AG17-2 PDR

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

August 9, 1999

MEMORANDUM TO: Donald A. Cool, Director
Division of Industrial and Medical Nuclear Safety
Office of Nuclear Reactor Regulation

Valeria H. Wilson

FROM: Valeria H. Wilson, Director
Division of Administrative Services
Office of Administration

SUBJECT: OFFICE CONCURRENCE ON FINAL RULE PACKAGE ENTITLED, "LIST OF APPROVED SPENT FUEL STORAGE CASKS: (HI-STAR 100) ADDITION"

The Office of Administration concurs on the final rule that amends Part 72. We have attached a copy of the package that presents our comments.

When this document is forwarded for publication, please include a 3.5 inch diskette that contains a copy of the document in WordPerfect as part of the transmittal package. The diskette will be forwarded to the OFR and the Government Printing Office for their use in typesetting the document.

In order to assist you in preparing the list of documents centrally relevant to this final rule that is required by NRC's regulatory history procedures, you should place the designator "AG17" in the upper right-hand corner of each document concerning the final rule that you forward to the Nuclear Documents System.

If you have any questions, please contact David Meyer, 415-7162 (DLM1), or Michael Harrison, 415-6865 (PMH), of the Office of Administration.

Attachment: As stated

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

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: Carl J. Paperiello, Director
Office of Nuclear Material Safety
and Safeguards

SUBJECT: FINAL RULE TO AMEND 10 CFR PART 72 LIST OF APPROVED SPENT FUEL STORAGE CASKS

Attached for your signature is a final rule (Attachment 1) amending Nuclear Regulatory Commission (NRC) regulations, by adding a cask system to the "List of Approved Spent Fuel Storage Casks." This amendment ~~would~~ allow the holders of power reactor operating licenses will to store spent fuel in the Holtec International Hi-Star 100 cask system under a general license.

Background: On January 11, 1999 (64 FR 1542), the NRC published a proposed rule to add the Holtec 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, both the SER and CoC for the Holtec International HI-STAR 100 cask system have been modified. The NRC finds that the cask system, as designed and when fabricated and used in accordance with the conditions specified in its CoC, meets the requirements of 10 CFR Part 72; thus, adequate protection of public is health and safety ~~would be~~ ensured. ~~With this final rule,~~ NRC is approving the use of the Holtec International HI-STAR 100 cask system under the general license in 10 CFR Part 72, Subpart K, by holders of power reactor operating licenses under 10 CFR Part 50. Simultaneously, the NRC will issue a final SER and CoC that will be effective on the effective date of the final rule.

Notices: The appropriate Congressional committees will be notified (Attachment 2). A notice to the Commission that the Executive Director for Operations has signed the attached Federal Register notice is attached for inclusion in the "Weekly Report to the Commission" (Attachment 3). The "Approved for Publication" (Attachment 4), the Environmental Assessment (Attachment 5), the SBREFA forms (Attachment 6), and the Press Release (Attachment 7) are also attached.

CONTACT: Stan Turel, NMSS/IMNS
(301) 415-6234

Phil Brochman, NMSS/SFPO
(301) 415-8592

Resources: No additional resources will be needed to implement this rule.

Coordination: The Offices of Administration, Enforcement, and Nuclear Reactor Regulation concur with these amendments. The Office of the General Counsel has no legal objection. The Office of the Chief Financial Officer has reviewed the final rule for resource implications and has no objections. The Office of the Chief Information Officer has reviewed the final rule for information technology and information management implications and concurs in it.

Attachments:
1. FRN of Final Rulemaking
2. Congressional Letters
3. Weekly Report to the Commission
4. "Approved for Publication"
5. Environmental Assessment
6. SBREFA Forms
7. Press Release

[7590-01-P]

NUCLEAR REGULATORY COMMISSION

10 CFR Part 72

RIN 3150-AG17

(HI-STAR 100)

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

Summary of Public Comments on the Proposed Rule

The NRC received nine comment letters on the proposed rule. The commenters included the applicant, the State of Utah, an individual member of the public, industry representatives, and several utilities. Copies of the public comments are available for review in the NRC Public Document Room, 2120 L Street, NW (Lower Level), Washington, DC 20003-1527.

Comments on Direct Final Rule

As part of the proposed rule~~making~~, the NRC staff requested public comment on a Direct Final Rulemaking process for future amendments to the list of approved spent fuel storage casks in 10 CFR 72.214. Direct final rulemaking is a technique for expediting the issuance of noncontroversial rules. Under this procedure, the NRC would publish the proposed amendment to the list as both a proposed and final rule in the Federal Register simultaneously. A direct final rule would normally become effective 75 days after publication in the Federal Register unless the NRC receives significant adverse comments on the direct final rule with 30 days after publication. If significant adverse comments are received, the NRC withdraws the direct final rule and addresses the comments received as comments on the proposed rule and will subsequently issue a final rule.

Comment: One commenter was in support 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 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 being able to affect the outcome of rulemaking procedures. One of these commenters believes that, given past problems with the casks, future approval should be subject to adequate and rigorous public scrutiny. Those opposed also believe 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 to cause the NRC to withdraw the published final rule. The two commenters did not believe that the addition to or revision of the listing was either noncontroversial or routine as evidenced by the number of comments they had on the Holtec HI-STAR 100 proposed rule.

Response: Based on the significant number of comments that were received on the Holtec HI-STAR 100 proposed rule, including a number of comments from the applicant, it appears that the direct final rule approach cannot 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. Amendments to the CoCs result in a rulemaking to revise the listing in 10 CFR 72.214. The NRC anticipates that the amendments to revise the listing will be noncontroversial because the cask design and analysis will have gone through the public comment process for the initial ~~Certificate~~ CoC listing and the revision will be limited to the subject of the amendment. The NRC is considering using a direct final rule for amendments to the cask systems in the listing.

Comments on the Holtec International HI-STAR 100 Cask System

Both the SER and CoC for the Holtec International HI-STAR 100 cask system were modified in response to public comments. The listing of the Holtec International HI-STAR 100 cask system within § 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:

One commenter recommends that the Holtec HI-STAR 100 design should not be approved, because Holtec has not provided reasonable assurance that the cladding and cask will retain their integrity under normal, off-normal, and accident conditions. Moreover, the commenter added that Holtec had not correctly calculated health impacts under bounding accidents ~~nor~~ or evaluated the impact of a sabotage event. Finally, the commenter believed that the TSAR and transportation safety analysis report did not provide assurances that the cask and cladding will retain their integrity under the thermal conditions that exist at an Independent Spent Fuel Storage Installation (ISFSI) and that the NRC's SER had "glossed them over." These issues were considered to be crucially important to protecting the public health and safety, and therefore, should be addressed before the Holtec CoC can be issued.

Response: The NRC disagrees. As explained in the following responses to specific comments, NRC concludes that Holtec has provided reasonable assurance that the cladding and cask will retain their integrity under normal, off-normal, and accident conditions, as reflected in the NRC staff's SER in Chapter 7. Based on reviews of Holtec calculations and the site boundary dose rate, the NRC concluded that the health (radiological) impact under bounding accidents has been correctly addressed and that general licensees using the HI-STAR 100 casks will meet all applicable regulatory requirements (see SER Chapter 10). With respect to sabotage, under 10 CFR 72.212 requirements, each general licensee must protect the spent fuel against radiological sabotage. The thermal discussion in Chapter 4 of the NRC staff's SER reflects its evaluation of the adequacy of Holtec's thermal analysis and design ~~which~~ that has been found to be acceptable. The specific thermal conditions that exist at an ISFSI are required to be measured. Also Section 3.2.4.1 of the SER states that the NRC staff has determined that the applicant has developed design pressures and temperatures (internal) obtained by assuming bounding conditions. Specific thermal conditions that exist at an ISFSI are required to be maintained in accordance with appendix B to the CoC to ensure that actual conditions are bound by the analyses in the SAR and approved by the NRC.

Comment: 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.

Comment: One commenter state~~s~~d 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 Certificate of Compliance. 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 Safety Evaluation Report and Topical Safety Analysis Report. The NRC staff has reviewed the analyses performed by HOLTEC and found them acceptable. No changes to the Certificate of Compliance, Technical Specifications, Safety Evaluation Report, or Topical Safety Analysis Report are recommended. These models are based on sound engineering sciences and processes.

Comment: One commenter requested that a trouble shooting manual be prepared that includes information on how many of what type cask is 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.

Specific Comments:

Two of the commenters provided comments on specific topics. The comments and responses have been grouped into five areas: 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 Technical Specifications (TSs), and the applicant's Topical Safety Analysis Report (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.

Cladding Integrity

Comment: One commenter stated that the HI-STAR 100 system is designed to withstand a maximum deceleration of 60 g, while a Lawrence Livermore National Laboratories (LLNL) report shows that the most vulnerable fuel can withstand a deceleration of 63 g in the most adverse orientation (side drop). Specifically, Holtec and the NRC staff have not demonstrated a reasonable assurance that the cladding will maintain its integrity. The commenter believed that Holtec's analysis: (1) does not take into account the possible increase in rate of oxidation of cladding of high burnup fuel; (2) Holtec relies for its analysis on a LLNL report that fails to distinguish the effects of reactor irradiation on fuel assemblies; and (3) Holtec also relies on the LLNL report's incorrect assumption that fuel assemblies act as a static rigid rod. 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, is that the lift height of the HI-STAR 100 cask must be reduced to lower the g-forces on the cladding. Also, the commenter provided a table, "Effects of Changing in Dynamic Impact Effects on Spent Fuel."

Response: The NRC agrees, in part, with the comment. This issue is addressed in NRC IN 98-29 which 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 ~~as such it~~ will only affect cladding in high burn-up fuels 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 increased rate of oxidization of cladding should not be a concern of the HI-STAR 100 Storage Cask System and no change to the SER and CoC is necessary.

An analysis of the oxidation rate has shown that when an increased rate of oxidization of the fuel cladding is considered, the HI-STAR 100 design has provided reasonable assurance that the cladding will maintain its integrity.

The g-load for high burn-up fuel with thickness of cladding reduced by 17% is calculated in the spreadsheet[1] to be 50.81g (column C). Although the HI-STAR 100 Cask System is designed to withstand a maximum deceleration of 60g, the actual predicted maximum side drop deceleration is less than 50g's. As shown in HI-STAR 100 storage SAR (Holtec Report HI-941184), the maximum side drop g-loading is 49.7g's. In the HI-STAR transportation SAR (Holtec Report HI-951251), the maximum g-loading for the side drop is 46g's. Thus, even when thinner cladding thickness due to increased rate of oxidization is considered, the calculated g-load[1], that the fuel rod cladding can withstand (50.8), shown by the commenter is still larger than the predicted maximum g-load for a hypothetical cask side drop accident condition.

Based on the authorized contents, NRC determined that increased rate of oxidization of cladding should not be a concern for the HI-STAR 100 Cask Systems. Therefore, the NRC disagrees with the comment that the lift height must be reduced for the Hi-Star 100 cask.

Comment: One commenter stated that Holtec's SAR for the HI-STAR 100 storage cask relies upon a 1987 report by LLNL[2] for its estimate of g force that will damage fuel cladding. The LLNL report fails to take into account the increased brittleness of irradiated fuel assemblies.[3] Because the irradiated fuel assemblies may have been embrittled, they would also be less

---

[1]State of Utah, "Comments on Proposed Rule to add Holtec HI-STAR 100 Cask System to the List of Approved Spent Fuel Storage Casks." (March 26, 1999)

[2] LLNL Report.

[3] See e.g, UCID-21246, Table 4, which makes no distinction between Young's modulus and yield strength of a range of fuel assemblies.

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. "Cladding ductility decreases and yield stress increases with increasing neutron fluence."[4]

The design 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.[5] 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, considered the effects of irradiation on cladding. Table 3 of the report delineated 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 showed 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 showed that the yield strength is consistently lower than the tensile strength which suggested that significant margin exists between yielding of the cladding and gross rupture. The allowable g-load calculation in the report is based on the yield stress. Thus, the approach that was used in the LLNL report and reflected in the SAR is conservative and acceptable.

---

4 "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," Battelle Pacific Northwest Lab, NUREG/CR-5009 (February 1988)

5 Thin cladding acquires brittle characteristics at a faster rate than thicker cladding during fuel life. See IN 98-29 at 2 ("If this total oxidation limit were to be exceeded during an accident, the cladding could become embrittled.").

Comment: A commenter stated that Holtec's calculations rely upon LLNL'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. The LLNL Report specifically states, "It is important to emphasize that the g loadings shown in Figs. 6 and 7 are static loadings." [6] This assumption is incorrect. 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

[6] LLNL Report.

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.

Comment: One commenter stated that the calculated health impacts under hypothetical accidents conditions, discussed in Chapter 7 of Holtec's HI-STAR 100 SAR, are not conservative. Holtec's original hypothetical design basis accident condition assumed that 100% of the fuel rods are non-mechanically 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. To comply with NRC regulations, an applicant has the option of following the guidance provided in NUREG-1609, "The Standard Review Plan for Transportation Packages for Radioactive Material." NUREG-1609 references the ANSI N14.5, "Leakage Tests on Packages for Shipment for Radioactive Materials" standard for evaluating the maximum volumetric leak rate based on 10 CFR Part 71, Appendix A criteria. Holtec implemented the ANSI standard and calculated a limiting leak rate for the various fuel designs. Holtec then reduced that volumetric flow rate and calculated an equivalent break diameter. Applying ASME Boiler and Pressure Vessel Code criteria, the cask is designed and manufactured to the rigors of the testing requirements of Subpart F to 10 CFR Part 71 for normal and hypothetical accident conditions. This manufacturing technique, structural reliability reviews, and associated non-destructive examination assure that the material conditions do not lead to failures worse than analyzed through the ANSI N14.5 evaluation. Subpart F to 10 CFR Part 71 addresses the testing requirements of packages for normal conditions of transport and hypothetical accident conditions.

The NRC believes that the confinement/containment design is adequately rigorous such that reasonable assurance is provided that under normal and accident conditions, as defined by the regulations, the cask will not exceed the volumetric break flow rates identified by HOLTEC. Therefore, the design basis change has been found to be conservative and meets applicable regulations.

Comment: 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 ~~prior to~~ 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 is 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~~, that is~~ (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: One commenter stated that Holtec calculated the radiation dose to an adult 100 meters from the accident 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 did not address the listed pathways. The site specific licensees would be required under 10 CFR Parts 20 and 50 to address ~~such~~ these pathways if they exist, and therefore, is beyond the scope of this rulemaking.

Comment: One commenter stated that Holtec has not specifically calculated potential radiation dose to children and this does not meet NRC regulations. Further, NRC's methodology for calculating the potential dose to children is deficient.

Response: The NRC disagrees with the comment. The use of the effective dose equivalent concept reduces the importance of age-dependent intake-to-dose factors. Age dependency is of primary importance in calculating organ doses. Organs that are age-dependent, such as the thyroid gland, have low importance with respect to calculating effective dose equivalent. Also, a factor of 2 is included in the calculation of concentration limits for release to air and water, which, in part, accounts for age dependency.

The International Community and ~~the~~ Federal Agencies (including EPA) agree that the overall annual public dose limit, from all sources, should be 1 mSv (100 mrem) ~~which~~ that 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), ~~which is~~ 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 addresses 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: One commenter asked if the streaming dose rates have been measured and if ~~they have~~ not ~~been~~ 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 ~~and~~ because it was not required. The applicant did ~~however~~ 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: One commenter stated that the applicant should have performed a specific analysis for off-normal conditions for confinement analysis and should have included a n 85K 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: One commenter state~~s~~ d that the licensees' report on specific site doses to the public should be ~~put~~ 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: 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 is discussed in Sections 1.2.2.1, 8.1, and 10.1.4 of the SAR~~.~~ and ~~The inflatable annulus seal~~ 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 ~~swiped~~ 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: 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 this ~~specific~~ 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: 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 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 rule-making, 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 established 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 by to be protected against radiological sabotage using provisions and requirements as specified in 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 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 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.

Comments on the specific transportation aspects of the cask system and existing regulations dealing with sabotage and terrorism are beyond the scope of this rulemaking.

Comment: One commenter asked whether an evaluation for a truck bomb sabotage event has been conducted.

Response: Spent fuel in the ISFSI is required to be protected against radiological sabotage using provisions and requirements as specified in 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.

Thermal Requirements

Comment: One commenter stated that the CoC temperature limits for the storage cask are deficient ~~in that~~ because they did 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 a 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 [4.1.1] 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 (MPC) 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: One commenter asked why a coating without zinc was not required for the VSC-24. The commenter further questioned why NRC allowed coatings to be applied to casks ~~at all~~, 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 assess the use and adequacy of the coatings. Specific comments relating directly to VSC-24 are beyond the scope of this rulemaking.

Comment: 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 SRP. Cask fabrication activities are inspected by the licensees, and the NRC staff to ensure that components are fabricated as designed.

Comment: 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 NRC believes that the questions and comments on the Boral neutron absorber are addressed in Sections 6.42 and 9.1.4 of the Safety Evaluation Report, and Sections 1.2.1.3.1, 6.3.2, and 9.1.5.3 of the Topical Safety Analysis Report. ~~Additional information~~

~~follows:~~ 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.

There is not a credible mechanism for "creep or slump" of Boral in the cask based on industry experience.

Section 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 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 it~~. Further, the NRC accepts the use of NS-4-FR as a neutron absorber for storage cask applications ~~and~~ that has been used in other casks. Therefore, surveillance and monitoring is not needed.

Comment: 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 preformed 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: One commenter asked how the pre-possession 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 ("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: 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 must ~~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 the TSAR Section 8.1.4, MPC Fuel Loading, Step 28 and SER Section 8.1.3.

Comment: 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: 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.

Comment: 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.

Comment: 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~~making~~. The cask user is required to perform pre-operational testing to determine the effectiveness of the cooling methods.

Comment: One commenter questioned whether the manufacture'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 ~~further~~ 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 manufacturer paints with the specified emissivity. The NRC has reviewed the applicant's analysis and found that paints with an emissivity greater than 0.85 ~~is~~ are acceptable.

Comment: 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 technical specifications (TS) but is part of the vacuum drying procedure. The TS 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 ~~This~~ is beyond the scope of this rulemaking. If the drying process is unsuccessful and the TS requirements can not be met within 30 days, the fuel assemblies must be moved from the cask and be ~~re~~placed in the spent fuel pool.

Comments: 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 ~~which~~ 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 off-site dose formula ~~which is~~ used to estimate off-site doses from every cask in the array. The center-to-center cask pitch was assumed to be 12 ~~ft~~ 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. Off-site 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 their 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: One commenter questioned what the criteria was for the polyester resin "poured" into radial channels, how it was tested, handled and inspected, and whether it 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 ~~which~~ 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 is beyond the scope of this rulemaking. However, poured neutron shielding has been successfully used in other cask designs.

Comment: 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 Technical Specifications of the ~~Certificate~~ 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 ~~Certificate~~ CoC, and other requirements in Part 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 ~~Certificate~~ CoC; (2) the radiological source terms are adequately controlled in the ~~Certificate~~ CoC by limits on maximum burnup, minimum cooling time, maximum initial uranium loading, and maximum decay heat; (3) dose rates are controlled in the ~~Certificate~~ 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 will be revised to clarify that minimum enrichment is not an operating control for the HI-STAR 100.

Comment: One commenter asked what has been considered as credible ways to loosen the fixed neutron poisons.

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

Comment: 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: 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 ~~indeed~~ necessary to ensure that there will be no physical interferences and to allow free thermal expansions.

Comment: One commenter stated that all welds should be monitored unless it has 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: 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.

spell out

Comment: One commenter asked how long before a UT 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: 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: 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.

spell out

Comment: One commenter disagrees with allowing the use of PT 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 ~~what~~ the real size of the actual weld ~~is,~~ 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 Interim staff Guidance (ISG) - 4, "Cask Closure Weld Inspections." Actual cask welds are examined in accordance with site-specific procedures ~~which~~ that are beyond the scope of rulemaking for the HI-STAR 100 system. Nondestructive Examination (NDE) methods are specified in accordance with American Society of Mechanical Engineers Boiler & Pressure Vessel Code, Section III "Rules for Construction of Nuclear Power Plant Components" and Section V "Nondestructive Examination" and are already described in TSAR 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: One commenter believes that the marking material for the casks should be designated and that the mark needs to be permanent.

Response: NRC agrees with the comment. The storage marking name plate is made from a 4-inch by 10 inch, 14 gage 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 non-permanent marking will be used.

Comment: One commenter requested information on "rupture disc replacements", how they are tested for replacement, what the time criteria is, 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 five years to assure functionality.

Comment: 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: 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) ~~which~~ 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: 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: One commenter ask~~s~~ed where the MPC shell weld is located and ~~. The commenter further asked~~ 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: One commenter state~~s~~d 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 recess are visually inspected before ~~prior to~~ 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: One commenter requested information on the criteria for the critical flaw size.

Response: The criteria for critical flow size is included in Interim Staff Guidance (ISG) No. 4, "Cask Closure Weld Inspections," May 21, 1999, Revision 1. The NRC review determined that Holtec's proposed methodology is consistent with this ISG.

Comment: One commenter asked how subcontractors are to be audited and inspected.

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

Comment: One commenter believe~~s~~d that the first cask for each utility should be tested at a full heat load and ask~~s~~ed 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~~ must 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.

additional

The cask user will ~~further~~ 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: One commenter asks 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 ~~prior to~~ 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 ~~and~~ The resulting temperature increase is much less than any accident condition previously ~~already~~ analyzed might produce.

Comment: One commenter asks how lifting height should be verified and states that the height should be recorded.

Response: The maximum lifting height maintains the operating conditions of the 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 3.1.7.a 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.

P Comment: One commenter questioned how the MPC closure ring, lid, vent, and drain covers are removed during unloading and what precautions are taken.

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

P Comment: 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)?

P Response: The NRC staff has evaluated this condition~~, and~~. The TS requires 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.

P Comment: One commenter asked if "dry" unloading operations are considered.

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

Comment: 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. The NRC notes however 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 their radiological protection program to further address and mitigate crud dispersal.

Comment: The applicant made comments relevant to the helium backfill pressure of the cask. After discussions with the NRC staff, Holtec withdrew this comment ~~in~~ during a telephone conversation on 5/7/99.

Response: Not applicable.

Comments on Proposed Technical Specifications:

Upon review of the public comments received on the proposed technical specifications 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 technical specifications were in order. These changes result in a clearer set of technical specifications and moves the technical specifications for the new generation of dual-purpose cask systems toward a standardized format.

Comment: It was suggested that controlling the bases for the technical specifications as part of the Certificate of Compliance (CoC) would result in administrative burdens to all involved. Such 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: 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 technical specifications is unwarranted. Therefore, much of the information regarding fuel specifications and some of the design and codes information was moved from the technical specifications to a separate appendix to the CoC. The NRC staff did, however, 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 technical specifications.

Upon consideration of public comments and further consideration within the NRC, the NRC staff has determined that the structure of Technical Specification Section 3.1-Spent Fuel Storage Cask (SFSC) Integrity, did not provide appropriately clear guidance. Therefore, the NRC staff has revised this section of the technical specifications and associated bases to reflect a more logical and focused approach. The number of limiting conditions for operations (LCOs) in this section has been reduced to three. The NRC staff believes that this will enhance the usefulness of the technical specifications.

Comment: One commenter stated that if surface contamination exceeds 2200 dpm/100 cm2 from gamma and beta emitting sources, and smearable contamination limits can not be reduced to acceptable levels, the Technical Specifications 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 Technical Specifications could not be met.

Response: The NRC agrees in part. The revised version of the TS (2.2.2) requires verification that removable contamination is within limits during loading operations and provide up to 7 days to restore the contamination within limits. The specifications do not list MPC or spent fuel removal actions. Therefore, the comment is no longer applicable. 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: 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: One commenter stated that it would increase the standardization of the TS 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, ~~and their~~ Their definitions need to be contained within the TS proper. This practice is consistent with the standard TSs developed for the U.S. nuclear power reactors.

Comment: 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: 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 Technical Specifications (TS).

Response: The NRC agrees, in part. that This information should be moved. The NRC believes that this information should be placed in the SAR. 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: 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: 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: 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 & TS 3.1.6 OVERPACK Helium Leak Rate.

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

Comment: 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: 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: One commenter recommended relocating the information contained in TS Section 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 ~~and~~ 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: A commenter recommended relocating the information contained in TS Section 4.6 and 4.8 to an Administrative Controls chapter due to their content, and relocating Section 4.7 to the SAR ~~since~~ 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 TS, 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), ~~since~~ because it is established NRC staff practice to place important administrative requirements, even one-time requirements, in the TSs.

Comment: 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 recommends 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; ~~and~~ 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: One commenter recommended relocating the figure attached to TS 3.2.1 to the TS Bases ~~since~~ 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: The commenter stated that the TS do not comply with 10 CFR 72.44(d), ~~which~~ that requires ~~that there be~~ 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: One commenter recommended that within the Technical Specifications, 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: One commenter recommended that within the Technical Specifications, 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 that ~~which~~ has been relocated to Appendix B~~,~~ has been revised.

Comment: One commenter requested clarification of TS, ~~S~~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 state~~s~~d 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 recommend~~s~~ed the addition of "decay heat" after "lesser" and before "loads" in the last line.

Response: The NRC agrees with these comments. TS ~~S~~section 3.3 has been revised and clarified accordingly. Reporting requirements refer to the first system loaded. Subsequent loadings only need to be reported when heat load increases.

Comment: One commenter recommended some editorial changes to revise Technical Specification 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: 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: One commenter recommend~~s~~ed 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: One commenter recommend~~s~~ed a revision to Technical Specifications, 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: One commenter recommend~~s~~ed the revision of Technical Specification Table 3-1, Item 1.c to change the lower helium tolerance to 10% 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: One commenter recommend~~s~~ed 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 disagrees with the comment. Appendix B lists all approved exceptions to the ASME code.

Comment: One commenter recommend~~s~~ed 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 recommend~~s~~ed 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: One commenter recommends the addition of a third option to TS LCO 3.1.7 and Bases B3.1.7 (or elsewhere in the TSs) ~~which~~ 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: One commenter believes that the 48-hour time limit within Technical Specifications 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 technical specification. Some of the time limits have been extended to provide for a controlled, deliberate response to the LCO condition.

Comment: 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 ~~of~~ 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 (SAT). 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 TS and it has been maintained.

Comment: 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 Certificate of Compliance

Comment: 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.

Comment: Two commenters recommended that a Bases Control Program be added to the TS or CoC.

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

Comment: One commenter requested information on the that 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: 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 Safety Evaluation Report

Comment: 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 non-destructive 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: 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: One commenter recommended ~~with~~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: One commenter recommended that in ~~the~~ 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: NRC agrees with the comment. The SER has been revised.

Comment: One commenter stated that ~~the~~ 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, in accordance with a user's as low as reasonably achievable (ALARA) practices, of the MPC lid closure weld.

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 ~~which~~ 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: 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: 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 recommends 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 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: 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.

Comments on the Applicant's Topical Safety Analysis Report

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

Comment: 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: 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 states 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: 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. Drawing 1399 and BM-1476 have been revised to reflect the change.

Comment: 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, ~~but~~ four drawings have not been ~~further~~ revised to match the transportation SAR. This is acceptable to the NRC staff because it reflects storage design features.

Comment: In the SAR, one commenter 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: 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: 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: The applicant suggested a clarification for the sequence for the ~~H~~ydrostatic 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: 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: 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.

Comment: One commenter (the applicant) recommend~~s~~ed 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 Amendments

The NRC staff modified both the SER and the CoC for the Holtec International HI-STAR 100 cask system in response to public comments. Holtec also modified its SAR. The listing of the Holtec International HI-STAR 100 cask system within § 72.214, "List of approved spent fuel storage casks" was not changed as a result of the public comments. However, the listing has been revised to remove the revision number from the SAR title. The NRC is no longer including the SAR revision numbers in the listing of Certificates in Part 72.214.

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 the 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

The NRC has determined under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in Subpart A of 10 CFR Part 51, 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 Materials 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 ~~proposed~~ 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 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 ~~would~~ 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 they notify the NRC in advance, the spent fuel is stored under the conditions specified in the cask's certificate of compliance, and the conditions of the general license are met. In that rule~~making,~~ four spent fuel storage casks were approved for use at reactor sites, and were listed in 10 CFR 72.214. That rule~~making~~ envisioned that storage casks certified in the future could be routinely added to the listing in section 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 ~~give~~ 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 ~~which~~ that can be used under a general license and would be in conflict with NWPA direction to the Commission to approve technologies for the use of spent fuel storage at the sites of civilian nuclear power reactors without, to the maximum extent practicable, the need for additional site reviews. This alternative also would tend to exclude new vendors from the business market without cause and would arbitrarily limit the choice of cask designs available to power reactor licensees.

This final rule~~making~~ ~~would~~ 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 ~~which~~ 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 ~~which~~ 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 ~~in that~~ 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~~making~~ 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 cask vendors. 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 ~~which~~ would impose backfits as defined in the backfit rule. Therefore, a backfit analysis is not required.

List of Subjects in 10 CFR Part 72

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

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

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

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

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

Section 72.44(g) also issued under secs. 142(b) and 148(c), (d), Pub. L. 100-203, 101 Stat. 1330-232, 1330-236 (42 U.S.C. 10162(b), 10168(c),(d)). Section 72.46 also issued under sec. 189, 68 Stat. 955 (42 U.S.C. 2239); sec. 134, Pub. L. 97-425, 96 Stat. 2230 (42 U.S.C. 10154). Section 72.96(d) also issued under sec. 145(g), Pub. L. 100-203, 101 Stat. 1330-235 (42 U.S.C. 10165(g)). Subpart J also issued under secs. 2(2), 2(15), 2(19), 117(a), 141(h), Pub. L. 97-425, 96 Stat. 2202, 2203, 2204, 2222, 2244, (42 U.S.C. 10101, 10137(a), 10161(h)). Subparts K and L are also issued under sec. 133, 98 Stat. 2230 (42 U.S.C. 10153) and sec. 218(a), 96 Stat. 2252 (42 U.S.C. 10198).

In section 72.214, Certificate of Compliance (CoC) 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 Numbers: HI-STAR 100

Dated at Rockville, Maryland, this______________ day of_______ ,1999.

For the Nuclear Regulatory Commission.

____

William D. Travers,
Executive Director for Operations.

ENVIRONMENTAL ASSESSMENT AND FINDING OF
NO SIGNIFICANT IMPACT
ON
AMENDMENT TO 10 CFR PART 72
HI-STAR 100
LIST OF APPROVED SPENT FUEL STORAGE CASKS: ADDITION

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

I. THE ~~PROPOSED~~ ACTION

The ~~proposed~~ final rule action ~~is to~~ 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 ~~would~~ 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

---

[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 non-radiation 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: 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 web site established by NRC for this rulemaking.

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.

###

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)			

____

____

____

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)

____

____

____

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)			

____

____

____

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 ~~would~~ 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 ~~would~~ 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 ~~which are~~ 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 ~~would~~ 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 ~~would~~ 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 ~~which are~~ 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

WEEKLY REPORT TO THE COMMISSION

OFFICE OF NUCLEAR MATERIALS SAFETY AND SAFEGUARDS

Final Rule Signed by EDO

On _______________, the Executive Director for Operations approved a final rule ~~which~~ 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 ~~would~~ 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 _____________ to the Office of the Federal Register for publication, unless otherwise directed by the Commission.

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 "Hi-Star 100 - List of Approved Spent Fuel Storage Casks: 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.

Date

William D. Travers,
Executive Director for Operations