Small Business Regulatory Enforcement Fairness Act

Under 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

Under the Regulatory Flexibility Act of 1980 (5 U.S.C. 605(b)), the NRC certifies that this rule will not, if promulgated, have a significant economic impact on a substantial number of small entities. This final rule affects only the licensing and operation of nuclear power plants, independent spent fuel storage facilities, and Transnuclear. 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 (§ 50.109 or § 72.62) does not apply to this direct final rule because this amendment does not involve any provisions that would impose backfits as defined. Therefore, a backfit analysis is not required.

List of Subjects in 10 CFR Part 72

Administrative practice and procedure, Hazardous waste, Nuclear materials, Occupational safety and health, Penalties, Radiation protection, Reporting and recordkeeping requirements, Security measures, Spent fuel, Whistleblowing

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 adopting the following amendments to 10 CFR part 72.

≫65 FR 25241
Published 5/1/00
Effective 5/31/00

10 CFR Part 72

RIN 3150–AG 31

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

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule

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

EFFECTIVE DATE: This final rule is effective on May 31, 2000.

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

SUPPLEMENTARY INFORMATION:

Background

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

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

Discussion

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

Based on NRC review and analysis of public comments, the NRC staff has modified, as appropriate, its proposed CoC, including its appendices, the Technical Specifications (TSs), and the

Template = SECY-028 SECY-02

NUCLEAR REGULATORY COMMISSION

Docket No 72-22 Official Exh. No. Staff 54

In the matter of Private Fuel Storage

Staff ✓ IDENTIFIED ✓

Applicant RECEIVED

Intervenor REJECTED

Cont'g Off'r

Contractor DATE 6-24-02

Other Witness

Reporter R. Davis

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USNRC

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OFFICE OF THE SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

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

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

B. Radiation Protection

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

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

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

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

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

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

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

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

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

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

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

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

Comment B.7: One commenter asked what the dose for the 2x5 cask array was at 100 meters.

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

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

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

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

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

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

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

Response: No response is necessary.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Response: The NRC disagrees with the comment. Although the NRC does not expect effluent release from the HI-STORM 100 under credible conditions, the applicant's method used to determine design basis dose rates from a hypothetical release are adequate to demonstrate that the confinement features are sufficient to meet the radiological requirements of 10 CFR 72.106. The NRC staff believes the methods applied by the applicant conservatively bound hypothetical dose rates to the general public. Further, 10 CFR 72 212(b)(6) requires the general licensee to review its reactor emergency plan and radiation protection program to determine its effectiveness and make changes if necessary when using a cask listed in 10 CFR Part 72, Subpart L.

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

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

C. Accident Analysis

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

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

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

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

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

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

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

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

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

Response: The design basis earthquake is the most severe earthquake that has been historically reported for a particular site and surrounding area, with sufficient margins for the limited accuracy, quantity, and period of time in which historical data have been accumulated. Structure, systems, and components important to safety are designed to withstand the effects of this earthquake without loss of capability to perform their safety functions. The design basis earthquake is described by an appropriate response spectrum anchored at the peak ground acceleration. The response is then amplified through the pad to obtain the input response spectrum at the top of the pad (or at the bottom of the cask) for cask seismic evaluation. Soil and storage pad interaction is a site-specific issue that will be addresses in the cask user's 10 CFR 72.212 evaluation and is beyond the scope of this rule.

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

Response: The NRC does not consider the scenario described by the