.

Summary of Final Revisions

The NRC staff modified the listing for the Holtec International HI-STAR 100 cask system within 10 CFR 72.214, "List of approved spent fuel storage casks," with respect to the title of the SAR as well as the CoC and its two appendices, the TSs, and the Approved Contents and Design Features. The NRC staff has also modified its SER.

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 Stan Turel, Office of Nuclear Material

Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 415-6234, e-mail spt@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-STAR 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 rulemaking 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.

2. In Section 72.214, Certificate of Compliance 1008 is added to read as follows:

§ 72.214 List of approved spent fuel storage casks.

* * * * *

Certificate Number: 1008

SAR Submitted by: Holtec International

SAR Title: HI-STAR 100 Cask System Topical Safety Analysis Report

Docket Number: 72-1008

Certification Expiration Date: (20 years after final rule effective date)

Model Number: HI-STAR 100

Dated at Rockville, Maryland, this 23rd day of August ,1999.

For the Nuclear Regulatory Commission.

William Travers

William D. Travers,
Executive Director for Operations.

ATTACHMENT 2

Congressional Letters

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

The Honorable Joe L. Barton, Chairman
Subcommittee on Energy and Power
Committee on Commerce
United States House of Representatives
Washington, DC 20515

Dear Mr. Chairman:

The U.S. Nuclear Regulatory Commission (NRC) intends to publish a final rule in the Federal Register that will amend the "List of approved spent fuel storage casks" (10 CFR 72.214). NRC is approving the Holtec International HI-STAR 100 cask system for storage of spent fuel under the conditions specified in the Certificate of Compliance (CoC). This cask, when used in accordance with the conditions specified in the CoC and NRC regulations, will meet the requirements of 10 CFR Part 72; thus, adequate protection of the public health and safety would be ensured. This cask is being listed under 10 CFR 72.214, "List of approved spent fuel storage casks" to allow holders of power reactor operating licenses to store spent fuel in this cask system, under a general license. Further, the NRC has approved a complementary application of this cask system for use in transporting spent fuel under 10 CFR Part 71. The CoC will terminate 20 years after the effective date of the final rule listing the cask in 10 CFR 72.214, unless the cask's CoC is renewed. The certificate contains conditions for use specific for this cask, addressing issues such as operating procedures, training, and spent fuel specification.

Sincerely,

Dennis K. Rathbun, Director
Office of Congressional Affairs

Enclosure:
Federal Register Notice

cc: Representative Ralph M. Hall

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

The Honorable James N. Inhofe, Chairman
Subcommittee on Clean Air, Wetlands, Private
Property and Nuclear Safety
Committee on Environment and Public Works
United States Senate
Washington, DC 20510

Dear Mr. Chairman:

The U.S. Nuclear Regulatory Commission (NRC) intends to publish a final rule in the Federal Register that will amend the "List of approved spent fuel storage casks" (10 CFR 72.214). NRC is approving the Holtec International HI-STAR 100 cask system for storage of spent fuel under the conditions specified in the Certificate of Compliance (CoC). This cask, when used in accordance with the conditions specified in the CoC and NRC regulations, will meet the requirements of 10 CFR Part 72; thus, adequate protection of the public health and safety would be ensured. This cask is being listed under 10 CFR 72.214, "List of approved spent fuel storage casks" to allow holders of power reactor operating licenses to store spent fuel in this cask system, under a general license. Further, the NRC has approved a complementary application of this cask system for use in transporting spent fuel under 10 CFR Part 71. The CoC will terminate 20 years after the effective date of the final rule listing the cask in 10 CFR 72.214, unless the cask's CoC is renewed. The certificate contains conditions for use specific for this cask, addressing issues such as operating procedures, training, and spent fuel specification.

Sincerely,

Dennis K. Rathbun, Director
Office of Congressional Affairs

Enclosure:
Federal Register Notice

cc: Senator Bob Graham

The Honorable Joe L. Barton, Chairman
Subcommittee on Energy and Power
Committee on Commerce
United States House of Representatives
Washington, DC 20515

Dear Mr. Chairman:

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Sincerely,

Dennis K. Rathbun, Director
Office of Congressional Affairs

Enclosure:
Federal Register Notice
cc: Representative Ralph M. Hall

Identical Letter sent to The Honorable James M. Inhofe with cc: to Senator Bob Graham

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ATTACHMENT 3

Daily Staff Notes

DAILY STAFF NOTES

OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

Final Rule Signed by EDO

On August 23, 1999, the Executive Director for Operations approved a final rule that amends 10 CFR Part 72.214, List of approved spent fuel storage casks, by adding the Holtec HI-STAR 100 cask system to the list of approved spent fuel storage casks. This amendment will allow the holders of power reactor operating licenses to store spent fuel in the approved cask under a general license.

This notice informs the Commission that, in accordance with the rulemaking authority delegated to the EDO, the EDO has signed this final rule and proposes to forward it on 8/31/99 to the Office of the Federal Register for publication, unless otherwise directed by the Commission.

ATTACHMENT 4

Approved for Publication

Approved For Publication

The Commission delegated to the EDO (10 CFR 1.31(c)) the authority to develop and promulgate rules as defined in the APA (5 U.S.C. 551 (4)) subject to the limitations in NRC Management Directive 9.17, Organization and Functions, Office of the Executive Director for Operations, paragraphs 0213, 038, 039, and 0310.

The enclosed final rule, entitled "List of Approved Spent Fuel Storage Casks: (HI-STAR 100) Addition," amends 10 CFR Part 72 to add the Holtec International HI-STAR 100 cask system to the list of approved spent fuel storage casks. This amendment will allow the holders of power reactor operating licenses to store spent fuel in the approved cask under a general license.

This final rule does not constitute a significant question of policy, nor does it amend regulations contained in 10 CFR Parts 7, 8, or 9 Subpart C concerning matters of policy. I, therefore, find that this rule is within the scope of my rulemaking authority and am proceeding to issue it.

August 23, 1999
Date

William Travers

William D. Travers,
Executive Director for Operations

ATTACHMENT 5

Environmental Assessment

ENVIRONMENTAL ASSESSMENT AND FINDING OF
NO SIGNIFICANT IMPACT
ON
AMENDMENT TO 10 CFR PART 72

LIST OF APPROVED SPENT FUEL STORAGE CASKS: (HI-STAR 100) ADDITION

Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
July 1999

I. THE PROPOSED ACTION

The proposed action amends 10 CFR Part 72 to add the Holtec International HI-STAR 100 cask system to the list of NRC-approved casks. The proposed action will provide a greater selection of NRC-approved casks for the storage of spent nuclear fuel at commercial nuclear power reactor sites under a general license without the need for additional site-specific approvals. These casks can be relied on to provide safe confinement of spent fuel at any reactor site when used in accordance with their certificates of compliance. In order to use an NRC-approved cask, the reactor licensee must ensure that the reactor site parameters and potential site-boundary doses are within the scope of the cask safety analysis report and reactor license.

II. THE NEED FOR THE PROPOSED ACTION

This rule is needed to add a cask to the "List of approved spent fuel storage casks" in 10 CFR 72.214. Holtec International has requested a certificate of compliance for the Holtec International HI-STAR 100 cask system in accordance with the procedures in 10 CFR Part 72, Subpart L, for obtaining NRC approval of new spent fuel storage cask designs. The NRC has completed a safety evaluation report for the cask and based upon that evaluation has determined that commercial nuclear power reactors will be able to use the cask design under a general license after the cask system is listed in 10 CFR 72.214.

III. ENVIRONMENTAL IMPACTS OF THE PROPOSED ACTION

There are over 30 years of experience with dry storage of spent fuel in the United States and other countries. The environmental impacts associated with storage of light water reactor (LWR) spent fuel (including dry storage) have been previously considered in other NRC rulemakings and licensing actions on which this assessment is tiered. In a proceeding entitled "Review and Final Revision of Waste Confidence Decision," published in the Federal Register on September 18, 1990 (55 FR 38474), the NRC found "reasonable assurance that, if necessary, spent fuel generated in any reactor can be stored safely and without significant environmental impacts for at least 30 years beyond the licensed life for operation of that reactor at its spent fuel storage basin, or at either onsite or offsite independent spent fuel storage installations." The "Environmental Assessment for 10 CFR Part 72 'Licensing Requirements for the Independent Storage of Spent Fuel and High-Level Radioactive Waste,' " NUREG-1092[1] (August 1984), and the Supplementary Information of a proposed rule published in the Federal Register on May 27, 1986 (51 FR 19106), contain specific analyses showing that the potential environmental impacts from dry storage of spent fuel in casks are small. The "Environmental Assessment for Proposed Rule Entitled 'Storage of Spent Nuclear Fuel in NRC-Approved Storage Casks at Nuclear Power Reactor Sites" for the proposed rule published in the Federal

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[1] Copies of NUREG-1092 may be purchased from the Superintendent of Documents, U.S. Government Printing Office, P. O. Box 37082, Washington, DC 20013-7082. Copies are also available from the National Technical Information Service, 5285 Port Royal Road, Springfield, VA 22161. A copy is also available for inspection and/or copying at the NRC Local Public Document Room, 2120 L Street, NW. (Lower Level), Washington, DC.

Register on May 5, 1989 (54 FR 19379), discussed the environmental impact of dry cask storage and the finding of no significant impact.

The major nonradiation environmental impacts for dry cask storage of spent fuel would be those related to fabrication of the casks. The steel required for these casks is expected to have very little impact on the steel industry. The amounts of lead and iron needed would not have significant incremental impacts on the mining and use of these metals. For concrete casks, the amount of concrete required would be small compared to industrial and construction uses. The amount of plastic, most commonly polyethylene used as a neutron shield, would not be more than about a ton per cask and would be insignificant compared to the millions of tons produced annually.

Incremental impacts caused by the operation of dry cask storage of spent fuel under a general license are not considered significant. No effluents are expected from the sealed dry storage casks. However, activities associated with cask loading and decontamination may result in some small incremental liquid and gaseous effluent. These operations will be conducted under 10 CFR Part 50 reactor operating licenses, and effluents will be controlled to be within existing reactor technical specifications. Because of the relatively large reactor sites, any incremental doses offsite due to direct radiation exposure from the spent fuel storage casks are expected to be small and when combined with the contribution from reactor operations will be well within the 0.25 mSv/yr (25 mrem/yr) limit to the whole body specified in 10 CFR 72.104. Incremental impacts in collective occupational exposure due to dry cask storage of spent fuel under a general license are expected to be only a small fraction of that occurring from operation of the nuclear power station.

During the promulgation of the amendments adding the new Subpart K to 10 CFR Part 72 (55 FR 29181; July 18, 1990), the NRC staff assessed public health consequences of dry cask storage accidents. The NRC staff also assessed public health consequences from acts of radiological sabotage and concluded that, to be successful, it would have to be carried out with the aid of explosives. Public health consequences from an explosive sabotage event would stem almost exclusively from the release of respirable particles. In an NRC study, an experiment was carried out to evaluate the effects of a severe, perfectly executed sabotage scenario against a simulated storage cask containing spent fuel assemblies. The whole-body dose to an offsite individual was calculated based on the release data and found to about 10 mSv (1 rem). The experiment and calculations led to the conclusion of low public health consequences. As a result of these evaluations, the NRC staff determined that because of the physical characteristics of the storage casks and the conditions of storage that include specific security provisions, the potential risk to public health and safety due to accidents or sabotage is extremely small.

Decommissioning dry cask spent fuel storage under a general license would be carried out as part of the power reactor site decommissioning plan. It would consist of removing the spent fuel from the site and decontaminating cask surfaces. The casks would then be released for re-use or disposal. No residual contamination is expected to be left behind on supporting structures. The incremental cost associated with decommissioning is expected to represent a small fraction of the cost of decommissioning an entire nuclear power station.

Because this amendment to 10 CFR Part 72 will not change the existing safety and environmental requirements for the storage of spent nuclear fuel, and because dry cask spent fuel storage under a general license will still have to meet these requirements, no change in environmental impact is anticipated. In previous rulemaking proceedings, the NRC determined that compliance with the requirements of 10 CFR Part 72 would ensure adequate protection of public health and safety. The NRC, through a safety evaluation report for the cask in this rulemaking, has determined that if the conditions specified in the certificate of compliance are met, adequate protection of public health and safety will be maintained. Based on the above assessment, the NRC finds that adding the Holtec International HI-STAR 100 dry spent fuel storage cask to the list of approved storage casks will not have a significant environmental impact.

IV. ALTERNATIVES TO THE FINAL ACTION (RULEMAKING)

The alternative to this proposed action is to withhold generic approval of this new design and require a site-specific licensing proceeding for each utility proposing to use this cask system. Although this would involve a different process for approving the cask design, the environmental impacts of approving this cask design would be the same. In light of this consideration, and given the insignificance of the environmental impacts, implementation of the proposed action is reasonable.

The NWPA directed that the NRC approve one or more technologies that have been developed and demonstrated by DOE for the use of spent fuel storage at the sites of civilian nuclear power reactors without the need for additional site-specific review to the extent practicable. The NWPA also directed that the NRC set forth procedures for licensing the technology by rulemaking. Regulations for accomplishing this are in place. Therefore, the no action alternative is unacceptable.

V. ALTERNATIVE USE OF RESOURCES

The only irreversible commitments of resources determined in this assessment were those materials needed for the casks.

VI. AGENCIES AND PERSONS CONTACTED

No agencies or persons outside the NRC were contacted in connection with the preparation of this environmental assessment.

VII. FINDING OF NO SIGNIFICANT IMPACT

Based on the foregoing environmental assessment, the NRC concludes that this rulemaking entitled "List of Approved Spent Fuel Storage Casks: (HI-STAR 100) Addition" will not have a significant incremental effect on the quality of the human environment. Therefore, the NRC has determined that an environmental impact statement is not necessary for this rulemaking.

Certain documents related to this rulemaking, including comments received by the NRC, may be examined at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC. These same documents also may be viewed and downloaded electronically via the interactive rulemaking website established by NRC for this rulemaking.

ATTACHMENT 6

Proposed Certificate of Compliance
and the
Preliminary Safety Evaluation Report

CERTIFICATE OF COMPLIANCE FOR SPENT FUEL STORAGE CASKS

The U.S. Nuclear Regulatory Commission is issuing this Certificate of Compliance pursuant to Title 10 of the Code of Federal Regulations, Part 72, "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (10 CFR Part 72). This certificate is issued in accordance with 10 CFR 72.238, certifying that the storage design and contents described below meet the applicable safety standards set forth in 10 CFR Part 72, Subpart L, and on the basis of the Final Safety Analysis Report (FSAR) of the cask design. This certificate is conditional upon fulfilling the requirements of 10 CFR Part 72, as applicable, and the conditions specified below.

Certificate No.	Effective Date	Expiration Date	Docket Number	Amendment No.	Amendment Date	Package Identification No.
1008			72-1008			USA/72-1008

Issued To: (Name/Address)

Holtec International
Holtec Center
555 Lincoln Drive West
Marlton, NJ 08053

Safety Analysis Report Title

Holtec International Inc., Safety Analysis Report for the HI-STAR 100 Cask System, Revision 10
Holtec Report HI-941184

Docket No. 72-1008

CONDITIONS

1. CASK

The HI-STAR 100 Cask System is certified as described in the Safety Analysis Report (SAR) and in NRC's Safety Evaluation Report (SER) accompanying the Certificate of Compliance. It is designed for both storage and transfer of irradiated nuclear fuel.

a. Model No.: HI-STAR 100 (MPC-24, MPC-68, MPC-68F)

The HI-STAR 100 Cask System is comprised of the multi-purpose canister (MPC), which contains the fuel, and the overpack which contains the MPC. The two digits after the MPC designate the number of reactor fuel assemblies for which the respective MPCs are designed. The MPC-24 is designed to contain up to 24 pressurized water reactor (PWR) fuel assemblies. The MPC-68 is designed to contain up to 68 boiling water reactor (BWR) fuel assemblies. Any MPC-68 containing fuel assemblies with known or suspected defects, such as ruptured fuel rods, severed rods, loose fuel pellets, or which cannot be handled by normal means due to fuel cladding damage, is designated as MPC-68F. The MPC-24 and the MPC-68 (including the MPC-68F) are identical in external dimensions and will fit into the same overpack design.

b. Description

The complete HI-STAR 100 Cask System for storage of spent nuclear fuel is comprised of two discrete components: the MPC and the storage/transport overpack. The HI-STAR 100 Cask System consists of interchangeable MPCs which constitute the confinement boundary for BWR or PWR spent nuclear fuel and an overpack which provides the helium retention boundary, gamma and neutron radiation shielding, and heat rejection capability. All MPCs have identical exterior dimensions which render them interchangeable. A single overpack design is provided which is capable of storing each type of MPC.

The HI-STAR 100 MPCs are welded cylindrical structures with flat ends. Each MPC is an assembly consisting of a honeycombed fuel basket, a baseplate, canister shell, a lid, and a closure ring. The outer diameter and cylindrical height of each MPC is fixed. However, the number of spent nuclear fuel storage locations in each of the MPCs depends on the fuel assembly characteristics. The MPC provides the confinement boundary for the stored fuel. The confinement boundary is a seal-welded enclosure constructed entirely of a stainless steel alloy. The inner surfaces of the HI-STAR 100 overpack form an internal cylindrical cavity for housing the MPC. The outer surface of the overpack inner shell is buttressed with intermediate shells of gamma shielding.

The fuel transfer and auxiliary equipment necessary for Independent Spent Fuel Storage Installation operation are not included as part of the HI-STAR 100 Cask System reviewed for a Certificate of Compliance under 10 CFR Part 72, Subpart L. Such equipment may include, but is not limited to, special lifting devices, transfer trailers or equipment, and vacuum drying/helium leak test equipment.

2. OPERATING PROCEDURES

Written operating procedures shall be prepared for cask handling, loading, movement, surveillance, and maintenance. The user's site-specific written operating procedures shall be consistent with the technical basis described in the SAR.

3. QUALITY ASSURANCE

Activities in the areas of design, procurement, fabrication, assembly, inspection, testing, operation, maintenance, repair, modification of structures, systems and components, and decommissioning that are important to safety shall be conducted in accordance with a Commission-approved quality assurance program which satisfies the applicable requirements of 10 CFR Part 72, Subpart G, and which is established, maintained, and executed with regard to the cask system.

CERTIFICATE OF COMPLIANCE
FOR SPENT FUEL STORAGE CASKS
Supplemental Sheet

4. HEAVY LOADS REQUIREMENTS

Each lift of a HI-STAR 100 spent fuel storage cask must be made in accordance with the existing heavy loads requirements and procedures of the licensed facility in which the lift is made. A plant-specific safety review (in accordance with 10 CFR 50.59 or 10 CFR 72.48, if applicable) is required to show operational compliance with existing plant-specific heavy loads requirements.

5. APPROVED CONTENTS

Contents of the HI-STAR 100 Cask System must meet the fuel specifications given in Appendix B to this certificate.

6. APPROVED DESIGN FEATURES

Features or characteristics for the site or cask must be in accordance with Appendix B to this certificate.

7. CHANGES TO THE CERTIFICATE OF COMPLIANCE

The holder of this certificate who desires to make changes to the certificate, which includes Appendix A (Technical Specifications) and Appendix B (Approved Contents and Design Features), shall submit an application for amendment of the certificate.

8. AUTHORIZATION

The HI-STAR 100 Cask System, which is authorized by this certificate, is hereby approved for general use by holders of 10 CFR Part 50 licenses for nuclear reactors at reactor sites under the general license issued pursuant to 10 CFR 72.210, subject to the conditions specified by 10 CFR 72.212, and the attached Appendix A and Appendix B.

FOR THE NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Attachments: 1. Appendix A
2. Appendix B

ATTACHMENT 7

Press Release

DRAFT PRESS RELEASE

NRC AMENDS REGULATIONS TO ADD

HI-STAR FUEL STORAGE CASK DESIGN TO APPROVED LIST

The Nuclear Regulatory Commission is amending its regulations to add an additional fuel storage cask design to those that utilities can use–under a general license and without site-specific approval–to store spent fuel at their nuclear power plants.

The new design is the Holtec International Hi-Star 100 cask system (Hi-Star), manufactured by Holtec International Inc., of Marlton, NJ. It can contain up to 24 pressurized water reactor fuel assemblies or 68 boiling water reactor fuel assemblies.

Under the terms of an NRC general license, any nuclear power reactor licensee can use a pre-approved cask if the company notifies the NRC in advance, meets the conditions of the cask's NRC certificate of compliance (CoC) and complies with NRC's regulations, including a requirement to ensure that the reactor site characteristics and potential site-boundary radiation doses are within the scope of the cask's safety analysis report and the reactor license.

Eight cask designs have previously been approved for use under a general license. The Hi-Star certificate contains conditions for use that are similar to those for other NRC-approved casks. However, the certificate for each cask design may differ in some specifics, such as operating procedures, training exercises, and spent fuel specifications.

The NRC staff has issued a preliminary safety evaluation report which finds that, if the conditions specified in the CoC are met, adequate protection of public health and safety will be maintained. The staff's environmental assessment determined that use of the Hi-Star cask design on reactor sites would have no significant incremental impacts on the environment.

The certificate would expire in 20 years unless it is renewed.

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ATTACHMENT 8

SBREFA Forms

Submission of Federal Rules
Under the Congressional Review Act

☐ **President of the Senate** ☐ **Speaker of the House of Representatives** ☐ **GAO**

Please fill the circles electronically or with black pen or #2 pencil.

1. Name of Department or Agency	2. Subdivision or Office
U.S. Nuclear Regulatory Commission	**NMSS**

3. Rule Title

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

4. Regulation Identifier Number (RIN) or Other Unique Identifier (if applicabl e)

RIN 3150-AG 17

5. Major Rule ○ Non-major Rule ◉

6. Final Rule ◉ Other ○ ______________________

7. With respect to this rule, did your agency solicit public comments? Yes ◉ No ○ N/A ○

8. Priority of Regulation (fill in one)

◉ Economically Significant; or
Significant; or
Substantive, Non Significant

○ Routine and Frequent or
Informational/Administrative/Other
(Do not complete the other side of this form
if filled in above.)

9. Effective Date (if applicable)

10. Concise Summary of Rule (fill in one or both) attached ○ stated in rule ◉

Submitted by: ______________________ (signature)

Name: ______________________

Title: ______________________

For Congressional Use Only:

Date Received: ______________

Committee of Jurisdiction: ______________________

3/23/99

	Yes	No	N/A
A. With respect to this rule, did your agency prepare an analysis of costs and benefits?	●	○	○
B. With respect to this rule, by the final rulemaking stage, did your agency			
1. certify that the rule would not have a significant economic impact on a substantial number of small entities under 5 U.S.C. § 605(b)?	●	○	○
2. prepare a final Regulatory Flexibility Analysis under 5 U.S.C. § 604(a)?	●	○	○
C. With respect to this rule, did your agency prepare a written statement under § 202 of the Unfunded Mandates Reform Act of 1995?	○	○	●
D. With respect to this rule, did your agency prepare an Environmental Assessment or an Environmental Impact Statement under the National Environmental Policy Actg (NEPA)?	●	○	○
E. Does this rule contain a collection of information requiring OMB approval under the Paperwork Reduction Act of 1995?	○	●	○
F. Did you discuss any of the following in the preamble to the rule?	○	●	○
• E.O. 12612, Federalism	○	○	○
• E.O. 126630, Government Actions and Interference with Constitutionally Protected Property Rights	○	○	○
• E.O. 12866, Regulatory Planning and Review	○	○	○
• E.O. 12875, Enhancing the Intergovernmental Partnership	○	○	○
• E.O. 12988, Civil Justice Reform	○	○	○
• E.O. 13045, Protection of Children from Environmental Health Risks and Safety Risks	○	○	○
• Other statutes or executive orders discussed in the preamble concerning the rulemaking process (please specify)			

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Submission of Federal Rules Under the Congressional Review Act

☐ **President of the Senate** ☐ **Speaker of the House of Representatives** ☐ **GAO**

Please fill the circles electronically or with black pen or #2 pencil.

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RIN 3150-AG 17

5. Major Rule ○ Non-major Rule ●

6. Final Rule ● Other ○ ______________________

7. With respect to this rule, did your agency solicit public comments? Yes ● No ○ N/A ○

8. Priority of Regulation (fill in one)

● Economically Significant; or
Significant; or
Substantive, Non Significant

○ Routine and Frequent or
Informational/Administrative/Other
(Do not complete the other side of this form if filled in above.)

9. Effective Date (if applicable)

10. Concise Summary of Rule (fill in one or both) attached ○ stated in rule ●

Submitted by: ______________________ (signature)

Name: ______________________

Title: ______________________

For Congressional Use Only:

Date Received: ____________

Committee of Jurisdiction: ______________________

3/23/99

	Yes	No	N/A
A. With respect to this rule, did your agency prepare an analysis of costs and benefits?	●	○	○
B. With respect to this rule, by the final rulemaking stage, did your agency			
1. certify that the rule would not have a significant economic impact on a substantial number of small entities under 5 U.S.C. § 605(b)?	●	○	○
2. prepare a final Regulatory Flexibility Analysis under 5 U.S.C. § 604(a)?	●	○	○
C. With respect to this rule, did your agency prepare a written statement under § 202 of the Unfunded Mandates Reform Act of 1995?	○	○	●
D. With respect to this rule, did your agency prepare an Environmental Assessment or an Environmental Impact Statement under the National Environmental Policy Actg (NEPA)?	●	○	○
E. Does this rule contain a collection of information requiring OMB approval under the Paperwork Reduction Act of 1995?	○	●	○
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• E.O. 12612, Federalism	○	○	○
• E.O. 126630, Government Actions and Interference with Constitutionally Protected Property Rights	○	○	○
• E.O. 12866, Regulatory Planning and Review	○	○	○
• E.O. 12875, Enhancing the Intergovernmental Partnership	○	○	○
• E.O. 12988, Civil Justice Reform	○	○	○
• E.O. 13045, Protection of Children from Environmental Health Risks and Safety Risks	○	○	○
• Other statutes or executive orders discussed in the preamble concerning the rulemaking process (please specify)			

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