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

September 9, 1999

OFFICE OF THE
GENERAL COUNSEL

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In the Matter of
Private Fuel Storage L.L.C.
(Independent Spent Fuel Storage Installation)
Docket No. 72-22-ISFSI

Dear Administrative Judges:

Attached for your information are two Federal Register notices published on September 3, 1999, which may be relevant to issues raised in this proceeding: (1) Final Rule, "List of Approved Spent Fuel Storage Casks: (HI-STAR 100) Addition," 64 Fed. Reg. 48259 (Sept. 3, 1999); and (2) Final Rule, "Changes to Requirements for Environmental Review for Renewal of Nuclear Power Plant Operating Licenses," 64 Fed. Reg. 48496 (Sept. 3, 1999).

The first Notice adds the HI-STAR 100 cask system to the list of approved spent fuel storage casks under 10 C.F.R. Part 72, and responds to comments received in that rulemaking proceeding. The second Notice modifies the Commission's generic determination concerning the cumulative environmental impacts of the transportation of nuclear waste and spent fuel to and from a nuclear reactor, to include spent fuel having a higher burnup and higher enrichment than is indicated in 10 C.F.R. Part 51, Table S-4.

Inasmuch as the Notices are publicly available by electronic means and paper copies are enclosed herewith, an additional electronic copy is not being provided at this time.

Sincerely,

Sherwin E Turk

Sherwin E. Turk
Counsel for NRC Staff

cc w/Encl.: Service List

these infrastructures and policies are adequate for disease control.

The commenters also said that information supplied by foreign regions should be made available to the public for review.

Currently, when a region requests permission to export animals and animal products to the United States, the supporting documentation supplied by the region is published by APHIS on the Internet at http://www.aphis.usda.gov/vs/reg-request.html. This Internet address can be accessed by the public. To request additional information, the individual listed under **FOR FURTHER INFORMATION CONTACT** may be contacted.

Therefore, for the reasons given in the proposed rule and in this document, we are adopting the proposed rule as a final rule, without change.

Effective Date

This is a substantive rule that relieves restrictions and, pursuant to the provisions of 5 U.S.C. 553, may be made effective less than 30 days after publication in the **Federal Register**. This rule relieves restrictions that require horses imported from Morocco to enter the United States only at the port of New York and be quarantined at the New York Animal Import Center in Newburgh, NY, for at least 60 days. This rule allows horses from Morocco to be shipped to and quarantined at ports designated in § 93.303, and reduces the quarantine period to an average of 3 days to meet the quarantine and testing requirements specified in § 93.308. Therefore, the Administrator of the Animal and Plant Health Inspection Service has determined that this rule should be effective 15 days after the date of publication in the **Federal Register**.

Executive Order 12866 and Regulatory Flexibility Act

This rule has been reviewed under Executive Order 12866. This rule has been determined to be not significant for purposes of Executive Order 12866 and, therefore, has not been reviewed by the Office of Management and Budget.

This rule will recognize Morocco as free of AHS. This action will allow horses from Morocco to be shipped to and quarantined at ports designated in § 93.303 and will reduce the quarantine and testing period to an average of 3 days to meet quarantine requirements specified in § 93.308.

U.S. importers of competition and breeding horses from Morocco will be affected by this rule. These importers will no longer be required to quarantine horses from Morocco for 60 days at the New York Animal Import Center in Newburgh, NY, at a cost of approximately $5,296 per horse.

In 1998, the United States imported 41,876 horses, valued at $206 million; none of these horses were imported into the United States from Morocco. Removing the requirement for a 60-day quarantine for horses from Morocco will make the importation of horses less expensive and logistically easier. As a result, we anticipate that U.S. importers of competition and breeding horses might begin importing horses from Morocco. Since the value of Morocco's exports of purebred horses in 1997 was approximately $44,000, we do not expect that the number of horses exported to the United States will be significant. Furthermore, most horses imported from Morocco will probably be in the United States on a temporary basis for particular events, such as for races or breeding, and then transported back to Morocco. For these reasons, we anticipate the overall economic effect on U.S. entities will be minimal.

Under these circumstances, the Administrator of the Animal and Plant Health Inspection Service has determined that this action will not have a significant economic impact on a substantial number of small entities.

Executive Order 12988

This final rule has been reviewed under Executive Order 12988, Civil Justice Reform. This rule: (1) Preempts all State and local laws and regulations that are inconsistent with this rule; (2) has no retroactive effect; and (3) does not require administrative proceedings before parties may file suit in court challenging this rule.

Paperwork Reduction Act

This final rule contains no information collection or recordkeeping requirements under the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq.*).

List of Subjects in 9 CFR Part 93

Animal diseases, Imports, Livestock, Poultry and poultry products, Quarantine, Reporting and recordkeeping requirements.

Accordingly, we are amending 9 CFR part 93 as follows:

PART 93—IMPORTATION OF CERTAIN ANIMALS, BIRDS, AND POULTRY, AND CERTAIN ANIMAL, BIRD, AND POULTRY PRODUCTS; REQUIREMENTS FOR MEANS OF CONVEYANCE AND SHIPPING CONTAINERS

1. The authority citation for part 93 continues to read as follows:

Authority: 7 U.S.C. 1622; 19 U.S.C. 1306; 21 U.S.C. 102–105, 111, 114a, 134a, 134b, 134c, 134d, 134f, 136, and 136a; 31 U.S.C. 9701; 7 CFR 2.22, 2.80, and 371.2(d).

2. In § 93.308, paragraph (a)(2) is revised to read as follows:

§ 93.308 Quarantine requirements.

(a) * * *

(2) Horses intended for importation from regions APHIS considers to be affected with African horse sickness may enter the United States only at the port of New York, and must be quarantined at the New York Animal Import Center in Newburgh, New York, for at least 60 days. This restriction also applies to horses that have stopped in or transited a region considered affected with African horse sickness. APHIS considers the following regions to be affected with African horse sickness: All the regions on the continent of Africa, except Morocco; Oman; Qatar; Saudi Arabia; and the Yemen Arab Republic.

* * * * *

Done in Washington, DC, this 30th day of August 1999.

Bobby R. Acord,

Acting Administrator, Animal and Plant Health Inspection Service.

[FR Doc. 99–23010 Filed 9–2–99; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

10 CFR Part 72

RIN 3150–AG17

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

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

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

EFFECTIVE DATE: This final rule is effective on October 4, 1999.

FOR FURTHER INFORMATION CONTACT: Stan Turel, telephone (301) 415–6234, e-mail spt@nrc.gov of the Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001.

SUPPLEMENTARY INFORMATION:

Background

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

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

Discussion

This rule will add the Holtec International HI–STAR 100 to the list of NRC approved casks for spent fuel storage in 10 CFR 72.214. Following the procedures specified in 10 CFR 72.230 of Subpart L, Holtec International submitted an application for NRC approval together with the Safety Analysis Report (SAR) entitled "HI–STAR 100 Cask System Topical Safety Analysis Report (SAR), Revision 8." The NRC evaluated the Holtec International submittal and issued a preliminary Safety Evaluation Report (SER) and a proposed Certificate of Compliance (CoC) for the Holtec International HI–STAR 100 cask system. The NRC published a proposed rule in the **Federal Register** (64 FR 1542; January 11, 1999) to add the HI–STAR 100 cask system to the listing in 10 CFR 72.214. The comment period ended on March 29, 1999. Nine comment letters were received on the proposed rule.

Based on NRC review and analysis of public comments, the staff has modified, as appropriate, its proposed CoC, including its appendices, the Technical Specifications (TSs), and the Approved Contents and Design Features, for the Holtec International HI–STAR 100 cask system. The staff has also modified its preliminary SER and has revised the title of the SAR in the listing of this cask design in 10 CFR 72.214.

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

The proposed CoC has been revised to clarify the requirements for making changes to the CoC by specifying that the CoC holder must submit an application for an amendment to the certificate if a change to the CoC, including its appendices, is desired. This revision conforms the change process to that specified in 10 CFR 72.48, as recently approved by the Commission. The CoC has also been revised to delete the proposed exemption from the requirements of 10 CFR 72.124(b) because a recent amendment of this regulation makes the exemption unnecessary (64 FR 33178; June 22, 1999). In addition, other minor, nontechnical, changes have been made to CoC 1008 to ensure consistency with NRC's new standard format and content for CoCs. Finally, extensive comments were received from Holtec International and other industry organizations suggesting changes to the TSs and the Approved Contents and Design Features. Some of these were editorial in nature, others provided clarification and consistency, and some reflected final refinements in the cask design. Staff agrees with many of these suggested changes and has incorporated them into the final documents, as appropriate.

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 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 October 4, 1999. Single copies of the CoC and SER are available for public inspection and/or copying for a fee at the NRC Public Document Room, 2120 L Street, NW. (Lower Level), Washington, DC.

Summary of Public Comments on the Proposed Rule

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

One commenter supported use of the direct final rule process for future revisions to the listing in 10 CFR 72.214, stating that it was imperative that the regulatory process be streamlined when there is no adverse safety concern. Two commenters were opposed to use of a direct final rule process stating that a direct final rule would diminish the public role in commenting on the approval of spent nuclear fuel casks and thereby the public's ability to affect the outcome of rulemaking procedures. One of these commenters believed that, given past problems with the casks, future approval should be subject to adequate and rigorous public scrutiny.

Those opposed also believed that 30 days (as would be allowed in a direct final rule process) is not sufficient time to prepare comments that may be significantly adverse so as to cause the NRC to withdraw the published final rule. The two commenters did not believe that an addition to or revision of the listing is likely to be either noncontroversial or routine as evidenced by the number of comments they had on the Holtec HI-STAR 100 proposed rule.

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

Comments on the Holtec International HI-STAR 100 Cask System

The comments and responses have been grouped into five areas: general comments, cladding integrity, health impacts, sabotage events, thermal requirements, and miscellaneous items. Several of the commenters provided specific comments on the draft CoC, the NRC staff's preliminary SER, the TSs, and the applicant's Topical SAR. Some of the editorial comments have been grouped as well as some of the comments on the drawings in the SAR. To the extent possible, all of the comments on a particular subject are grouped together. The listing of the Holtec International HI-STAR 100 cask system within 10 CFR 72.214, "List of approved spent fuel storage casks," has not been changed as a result of the public comments. A review of the comments and the NRC staff's responses follow:

General Comments

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

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

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

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

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

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

Cladding Integrity

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

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

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

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

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

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

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

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

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

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

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

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

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

Radiation Protection

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Sabotage Events

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

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

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

The physical protection systems at an ISFSI and its associated reactor are similar in design features to ensure the detection and assessment of unauthorized activities. Alarm annunciations at the general license ISFSI are monitored by the alarm stations at the reactor site. Response to intrusion alarms is required. Each ISFSI is periodically inspected by NRC, and the licensee conducts periodic patrols and surveillances to ensure that the physical protection systems are operating within their design limits. It is the ISFSI licensee who is responsible for protecting spent fuel in the casks from sabotage rather than the certificate holder. Comments on the specific transportation aspects of the cask system and existing regulations specifying what type of sabotage events must be considered are beyond the scope of this rulemaking.

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

Response: The staff has evaluated the effects of a truck bomb located adjacent to storage casks. Spent fuel in the ISFSI is required to be protected against radiological sabotage using provisions and requirements as specified in 10 CFR 72.212(b)(5). Each utility licensed to have an ISFSI at its reactor site is required to develop physical protection plans and install a physical protection system that provides high assurance against unauthorized activities that could constitute an unreasonable risk to the public health and safety. The physical protection systems at an ISFSI and its associated reactor are similar in design to ensure the detection and assessment of unauthorized activities. Response to intrusion alarms is required. Each ISFSI is periodically inspected by NRC, and the licensee conducts periodic patrols and surveillances to ensure that security systems are operating within their design limits. The NRC believes that the inherent nature of the spent fuel and the spent fuel storage cask provides adequate protection against a vehicle bomb, and has concluded that there are no safety concerns outside the controlled area.

Thermal Requirements

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

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

Miscellaneous Items

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

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

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

Response: NRC regulations do not require an ASME stamp for a cask. The design and fabrication requirements for a certified dry cask storage system are described in 10 CFR Part 72 and the NRC staff's Standard Review Plan, NUREG 1536, "Standard Review Plan for Dry Cask Storage Systems." Applicant submittals are reviewed to the criteria in the Standard Review Plan. Cask fabrication activities are inspected by the licensees and the NRC staff to ensure that components are fabricated as designed.

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

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

Sections 1.2.1.3.1 and 9.1.5.3 of the SAR describe the testing procedures for Boral. Boral will be manufactured and tested under the control and surveillance of a quality assurance and quality control program that conforms to the requirements of 10 CFR Part 72, Subpart G. A statistical sample of each manufactured lot of Boral is tested by the manufacturer using wet chemistry procedures and/or neutron attenuation techniques.

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

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

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

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

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

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

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

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

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

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

Response: The tipover, end drops, and horizontal drop analyses form part of the structural design basis for the HI–STAR 100 cask design. Holtec described drops and tipover analyses in SAR Section 3.4.9. The NRC's evaluation of the vendor's analyses is described in SER Sections 3.2.3.1 and 3.2.3.2. The NRC found the results of these analyses to be satisfactory in that the calculated stresses were within the allowable criteria of the American Society of Mechanical Engineers (ASME) Code. Before using the HI–STAR 100 casks, the general licensee must evaluate the foundation materials to ensure that the site characteristics are encompassed by the design bases of the approved cask. The events listed in the comment are among the site-specific considerations that must be evaluated by the licensee using the cask.

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

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

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

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

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

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

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

Response: The drain down time is not specified in the TSs but is part of the vacuum drying procedure. The TSs state that the vacuum drying must be completed within 7 days. There is not a specific procedure in the application to monitor the water content; however, that will be addressed by the cask user on a site-specific basis and is beyond the scope of this rulemaking. If the drying process is unsuccessful and the TS requirements cannot be met within 30 days, the fuel assemblies must be moved from the cask and be placed in the spent fuel pool.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Response: The NRC disagrees with this concern over the integrity of the bolting material. The 54, 1⅝-inch-diameter, closure plate bolts are made from ASME SB–637–N07718 material per SAR BM–1476. N07718, a nickel-chromium alloy, does not become brittle at colder temperatures. N07718 is a high strength, corrosion resistant material used in applications with a temperature range from −423 °F (−253 °C) to 1300 °F (704 °C) (Source: Inconel Alloy 718, Inco Alloys International, fourth edition, 1985). This material will not rust, unlike carbon steels in corrosive environments. In addition, the material retains significant ductility down to −320 °F (−196 °C) as shown by impact test results (Source: Inconel Alloy 718, Table 27). Therefore, the NRC has no concerns about the bolting material.

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

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

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

Response: NRC disagrees with this comment. The NRC position on inspection of closure welds is contained in ISG–4, "Cask Closure Weld Inspections." Actual cask welds are examined in accordance with site-specific procedures that are beyond the scope of rulemaking for the HI–STAR 100 system. Nondestructive Examination (NDE) methods are specified in accordance with Section III "Rules for Construction of Nuclear Power Plant Components," and Section V "Nondestructive Examination," of the ASME Code and are already described in SAR Tables 9.1.1, 9.1.2, and 9.1.3. A permanent record of completed welds will be made using video, photographic, or other means that can provide a retrievable record of weld integrity. As per accepted industry practice, the record is typically in color format, in order to capture the red dye typically used for PT examinations.

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

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

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

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

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

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

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

Response: Before using the HI–STAR 100 casks, the general licensee must evaluate the site to determine whether or not the chosen site parameters are enveloped by the design bases of the approved cask as required by 10 CFR 72.212(b)(3). The licensee's site evaluation should consider the effects of nearby transportation and military activities. Generally, a cask's inherent design will withstand tornado missiles and collision forces imposed by light general aviation aircraft (i.e., 1500–2000 pounds) that constitute the majority of aircraft in operation today. The events listed in the comment are among the site-specific considerations that must be evaluated and are beyond the scope of this rulemaking.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Response: Not applicable.

Comments on Proposed TSs

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Comment No. 72: One commenter recommended relocating the information contained in TS Sections 4.3 and 4.5 to the SAR, and recommended eliminating TS Section 4.4, stating that this section is a duplication of existing regulatory requirements.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Comments on the Draft CoC

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

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

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

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

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

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

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

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

Comments on the NRC Staff's SER

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Comments on the Applicant's Topical SAR

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Summary of Final Revisions

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

Agreement State Compatibility

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

Finding of No Significant Environmental Impact: Availability

Under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in Subpart A of 10 CFR part 51, the NRC has determined that this rule is not a major Federal action significantly affecting the quality of the human environment and therefore an environmental impact statement is not required. This final rule adds an additional cask to the list of approved spent fuel storage casks that power reactor licensees can use to store spent fuel at reactor sites without additional site-specific approvals from the Commission. The environmental assessment and finding of no significant impact on which this determination is based are available for inspection at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC. Single copies of the environmental assessment and finding of no significant impact are available from Stan Turel, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 415–6234, e-mail spt@nrc.gov.

Paperwork Reduction Act Statement

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

Public Protection Notification

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

Voluntary Consensus Standards

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

Regulatory Analysis

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

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

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

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

Small Business Regulatory Enforcement Fairness Act

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

Regulatory Flexibility Certification

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

Backfit Analysis

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

List of Subjects in 10 CFR Part 72

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

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

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

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

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

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

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

§ 72.214 List of approved spent fuel storage casks.

* * * * *

Certificate Number: 1008
SAR Submitted by: Holtec International
SAR Title: HI-STAR 100 Cask System Topical Safety Analysis Report
Docket Number: 72–1008
Certification Expiration Date: (20 years after final rule effective date)
Model Number: HI-STAR 100

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

For the Nuclear Regulatory Commission.

William D. Travers,

Executive Director for Operations.

[FR Doc. 99–23075 Filed 9–2–99; 8:45 am]

BILLING CODE 7590–01–P

FEDERAL RESERVE SYSTEM

12 CFR Part 201

[Regulation A]

Extensions of Credit by Federal Reserve Banks; Change in Discount Rate

AGENCY: Board of Governors of the Federal Reserve System.

ACTION: Final rule.

SUMMARY: The Board of Governors has amended its Regulation A on Extensions of Credit by Federal Reserve Banks to reflect its approval of an increase in the basic discount rate at each Federal Reserve Bank. The Board acted on requests submitted by the Boards of Directors of the twelve Federal Reserve Banks.

EFFECTIVE DATE: The amendments to part 201 (Regulation A) were effective August 24, 1999. The rate changes for adjustment credit were effective on the dates specified in 12 CFR 201.51.

FOR FURTHER INFORMATION CONTACT: Jennifer J. Johnson, Secretary of the Board, (202) 452–3259; for users of Telecommunications Device for the Deaf (TDD), contact Diane Jenkins, (202) 452–3544, Board of Governors of the Federal Reserve System, 20th and C Streets NW., Washington, D.C. 20551.

SUPPLEMENTARY INFORMATION: Pursuant to the authority of sections 10(b), 13, 14, 19, et al., of the Federal Reserve Act, the Board has amended its Regulation A (12 CFR part 201) to incorporate changes in discount rates on Federal Reserve Bank extensions of credit. The discount rates are the interest rates charged to depository institutions when they borrow from their district Reserve Banks.

The "basic discount rate" is a fixed rate charged by Reserve Banks for adjustment credit and, at the Reserve Banks' discretion, for extended credit. In increasing the basic discount rate from 4.5 percent to 4.75 percent, the Board acted on requests submitted by the Boards of Directors of the twelve Federal Reserve Banks. The new rates were effective on the dates specified below.

With financial markets functioning more normally, and with persistent strength in domestic demand, foreign economies firming, and labor markets remaining very tight, the degree of monetary ease required to address the global financial market turmoil of last fall is no longer consistent with sustained, non-inflationary, economic expansion. The 25-basis-point increase in the discount rate was associated with a similar increase in the federal funds rate announced at the same time.

Regulatory Flexibility Act Certification

Pursuant to section 605(b) of the Regulatory Flexibility Act (5 U.S.C. 605(b)), the Board certifies that the change in the basic discount rate will not have a significant adverse economic impact on a substantial number of small entities. The rule does not impose any additional requirements on entities affected by the regulation.

Administrative Procedure Act

The provisions of 5 U.S.C. 553(b) relating to notice and public participation were not followed in connection with the adoption of the amendment because the Board for good cause finds that delaying the change in the basic discount rate in order to allow notice and public comment on the change is impracticable, unnecessary, and contrary to the public interest in fostering sustainable economic growth.

The provisions of 5 U.S.C. 553(d) that prescribe 30 days prior notice of the effective date of a rule have not been followed because section 553(d) provides that such prior notice is not necessary whenever there is good cause for finding that such notice is contrary to the public interest. As previously stated, the Board determined that delaying the changes in the basic discount rate is contrary to the public interest.

List of Subjects in 12 CFR Part 201

Banks, banking, Credit, Federal Reserve System.

For the reasons set out in the preamble, 12 CFR part 201 is amended as set forth below:

PART 201—EXTENSIONS OF CREDIT BY FEDERAL RESERVE BANKS (REGULATION A)

1. The authority citation for 12 CFR part 201 continues to read as follows:

Authority: 12 U.S.C. 343 *et seq.*, 347a, 347b, 347c, 347d, 348 *et seq.*, 357, 374, 374a and 461.

2. Section 201.51 is revised to read as follows:

§ 201.51 Adjustment credit for depository institutions.

The rates for adjustment credit provided to depository institutions under § 201.3(a) are:

Federal Reserve Bank	Rate	Effective
Boston	4.75	August 24, 1999.
New York	4.75	August 24, 1999.
Philadelphia	4.75	August 24, 1999.
Cleveland	4.75	August 24, 1999.

federal register

Friday
September 3, 1999

Part III

Nuclear Regulatory Commission

10 CFR Part 51

Changes to Requirements for Environmental Review for Renewal of Nuclear Power Plant Operating Licenses; Final Rules

NUCLEAR REGULATORY COMMISSION

10 CFR Part 51

RIN 3150–AG05

Changes to Requirements for Environmental Review for Renewal of Nuclear Power Plant Operating Licenses

AGENCY: Nuclear Regulatory Commission.

ACTION: Final Rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is amending its regulations on the environmental information required in applications to renew the operating licenses of nuclear power plants. This amendment expands the generic findings about the environmental impacts due to transportation of fuel and waste to and from a single nuclear power plant. Specifically, this amendment adds to findings concerning the cumulative environmental impacts of convergence of spent fuel shipments on a single destination, rather than multiple destinations, and the environmental impact of transportation of higher enriched and higher burnup spent fuel during the renewal term. The effect of this amendment is to permit the NRC to make a generic finding regarding the impacts so that an analysis of these impacts will not have to be repeated for each individual license renewal application. This action reduces the regulatory burden on applicants for license renewal by replacing individual plant operating license renewal reviews with a generic review of these topics. Also, this amendment incorporates rule language to be consistent with the findings in NUREG–1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants" (May 1996), which addresses local traffic impacts attributable to continued operation of the nuclear power plant during the license renewal term.

In analyzing the environmental impact of transporting spent fuel and waste in the vicinity of a single repository, the NRC evaluated the impact in the vicinity of Yucca Mountain and specifically the impacts in the vicinity of Las Vegas, NV. The NRC elected to evaluate the impacts in the vicinity of Yucca Mountain because Yucca Mountain is the only location currently being evaluated for a repository under the Nuclear Waste Policy Act. The NRC's analysis of the impacts in the vicinity of Yucca Mountain in this instance does not prejudge the eventual licensing of Yucca Mountain as a repository. Rather, it reflects NRC's existing license renewal process by reflecting current repository activities and policies. If an application is filed by the Department of Energy (DOE), the licensing process for a repository in the vicinity of Yucca Mountain will constitute an entirely separate regulatory action from the proposed final rule. Furthermore, if, based on technical or national policy considerations, some site other than Yucca Mountain is selected in the future for study as a repository, the NRC will evaluate the applicability of the generic environmental impact statement for the license renewal process to other proposed repository sites.

EFFECTIVE DATE: October 4, 1999.

FOR FURTHER INFORMATION CONTACT: Donald P. Cleary, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, telephone: 301–415–3903; e-mail: DPC@nrc.gov.

SUPPLEMENTARY INFORMATION:

Background

On June 5, 1996 (61 FR 28467), the Commission published in the **Federal Register** a final rule amending its environmental protection regulations in 10 CFR part 51 to improve the efficiency of the process of environmental review for applicants seeking to renew a nuclear power plant operating license for up to an additional 20 years. The rulemaking was based on the analyses reported in the final report of NUREG–1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants" (GEIS) (May 1996). The rulemaking drew on the considerable experience of operating nuclear power plants in order to generically assess many of the environmental impacts, so that repetitive reviews of issues whose impacts are well understood could be minimized. In the statement of considerations accompanying the final rule, the Commission stated that before the final rule became effective, the Commission was seeking comments on the treatment of low-level waste (LLW) storage and disposal impacts, the cumulative radiological effects from the uranium fuel cycle, and the effects from the disposal of high-level waste (HLW) and spent fuel. In response to the June 5, 1996, final rule, a number of commentors stated that the requirements for the review of transportation of HLW in the rule were unclear with respect to (1) the use and legal status of 10 CFR 51.52, "Table S–4— Environmental Impact of Transportation of Fuel and Waste To and From One Light-Water-Cooled Nuclear Power Reactor," in plant-specific license renewal reviews; (2) the conditions that must be met before an applicant may adopt Table S–4; and (3) the extent to which the generic effects of transporting spent fuel to a HLW repository should be considered in a plant-specific license renewal review.

After considering the comments received on the rule, the Commission republished the rule in the **Federal Register** on December 18, 1996 (61 FR 66537). The rule at 10 CFR 51.53(c)(3)(ii)(M) continued to require, "The environmental effects of transportation of fuel and waste shall be reviewed in accordance with 10 CFR 51.52." However, in response to comments received, the following requirement was added:

> The review of impacts shall also discuss the generic and cumulative impacts associated with transportation operation in the vicinity of a high-level waste repository site. The candidate site at Yucca Mountain should be used as a representative site for the purpose of impact analysis as long as that site is under consideration for licensing.

Also in response to the comments, the Commission stated that:

> As part of its effort to develop regulatory guidance for this rule, the Commission will consider whether further changes to the rule are desirable to generically address: (1) the issue of cumulative transportation impacts and (2) the implications that the use of higher burnup fuel have for the conclusions in Table S–4. After consideration of these issues, the Commission will determine whether the issue of transportation impacts should be changed to Category 1.[1]

In SECY–97–279, titled "Generic and Cumulative Environmental Impacts of Transportation of High-Level Waste (HLW) in the Vicinity of a HLW Repository," dated December 3, 1997, the NRC staff informed the Commission that it was the staff's preliminary view that its supplemental analyses of the generic and cumulative impacts of the transportation of HLW and of the implications of higher burnup fuel for transportation impacts support a reasonable technical and legal determination that transportation of HLW is a Category 1 issue and may be generically adopted in a license renewal application. In a Staff Requirements Memorandum (SRM) dated January 13,

[1] In NUREG–1437 and in the rule, Category 1 issues are those environmental issues for which the analysis and findings have been determined to be applicable to all nuclear power plants or to plants with specific types of cooling systems or other common plant or site characteristics. Absent new information that significantly changes the finding, these generic findings may be adopted in plant license renewal reviews. Category 2 issues are those that analysis has shown that one or more of the criteria of Category 1 cannot be met and, therefore, additional plant-specific review is required.

1998, the Commission directed the NRC staff to proceed with rulemaking to amend 10 CFR 51.53(c)(3)(ii)(M) to categorize the impacts of transportation of HLW as a Category 1 issue. In a memorandum dated July 1, 1998, the NRC staff informed the Commission of its plans for amending 10 CFR part 51.

In that memorandum the NRC staff also proposed, as an administrative amendment, to address local traffic impacts attributable to continued operation of the plant during the license renewal term. This issue was identified as a Category 2 issue in NUREG–1437, Section 4.7.3.2 and the overall issue of transportation was designated as Category 2 in the rule (see 10 CFR Part 51, Subpart A, Appendix B, Table B–1, "Public Services, Transportation"). However, the specific issue of local transportation impacts during the renewal term was inadvertently omitted from 10 CFR 51.53(c)(3)(ii)(J) and its inclusion in Table B–1 is not explicitly stated. The basic transportation concern identified in NUREG–1437 is the potential adverse contribution of a larger plant work force to traffic flow in the vicinity of the power plant.

To address the above issues, the Commission issued proposed amendments to 10 CFR part 51 on February 26, 1999 (64 FR 9884), and provided a public comment period of 60 days. The supplemental analysis, which supports this rule, is reported in NUREG–1437, Vol. 1, Addendum 1, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Main Report Section 6.3—'Transportation,' Table 9.1 'Summary of findings on NEPA issues for license renewal of nuclear power plants,' Final Report." The draft for comment was published in February 1999 and the final report is expected to be published in August 1999.

The public comment period closed on April 27, 1999. Extensive public comments were received, including concerns by some commentors about the length of the comment period. Although the NRC did not extend the public comment period, the NRC staff did consider comments dated as late as June 25, 1999, and received as late as early July 1999. The NRC staff's responses to the comments are provided below. As explained in more detail below, the comments have led to both the use of more conservative assumptions in the analysis reported in Addendum 1 and a fuller explanation of the analysis. The regulatory text has been edited for clarification but there is no material change from the proposed rule.

Discussion

Relationship of This Rulemaking to Repository Licensing

The NRC is promulgating this rule in order to meet its National Environmental Policy Act (NEPA) responsibilities to consider the environmental impact of its license renewal decisions. In 1996 (61 FR 28467 and 61 FR 66537), the NRC published a rule that codified conclusions regarding the environmental impacts of license renewal (see 10 CFR part 51, Appendix B to subpart A). The amendment issued in the present Notice constitutes a relatively small addition to those previously published conclusions. In particular, as discussed above, this amendment ensures among other things that the NRC has considered the likely impacts of transporting spent fuel generated during the license renewal period over a single transportation corridor in the vicinity of a waste repository.

Because the Yucca Mountain site in Nevada currently represents the most likely candidate for a repository, the NRC has used that site as a representative site for its analysis in lieu of considering transportation to an unspecified, hypothetical site. The decision to use Yucca Mountain for the purposes of the current analysis, however, in no way increases or decreases the likelihood that Yucca Mountain will in fact be licensed as a repository for the nation's high level waste. Instead, it simply provides the NRC with the information it needs to gauge the potential impacts from licensing nuclear power plants for an additional 20 year period. If an application is filed by the Department of Energy (DOE), the licensing process for a repository in the vicinity of Yucca Mountain will constitute an entirely separate regulatory action from this final rule. Any NRC decision on a repository license will be accompanied by separate safety and environmental analyses that will include a thorough examination of the environmental impacts stemming from the construction and operation of the repository. If the analyses prepared for the repository licensing decision yield results that are inconsistent with those reached in the present notice, it is likely that the NRC will have to amend the conclusions in Table B–1 of Part 51 to conform with the new findings.

Amendments to the Rule

The current regulations require each applicant for license renewal to review the environmental effects of transportation of fuel and waste in accordance with 10 CFR 51.52, and to discuss the generic and cumulative impacts associated with transportation in the vicinity of the candidate HLW repository site at Yucca Mountain (see 10 CFR 51.53(c)(3)(ii)(M)). The NRC staff has performed a generic assessment of these cumulative impacts, which is reported in NUREG–1437, Vol. 1, Addendum 1. The analysis focused on Clark County, Nevada because it represents the area with the largest population in the vicinity of the potential repository. The final rule codifies the conclusions of this analysis in 10 CFR Part 51. In addition, the NRC staff has generically considered the potential impacts of transporting higher enriched and higher burnup fuel than is currently covered in 10 CFR 51.52 and is codifying these findings with this final rule. That assessment concludes that the impacts of transporting fuel and waste generated during the license renewal period are small and are consistent with the impacts of the values in Table S–4 of the Commission's regulations (§ 51.52). Under the Commission's regulations for the environmental review of license renewal decisions (see 10 CFR part 51, subpart A, appendix B), the Commission may reach a conclusion of "small" impact for a particular issue if the:

> * * * environmental effects are not detectable or are so minor that they will neither destabilize nor noticeably alter any important attribute of the resource. For the purposes of assessing radiological impacts, the Commission has concluded that those impacts that do not exceed permissible levels in the Commission's regulations are considered small as the term is used in this table.

The final rule amends the issue of transportation of fuel and waste from Category 2 to Category 1. In order to reach this Category 1 conclusion on an issue and thus not require site specific analysis of the issue pursuant to § 51.53(c)(3)(i), the Commission has made the following findings in accordance with the definitions set out in 10 CFR Part 51, Subpart A, Appendix B:

(1) The environmental impacts associated with the issue have been determined to apply either to all plants or, for some issues, to plants having a specific type of cooling system or other specified plant or site characteristic;

(2) A single significance level, in this case "small" has been assigned to the impacts (except for collective off site radiological impacts from the fuel cycle

and from high level waste and spent fuel disposal [2]); and

(3) Mitigation of adverse impacts associated with the issue has been considered in the analysis, and it has been determined that additional plant-specific mitigation measures are likely not to be sufficiently beneficial to warrant implementation.

As a result of this Category 1 finding, neither applicants nor the NRC staff will need to prepare a separate analysis of the issue for individual license renewal applications as long as no new and significant information exists. The analysis in NUREG-1437, Vol. 1, Addendum 1 which forms the technical basis for the rulemaking, relies on a series of conservative assumptions. As such, the results of the analysis overestimate the environmental impacts of spent fuel shipments converging on one location, such as Yucca Mountain. Although the NRC staff has assessed these impacts as if Yucca Mountain would be the only HLW repository, the NRC staff believes that the impacts calculated for Yucca Mountain bound the impacts that would be experienced for a site other than Yucca Mountain. It is unlikely that any other repository site would have an exposed population greater than that assumed for Las Vegas and it is unlikely that spent-fuel shipments from all points of origin converge on and are transported through one metropolitan area. If an alternative to a high level waste repository at Yucca Mountain is considered in the future, the NRC may need to determine whether such an alternative includes new and significant information that may change the regulatory outcome.

In addition to considering the cumulative impacts of transportation in the vicinity of a repository, the NRC also considered whether use of higher burnup or higher enriched fuel that is shipped to a repository results in impacts consistent with the NRC regulations (§ 51.52, 'Table S–4—Environmental Impact of Transportation of Fuel and Waste To and From One Light-Water-Cooled Nuclear Power Reactor'). The environmental consequences of incremental increases in the burnup of fuel and the associated use of higher enrichment fuel are discussed in Section 6.2.3 of NUREG-1437. Section 6.2.3 addresses the sensitivity of the data presented in Table S–3 and Table S–4 to the growing use of higher enriched fuel and higher fuel burnup. Table S–3 summarizes natural resource use and effluents to the environment for the uranium fuel cycle, from mining to ultimate disposal of spent fuel. The discussion of the implications for the environmental impact data reported in Table S–4 was not repeated or referenced in Section 6.3, which addresses the incremental impacts of license renewal on the transportation of fuel and waste to and from nuclear power plants. Addendum 1 and this final rule clarify the NRC findings on the sensitivity of values in Table S–4 to the use of higher enrichment fuel and higher burnup fuel presently in use. The analysis concludes that shipment of higher enriched or higher burnup fuel results in impacts consistent with the impacts in Table S–4, 10 CFR 51.52. It should be noted that cask designs used to transport or store higher enriched fuel and higher burnup fuel require specific NRC review and approval.

In the course of preparing the final rule, several non-substantive changes to the wording and organization of the regulatory text were made in order to maintain the rule's internal consistency. First, the content of the proposed language in § 51.53(c)(3)(ii)(J) regarding local transportation impacts in the vicinity of the licensed plant was also placed into Table B–1 under "Public Services, Transportation" under the Socioeconomics section of the Table. Similarly, the proposed language in § 51.53(c)(3)(ii)(M) has not been included in the final rule because the matters covered by § 51.53(c)(3)(ii) only apply to Category 2 issues and, as such, the inclusion of matters related to a Category 1 issue in that section would not have been appropriate. Instead, the content of the language that had been proposed for § 51.53(c)(3)(ii)(M) is adequately covered by the amended entry in Table B–1 itself under the issue of "Transportation" in the Uranium Fuel Cycle and Waste Management section.

Response to Comments

Thirty-one comment letters were received on the proposed rule from power reactor licensees, State and local Government agencies, the nuclear power industry and its legal affiliations, a public interest group, and an individual. Most of the comments were from the State of Nevada, Clark and Nye Counties, Nevada, and local government entities in Nevada. These comments focused on the NRC not involving Nevada in scoping and designing the study in Addendum 1 and on perceived deficiencies in the scope and thoroughness of the analysis in the Addendum. The State of Utah also submitted extensive comments that focused on concerns with the scope and thoroughness of the supporting analysis in Addendum 1, including the lack of consideration of the proposed Private Fuel Storage Facility at Skull Valley, Utah. Industry comments focused on clarifications in the rule language.

The written comments have been summarized and grouped into issue categories. As a result of the NRC staff's review of all written comments, some modifications and clarifications have been incorporated into Addendum 1—notably, the use of more conservative assumptions in the analyses and a fuller explanation of those analyses. In addition, the rule language has been edited for clarification. The NRC staff has also prepared responses, given below, to the issues raised by the commentors.

Issue 1—Public Notice

Comment: The titles of the notices published in the **Federal Register** were inaccurate and misleading because they do not clearly indicate the subject matter of the proposed rule and Addendum 1 that addresses transportation of spent nuclear fuel.

Response: The NRC believes that the titles properly reflect the regulatory action being taken. As required by NRC regulations,[3] a notice of the proposed rule and a Notice of Availability of Addendum 1 were published in the **Federal Register** (64 FR 9884 and 64 FR 9889, February 26, 1999). While the notice's title did not include the specific term "transportation," the titles define the subject matter of the regulation to be affected; the title of the proposed rule is "Changes to Requirements for Environmental Review for Renewal of Nuclear Power Plant Operating Licenses." The title of the Notice of Availability is "Changes to Requirements for Environmental Review for Renewal of Nuclear Power Plant Operating Licenses, Availability of Supplemental Environmental Impact Statement." Addendum 1 supplements specific sections of NUREG–1437, Generic Environmental Impact Statement for License Renewal of Nuclear Plants (May 1996). This limited function is indicated by the title of Addendum 1, Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Main Report Section 6.3—"Transportation," Table 9.1 "Summary of findings on NEPA issues

[2] This exception only applies to the two entries in Table B–1 labeled "Offsite radiological impacts (collective effects)" and "Offsite radiological impacts (spent fuel and high level waste disposal).

[3] 10 CFR 2.804, "Notice of proposed rulemaking" and 10 CFR 51.117, "Draft environmental impact statement'notice of availability."

for license renewal of nuclear power plants," Draft Report for Comment.

The rule change and the supporting Addendum 1 affect only the plant-specific environmental analysis required to be submitted in the Environmental Report of an applicant for the renewal of a nuclear power plant operating license and the plant-specific supplemental environmental impact statement prepared by the NRC. Even though the analysis in Addendum 1 focuses on spent-fuel shipments converging on the proposed repository at Yucca Mountain, Nevada, that analysis and the resulting rule affect only the review requirements for renewal of an individual nuclear power plant operating license. It is not intended that Addendum 1 or the revised rule support any other regulatory decision by the NRC.

Issue 2—Communications

Comment: NRC failed to consult with Nevada State agencies, Nevada local governments, and with Nevada Indian Tribes.

Response: As discussed above, a variety of organizations and government agencies submitted substantive comments in response to the proposed rule. The NRC has considered these comments and, in many cases, altered its analysis as a result of this input. Prior to issuance of the proposed rule for comment, however, the NRC did not seek any pre-publication input from Nevada state agencies, Nevada local Governments, and Nevada Indian Tribes for the following reasons. First, the rule involves a narrow aspect of the environmental review of individual nuclear power plant license renewal decisions, which is a regulatory decision completely separate from the regulatory requirements that will guide the NRC licensing review of a HLW repository and from the decision process leading to a DOE site recommendation on Yucca Mountain, Nevada, the site DOE currently has under study. This rule amends the December 18, 1996, rule with respect to two questions not adequately answered:

1. Are the current environmental impact values in Table S–4, based on several destinations, still reasonable to incorporate in a license renewal review that assumes a single destination for spent fuel at Yucca Mountain, Nevada?

2. Are the current environmental impact values in Table S–4 (which are based on fuel enriched to no greater than 4 percent, the average level of irradiation of spent fuel not exceeding 33,000 MWd/MTU, and shipment no less than 90 days after discharge from the reactor) still reasonable to incorporate in a license renewal review of plants that may use fuel enriched up to 5 percent and potentially ship spent fuel with a burnup of up to 62,000 MWd/MTU?

The amendment has no direct regulatory impact on any entity within Nevada. The selection of Yucca Mountain for the generic evaluation of transportation impacts was made because that site is currently the only one under consideration for a high-level-waste (HLW) repository. Before HLW is actually transported to Yucca Mountain, Nevada, the State, local Governments, Indian Tribes, and the public have the opportunity to provide input on site-specific transportation impacts by commenting on DOE's draft EIS for the proposed repository at the Yucca Mountain site, which was made available for a 180-day comment period beginning on August 13, 1999 (http://www.ynp.gov).

Also, the need for and scope of the current rule amendment were identified within the context of a preceding rulemaking that specified the plant-specific content of the environmental review of applications for the renewal of individual nuclear power plant operating licenses. The previous final rule was published in the **Federal Register** first on June 5, 1996 (61 FR 28467), and again with minor modifications on December 18, 1996 (61 FR 66537). The Commission stated in the December **Federal Register** notice, "as part of its efforts to develop regulatory guidance for this rule, the Commission will consider whether further changes to the rule are desirable to generically address: (1) The issue of cumulative transportation impacts and (2) the implications that the use of higher burn-up fuel have for the conclusions in Table S–4. After consideration of these issues, the Commission will determine whether the issue of transportation impacts should be changed to Category 1."

Issue 3—Transportation Analysis

Comment: NRC failed to consult relevant Yucca Mountain transportation risk and impact studies.

Response: The publications cited by commentors have been reviewed for information that may be of direct use within the limited focus and purpose of the current rule. Most of the information in these documents was found to be potentially more relevant to a detailed site-specific review of Yucca Mountain than to the generic analysis for this rule. That information has been brought to the attention of those organizational units within the NRC responsible for activities relating to DOE's study on the Yucca Mountain site so they can appropriately consider the information in any future prelicensing activities involving Yucca Mountain. Specific to the current rule, the demographic data used as inputs to the RADTRAN computer code, which was used to generate the impact analysis in Addendum 1 were more current than data used in many of the studies cited by the commentors.

Comment: NRC failed to consult the full spectrum of transportation mode and route scenarios.

Response: The purpose of this rule and associated analysis is to reach conclusions regarding the likely environmental impact of license renewal. As noted above, this amendment is an addition to generic assessments of license renewal environmental impacts already codified in the Commission's regulations at 10 CFR part 51, subpart A, appendix B. It is not an environmental impact statement for a repository at Yucca Mountain for which DOE is responsible and, as such, does not delve into the expansive range of different transportation modes and route scenarios that would be considered in the context of a decision on Yucca Mountain as the possible site for the facility itself. Instead, the NRC has sought to determine a conservative estimate of the likely impacts from transporting fuel and waste generated, during the license renewal term, in the vicinity of a potential repository. In doing so, the NRC considered only those transportation modes and route scenarios that would likely result in the greatest impacts. For the proposed rule, the NRC staff—in consultation with the DOE staff—determined that truck shipments through densely populated areas of Clark County, Nevada, would have the highest potential impacts among the alternative transportation scenarios and modes that would receive serious consideration in decisions relating to the suitability of the site undergoing study for a repository at Yucca Mountain. The NRC continues to believe that using these route scenarios and modes to generate conservative estimates is reasonable for the purpose of this rulemaking.

Comment: There was insufficient consideration of routine transportation radiological risks due to use of an average dose rate lower than the regulatory limit.

Response: The RADTRAN analysis reported in the final Addendum 1 has been modified to use the most conservative assumption that the radiation levels for all shipments are at the regulatory limit of 0.1 mSv/hour [10 mrem/hour] at 2 m [6.6 ft] from the shipment vehicle surface. As noted in Section 2.2.3 of Addendum 1, this assumption is sufficiently conservative to bound the analysis of routine transportation radiological risk and allow a reasonable assessment of that risk. Actual average radiation levels and associated doses would be much lower because shipments must be designed so that the regulatory limits are not exceeded. The use of the regulatory limits in the revised analysis results in higher dose estimates for incident-free transportation. However, these revised estimates are still small as defined in 10 CFR Part 51, Subpart A, Appendix B. Consequently, the conclusion regarding the radiological risks of routine transportation remains valid.

Comment: There was insufficient consideration of routine transportation radiological risks to members of the public residing, working, or institutionally confined at locations near shipping routes.

Response: The analysis encompasses members of the public residing, working, or institutionally confined at locations near shipping routes by assuming that the resident population along the transportation routes is exposed to every shipment. The text of Sect. 2.3 of Addendum 1, has been revised to state this assumption and its effects on the revised analysis more clearly. In addition, more conservative assumptions of truck speed have been used in the revised RADTRAN analysis thus extending the exposure time to individuals along the transportation route. These assumptions further ensure that members of the public cited by the commentors would be encompassed by the dose and risk assessments. As expected, the use of these more conservative assumptions leads to higher estimates of radiation dose to the public. However, these revised dose estimates remain well below regulatory limits for members of the public and small compared to natural background and other sources of radiation exposure.

Several commentors indicated that Addendum 1 should focus on unique and location-specific circumstances of the transportation routes and population centers. However, the analysis in Addendum 1 is generic and was designed to support only the limited scope of the decision regarding this rule change. The NRC believes that the routes chosen represent a conservative analysis due to the higher number of people who live along these routes. Because the purpose of this rule is to provide a generic analysis for the limited purpose of determining the likely impact of transportation during the license renewal term, the large analytical effort required for the identification of specific population locations and traffic circumstances is not warranted within the context of the current rule. Although the comments raise valid issues, those concerns should be resolved within the context of studying, and making decisions concerning, the suitability of the candidate repository site at Yucca Mountain and regulatory requirements governing transportation of spent fuel.

Comment: There was insufficient consideration of radiological risks resulting from traffic gridlock incidents.

Response: Traffic gridlock incidents are not specifically analyzed in NUREG–1437 because of the limited scope and generic nature of the analysis (see response to comment on consideration of risks to members of the public, above). However, the revised RADTRAN analysis conservatively includes approximately two hours of stationary time in Clark County (during a 100 to 140 mile trip depending upon the route) for each truck shipment; and traffic gridlock could be one of the reasons for the truck being stationary.

To a limited extent, the incorporation of more conservative assumptions of truck speed into the revised RADTRAN analysis compensates for an analysis of traffic gridlock by allowing for increased exposure time at any given point during transport. As noted earlier, these revised assumptions lead to higher but still small dose estimates. In addition, the routes used in the analysis in Addendum 1 were deliberately chosen to maximize estimated dose. Actual routes would be less likely to have significant areas where traffic gridlock occurs. The selection of the actual routes, for example, would comply with the U.S. Department of Transportation's Federal Highway Administration regulations (49 CFR Part 397, Subpart D) that require minimizing the time in transit (i.e., avoiding periods of great traffic congestion) for routing radioactive shipments.

Comment: There was insufficient consideration of routine transportation radiological risks to vehicle inspectors and escorts.

Response: The RADTRAN analysis in the revised Addendum 1 uses the regulatory dose rate limit of .02 mSv/hour (2 mrem/hour) for the vehicle crew. In addition, a discussion of potential doses to escorts has been included in Addendum 1, Section 2.2.3. In the analysis, both the escorts and drivers are assumed to be exposed to the regulatory limit, although the dose to the escorts would realistically be less than that to the drivers. Even with these more conservative assumptions, the estimated dose and risk to the crew are small and below regulatory limits.

The risk to vehicle inspectors would be encompassed by the addition of stationary time for the transport truck in Clark County (see response to comment about traffic gridlock, above). Again, the estimated dose and risk are increased by the use of more conservative assumptions; but they remain small and below regulatory limits.

Comment: There was insufficient consideration of severe transportation accident risks.

Response: The Commission has evaluated the potential radiological hazards of severe transportation accidents involving truck and rail spent nuclear fuel (SNF) shipments (NUREG/CR–4829, "Shipping Container Response to Severe Highway and Railway Accident Conditions" February 1987, commonly referred to as the modal study). The modal study evaluated SNF shipping casks certified to NRC standards against thermal and mechanical forces generated in actual truck and rail accidents. This evaluation included an assessment of cask performance for a number of severe transportation accidents, including the Caldecott Tunnel fire. The modal study concluded that there would be no release in 994 of 1,000 real accidents, and that a substantially lower fraction of accidents could result in any significant release. These results when combined with the probability of a severe accident involving a shipment of SNF, demonstrate that the overall risk associated with severe accidents of SNF shipping casks is very low. The results of the modal study were factored into the analysis for this rulemaking, as an input to the RADTRAN computer code. Additional analyses were performed to address the possible impacts of accidents involving higher burnup fuel.

The consequences associated with an individual SNF shipment have an upper bound, based on the amount of material in the package, the availability of mechanisms to disperse the radioactive contents, the locations and number of receptors, and post-event intervention than would occur. Further, this upper bound in transit might reasonably be expected to be less than that at the origin or destination points (where more SNF would be stored), and some events themselves might be expected to have greater consequences than the damage they cause to the SNF cask. The NRC recognizes that there are some conceivable events (not necessarily traditional 'transportation accidents'), that might be hypothesized to occur to a SNF cask while in transport. Even though these events have an extremely low probability of occurring, they might result in high consequences if they were to occur. The NRC considers these events to be remote and speculative and thus, does not call for detailed consideration. Because the NRC traditionally considers risk to be the product of the probability of an event and its resultant consequences, events with such low probability of occurring have a negligible contribution to the overall risk. In addition, as the probabilities of the events become very low, the value of insights to be gained, for use in regulatory decisions, is not apparent.

Comment: The study underestimates Clark County's residential population and growth rate. In addition, the study does not account for the large nonresident population, resulting in underestimates of risk and impacts.

Response: In keeping with the generic nature and limited intent of the analysis, the original analysis used best available data and best estimates of existing population and population growth rates. In response to commentors' concerns and to reflect the potentially large population growth rate of Clark County, the NRC staff has incorporated higher population estimates into the analysis to provide conservative (higher than best estimate) assessments of potential impacts. However, as indicated by the comment, the task of estimating the impacts on the area population is more complex than assuming a population growth rate. Both the rate of growth of the population and changes in location of the population within the county are important. As stated in Addendum 1, populations within a half mile of the transportation route are the most affected by the transportation activities. Therefore, in order to ensure that the size of the affected population is conservative, the NRC staff's analysis not only increases over time the existing population densities along the assumed transportation routes, but also forecasts increased residential, business, and transient/tourist populations in the areas of likely development.

Issue 4—Cumulative Impacts

Comment: NRC failed to consider cumulative impacts of all spent fuel, HLW, and low-level-waste shipments.

Response: Table S–4 shows the environmental impacts of transportation of fuel and waste directly attributable to one nuclear power plant. The current rulemaking was narrowly focused on the question of whether the impact values given in Table S-4 would be different with spent fuel shipments converging on one destination, Yucca Mountain—the candidate site under study by DOE for a repository, rather than several destinations. Table S–4 does not consider non-commercial power reactor shipments of fuel and waste. Nevertheless, a discussion of the cumulative impacts of transporting spent fuel, HLW, and low-level waste through southern Nevada has been added to Addendum 1 (Section 2.4). To estimate the potential cumulative effects of DOE shipments of LLW to the Nevada Test Site as well as shipments of HLW to a possible repository, the NRC staff used information published in DOE's Waste Management Programmatic EIS (DOE/EIS–0200—F) May 1997. To ensure that cumulative impacts are not underestimated, the NRC staff selected alternatives in the EIS that led to the highest numbers of shipments to the Nevada Test Site and Yucca Mountain. The results of the analysis indicate that the cumulative doses and expected cancer fatalities resulting from the civilian SNF and the DOE shipments are small compared to the risk of cancer from other causes.

Comment: Commentors stated that cumulative impacts along the Wasatch Front must be considered.

Response: The State of Utah maintains that a study similar to the one conducted for Las Vegas and Clark County must be conducted for the cumulative impacts along the Wasatch Front that would originate from the proposed Private Fuel Storage Facility to be located at Skull Valley, Utah. Such an analysis is beyond the scope of this generic rulemaking because the Commission directed that cumulative impacts attributed to transportation be analyzed only in the vicinity of Yucca Mountain. However, the NRC is currently reviewing a site-specific application for construction and operation of the proposed Private Fuel Storage Facility at Skull Valley in a separate regulatory action. A site-specific study of the cumulative impacts of transportation is part of that review. The study will be reported in a draft Environmental Impact Statement to be published for public comment. Its availability will be noticed in the **Federal Register**.

Issue 5—Legal Requirements

Comment: NRC failed to conduct a legally sufficient risk assessment. Use of a model such as RADTRAN is not in and of itself sufficient to meet the requirements of the National Environmental Policy Act. The NRC must consider consequences of low-probability, high-consequence accidents not included in RADTRAN, including unique local conditions, unforeseen events, sabotage, and human error in cask design. The NRC should adopt the comprehensive risk assessment approach for SNF and HLW transportation described in Golding and White, Guidelines on the Scope, Content, and Use of Comprehensive Risk Assessment in the Management of High-Level Nuclear Waste Transportation (1990).

Response: See the response above regarding consideration of severe accident risk (low probability, high consequence accidents) during transportation.

The NRC's regulatory program will continue to ensure that the risk of severe transportation accidents are minimized. Physical security for spent fuel transportation is regulated under 10 CFR 73.37. The regulatory philosophy is designed to reduce the threat potential to shipments and to facilitate response to incidents and recovery of packages that might be diverted in transit. Although the analysis supporting the current rule does not account for the potential for human error, activities related to the design, fabrication, maintenance, and use of transportation packages are conducted under an NRC-approved Quality Assurance Program. This helps to provide consistency in performance and helps reduce the incidence of human error. While a location-specific transportation risk assessment is included in the DOE EIS for the decisions relating to a possible Yucca Mountain repository, the NRC staff believes that the analysis conducted for this rulemaking provides an adequate consideration of the impacts from license renewal. Further, through its regulatory, licensing, and certification functions, the NRC has tried to ensure that transportation of SNF is performed safely with minimum risk to the public, and that vehicle crashes while transporting SNF do not result in severe accidents. Similarly, DOE is expected to ensure that the routes and procedures chosen for SNF transport to the repository provide ample protection of the public health and safety and the NRC reviews and approves the selected routes.

The analysis in Addendum 1 shows that even with conservative assumptions, the cumulative radiological and non-radiological accident risks of SNF transport in Clark County are small. However, there are a number of opportunities to further reduce human health impacts. These include transporting SNF by rail rather than by truck. This would reduce human health effects by reducing the number of shipments and the likelihood of accidents. In addition, shipping SNF via the proposed beltway would reduce health impacts compared to shipping via the current interstate highway system. The implementation of such mitigative measures must await future decisions that fall well outside of the scope of this rulemaking. In addition, for the purposes of individual license renewal rule decisions, no plant specific mitigation measures were found appropriate for addressing the impacts identified in the Addendum. The NRC staff notes that DOE addresses transportation impacts, mitigation measures, and alternative transportation modes in its EIS for the proposed repository at Yucca Mountain.

Issue 6—Socioeconomics

Comment: NRC failed to consider socioeconomic impacts.

Response: Several commentors raised an issue of public perception of risk of waste shipments and its effect on tourism and property values. Under the National Environmental Policy Act (NEPA), the NRC is obligated to consider the effects on the physical environment that could result from the proposed action. Effects that are not directly related to the physical environment must have a reasonably close causal relationship to a change in the physical environment. The Supreme Court ruling in *Metropolitan Edison Co.* v. *People Against Nuclear Energy,* 460 U.S. 766 (1983) has narrowly circumscribed, if not entirely eliminated, an agency's NEPA obligation to consider impacts arising solely from the public's perception that an agency's action has created risks of accidents. Accordingly, it is not necessary to consider the impacts on tourism and property values from the public's perception of risk.

The socioeconomic impacts of plant refurbishment and continued operation during the renewal period are discussed in the plant-specific supplement to the GEIS for each individual license renewal applicant. The NRC recognizes that there will likely be increased costs in the unlikely event of an accident. However, for the majority of transportation accidents that may occur, the associated costs are small. For the most severe accidents analyzed by the RADTRAN computer code, the costs could be substantial. Given the low probability of such accidents, the socioeconomic impacts of transportation of SNF do not alter the Commission's conclusions regarding the impacts of this issue.

Issue 7—Higher Burnup Fuel

Comment: There was insufficient consideration of extended fuel burnup issues.

Response: Section 3 of Addendum 1 addresses the issues associated with extended fuel burnup in detail. The NRC staff's analysis of higher burnup fuel examined the issues of radiation doses due to higher dose rates during shipment, higher radiation doses in the event of transportation accidents, and the potential for a criticality in the very unlikely event that high burnup fuel geometry is altered during a transportation accident.

The analysis done by the NRC staff concluded that higher burnup fuel would likely cause higher dose rates during transportation and that dose rates following transportation accidents with radiological releases would also increase, all other things being equal. However, despite the increased dose rates the potential impacts on the transport crews and the affected members of the public would still be acceptably small. The analysis of the potential for criticality following a change in fuel geometry as the result of a transportation accident determined that such an event was not a concern.

Issue 8—Environmental Justice

Comment: NRC failed to consider Environmental Justice.

Response: The analysis suggests that the routes through downtown Las Vegas, Nevada may run through areas containing a higher proportion of low-income and minority groups than the beltway routes. However, as discussed in Sections 2.3 and 2.4 Addendum, the radiological and nonradiological impacts of transportation of SNF are small. In addition, these small impacts are dispersed throughout the entire routes and do not appear to fall disproportionately in any one area. Based on the analysis performed the NRC staff concludes the overall impacts of transportation of SNF will not likely be disproportionately high or adverse for any minority or low-income population.

Issue 9—Regulatory Text

Comment: Several suggestions for clarifying the regulatory text were offered.

Response: The rule has been revised to make it clear that the environmental impact values in Table S–4 (10 CFR 51.52) may be used to account for the environmental effects of transportation of fuel and waste to and from a nuclear power plant at a repository such as Yucca Mountain, Nevada, which is under consideration as a HLW repository. If, in the future, Yucca Mountain is removed from consideration as a HLW repository, the Commission will evaluate whether the generic analysis performed for the current rule is applicable to other sites that are considered. If fuel enrichment greater than 5 percent Uranium-235 and fuel burnup of greater than 62,000 MWd/MTU are approved by the Commission, the Commission will consider a rulemaking to assess the continuing generic applicability of Table S–4 to environmental reviews for license renewal.

Comment: The addition to the rule of local transportation impacts associated with continued operation of a plant during the license renewal period needs further clarification in the rule language and in the Supplementary Information.

Response: The rule was revised to clarify that the issue of "Public services, Transportation" in Table B–1 of Appendix B to Subpart A of 10 CFR Part 51 involves the contribution of highway traffic directly attributable to refurbishment and continued operation of a plant during the license renewal period to changes in the service levels of highways in the vicinity of the plant. The majority of traffic directly attributable to a plant is commuting plant workers.

Comment: Paragraph (M) of 10 CFR 51.53(c)(3)(ii) should be deleted.

Response: The rule language has been amended and Paragraph (M) has been deleted. This change from the proposed rule was necessary in order to provide consistency with 51.53(c)(3)(ii), as this section only deals with Category 2 issues. Since the cumulative impacts of transportation of SNF in the vicinity of Yucca Mountain is no longer a Category 2 issue, inclusion in 51.53(c)(3)(ii) is no longer necessary.

Other Comments

This section addresses the comments that are not encompassed by the issue summaries and responses given above. In addition, some comments were received after the close of the comment period. These comments were reviewed, and most were found to be similar to comments already addressed by the issue summaries and responses. However, the comments that raised new ideas relevant to Addendum 1 are also presented in this section. For these late comments, revisions to Addendum 1 were necessarily minimal.

Comment: Addendum 1 assumes that truck transport would have the highest doses. This assumption is not necessarily valid. Also, a different route that avoids Las Vegas should be addressed. (A route through Nellis Air Force Base and down US–95 is being considered by DOE and it has been shown to have higher risks of accident fatalities and to increase the radiological risk.) Routes chosen in Addendum 1 do not bound the analysis properly.

Response: The transportation and route scenarios and their underlying assumptions were designed to reflect situations that most likely would result in highest doses in order to bound the analysis properly as the routes chosen for this analysis were the most populated routes in the State of Nevada. Also, as noted in an earlier response, the NRC staff consulted DOE in determining that truck shipments through densely populated areas of Clark County, Nevada, would have the highest potential impacts among the alternative transportation scenarios that would be given serious consideration in decisions relating to the suitability of the site undergoing study for a repository at Yucca Mountain.

The comment that a route from Nellis Air Force Base down US–95 is higher risk than those selected by the NRC staff provided no specific details concerning that assertion. In the NRC staff's view, any route that bypasses major centers of population will have significantly lower radiological impacts. With regard to traffic accident rates, while it may be true that certain routes will have accident rates that are higher than average, the average rates are low enough that modest increases from the average will not significantly change the staff's conclusions.

Comment: SNF from California would go through Las Vegas twice (in route to Skull Valley and subsequently to Yucca Mountain), resulting in increased risk.

Response: If the proposed SNF storage facility is licensed and built, some SNF may go through Clark County on the way to Skull Valley, Utah. The NRC staff has not analyzed this possible impact because it is not clear at this time that the proposed Skull Valley facility will be licensed or that the SNF would go through Las Vegas if the facility were built. In addition, SNF from California makes up only a small fraction of the SNF that would be shipped. The NRC staff concludes that the conservative assumptions used in the analysis more than compensate for minor changes in transportation plans that may develop for that fraction of the total SNF.

Comment: The NRC should provide affected parties with some statement of the regulatory effect of the interrelationships between the numerous other similar analyses.

Response: As a general matter, the National Environmental Policy Act (NEPA) requires all Federal agencies to perform an environmental review for certain actions they propose to conduct. In the context of nuclear waste management, several agencies have regulatory and operational responsibilities which may involve various proposed actions that, in turn, require the preparation of environmental impact statements (EISs). Inevitably, there may be a degree of overlap in the types of impacts discussed in these various EISs. However, the analysis developed by the NRC for the purposes of license renewal is not binding on future actions and associated environmental impact analyses.

The NRC proposed action that has triggered the preparation of this rulemaking and the associated analysis of environmental impact is the agency's responsibility to review applications for the renewal of nuclear power plant licenses. In light of the discrete purpose of this rulemaking, the NRC has sought to gauge the impacts of license renewal given the information currently available on those impacts including the transportation of spent fuel. Even though these impacts do not occur at the plant site during license renewal, the NRC has considered them here pursuant to its NEPA responsibilities.

Future EISs prepared by other agencies on proposed actions in the waste management arena (e.g., any recommendation by DOE on approval of the Yucca Mountain site for development of a repository) will undoubtedly address some of the same impacts covered by the analysis described in this notice. Some of these other impact statements are anticipated to be more detailed given their purpose and the availability of additional information in the future. This, however, does not diminish the adequacy of the NRC's action. This analysis is sufficient for the purpose it serves and it provides the Commission with the information needed to weigh the likely environmental impacts of SNF transportation for individual license renewals applications and reach informed decisions regarding the acceptability of these applications. The rule does not, however, dictate any particular result for future actions taken with regard to a waste repository or other waste management matters. Specifically, any generic conclusions by the Commission concerning the cumulative environmental impacts of transportation associated with nuclear power plants would in no way affect any DOE decision concerning the suitability of Yucca Mountain or any consideration that DOE may give to transportation impacts in making that decision.

Comment: Addendum 1 is not meaningful to the public. For example, it is impossible to determine if the spent fuel isotope inventory shown in the sample pages of the RADTRAN printout matches the fuel considered in the Addendum.

Response: In preparing Addendum 1, the NRC staff has attempted to write to a broad and diverse audience as much as possible. The NRC staff acknowledges that this rulemaking involves complicated, technical issues. However, the NRC staff has attempted to present these matters in the most clear manner possible. Addendum 1 has been revised and Table 2 provides the fuel isotope inventory that can be compared to the sample pages of the RADTRAN computer code printout.

Comment: The study area is inaccurately defined and the location of some cities is incorrectly stated.

Response: During the preparation of Addendum 1, the initial study area selected for analysis emphasized the urban areas in and near Las Vegas. Route selections were based in part on their proximity to those areas, not to county borders. However, in response to public comments, the study area was expanded to include the entire county. Consequently, the "entry" point for SNF shipments shifted to cities such as Mesquite.

Comment: Addendum 1 should discuss potential mitigation measures, not rely on the DOE Yucca Mountain EIS for that discussion.

Response: The analysis in Addendum 1 shows that, even with conservative assumptions, the cumulative radiological and non-radiological accident risks of SNF transport in Clark County are small. However, there are a number of opportunities to further reduce human health impacts. These include transporting SNF by rail rather than by truck. This would reduce human health effects by reducing the number of shipments and the likelihood of accidents. In addition, shipping SNF via the proposed beltway would reduce health impacts compared to shipping via the current interstate highway system. The implementation of such mitigative measures must await future decisions that fall well outside of the scope of this rulemaking. In addition, for the purposes of individual license renewal rule decisions, no plant specific mitigation measures were found appropriate for addressing the impacts identified in the Addendum. The NRC notes that DOE addresses transportation impacts, mitigation measures, and alternative transportation modes in its EIS for the proposed action to develop a repository at Yucca Mountain.

Comment: Addendum 1 does not mention that the proposed repository which is the destination for shipments of spent nuclear fuel is in Nye County.

Response: A statement noting that the proposed Yucca Mountain repository is in Nye County has been added to Addendum 1.

Comment: No statements of baseline conditions are given in Addendum 1.

Response: Addendum 1 uses background and natural radiation levels as the baseline conditions against which dose estimates can be compared. Both are presented in Addendum 1 and are based in large part on information published by the National Council on Radiation Protection and Measurements.

Comment: The analysis in Addendum 1 is limited to human health effects. Other potential impacts should be considered.

Response: Addendum 1 was prepared to provide information regarding a proposed rule to determine whether the transportation of higher enriched, higher burnup fuel to a single destination is consistent with the values of Table S–4. Because the pertinent section of Table S–4 concerns impact values for human health effects, Addendum 1 concentrates on potential cumulative impacts to human health. However, Section 2.3 of Addendum 1 has been revised to look at the potentially most significant non-human health effect which is the potential increase in traffic volume in Clark County as the result of the transportation of SNF. The NRC staff conclusion is that the impacts are small.

Comment: The analysis assumes the use of the large-capacity GA–4/9 truck cask, which has not been certified and must be used in combination with specially designed trucks that have not been tested. It also assumes that these cask and truck systems will be available in sufficient quantity for the shipments. The commentor seeks assurance that the assumed truck cask system is feasible and that DOE's proposed regional service contractor approach would feasibley result in the use of such a system for all shipments in the potential truck shipment campaign.

Response: The analysis done by the NRC staff assumes that an adequate number of certified casks would be available. Addendum 1 used extremely conservative assumptions regarding SNF shipments and casks to ensure that the analysis would lead to maximum dose estimates. For example, the analysis of incident-free transportation impacts assumes the use of legal-weight trucks for shipment of the SNF, which results in more and smaller shipments. For the accident analysis, the use of the largest-capacity casks was assumed in order to maximize the amount of SNF that would be involved in the accident. These parameters were intended to bound the parts of the analysis, not to describe parts of the actual SNF shipment protocol such as the specific casks that will be used.

Comment: The analysis appears to assume that oldest spent nuclear fuel would be shipped first to the repository. If so, how will institutional measures achieve this sequencing? If they do not, how will the maximum potential radioactive risk in shipment and storage or disposal be addressed?

Response: The spent fuel will be shipped in casks certified by the NRC. In fact, the current practice of NRC issuing certificates of compliance for casks used for shipment of power reactor fuel is to specify 5 years as the minimum cooling period in a certificate.

Comment: Addendum 1 uses national accident rate statistics. State and/or local rates would be more appropriate.

Response: For the analysis of radiological accidents, data specific to Nevada were used in the RADTRAN computer code runs. However, for the analysis of non-radiological accidents, the NRC staff required data regarding not only accident rates but also injury and fatality statistics. Those data were not available except from the U.S. Department of Transportation.

Comment: Water resource supplies within boundaries of the State of Nevada belong to the public. All waters are subject to appropriation for the beneficial use only under state law.

Response: The water resources of the state will be unaffected by the transport of SNF through Clark County.

Comment: Report failed to provide conditions for informed consent which requires disclosure to those affected, their understanding , and voluntary acceptance.

Response: NRC regulations already contain values that the NRC considers to be acceptable environmental impacts from the shipment of SNF and other radioactive waste. In Addendum 1 the NRC staff is, in part, ensuring that the overall impacts of the transportation of the additional SNF that will be generated as the result of nuclear power plant license renewal are bounded, given the best information the NRC staff has at this time, by those values previously found acceptable. The values specified in the regulations are supported by analysis and were adopted into the regulations only after providing opportunity for public comment as part of the NRC's rulemaking process. As such, the NRC has followed all applicable legal requirements and appropriately carried out its responsibility to consider the environmental impacts of its license renewal decision.

Comment: The NRC staff uses "flawed" science as evidenced by factors including a questionable definition of risk which fails to account for severe accidents, use of misleading if not false average radiation dose rates, manipulation of dose rate data to obtain acceptable results and lack of empirical data especially that applicable to transportation of SNF.

Response: The decision before the Commission is whether the impacts of license renewal are so severe that they should preclude the option of license renewal. As such, the Commission has considered a reasonable estimate of impacts and not included remote and speculative scenarios that do not add to our regulatory decision (see also response to comment on severe accidents, above).

In the analyses described in Addendum 1 the NRC staff uses dose rates that reflect the applicable regulatory limit rather than average dose rates. Even with these very conservative assumptions for dose rates, transportation modes, transportation routes, and a number of other factors, radiation impacts on the transport crews and the general public were not only found to be within all regulatory limits but small as well and there was no need to adjust the assumptions.

Throughout Addendum 1 the NRC staff discusses the assumptions that were made and where applicable the empirical data used to support those assumptions is referenced. With respect to making judgements about the shipment of spent fuel the NRC staff has the benefit of data from over 40 years of experience in shipping SNF in this country as well as overseas.

Comment: High level waste management and transportation should not be a generic issue and Yucca Mountain should not be used for the study as DOE is behind schedule and it is not an approved site for SNF.

Response: Given that the potential environmental impacts of the transportation of SNF resulting from license renewal are similar for all nuclear power plants who seek to renew their operating licenses, and that the NRC staff's analysis contained in Addendum 1 concludes that the impacts are likely to be small, the Commission feels it is appropriate to reclassify the issue as a Category 1 issue. Use of Yucca

Mountain, Nevada for purposes of the staff's analysis, as the destination of the SNF is appropriate as it is the only site presently under study. It must be emphasized that this generic environmental impact statement is required to make use of the best information available and at this time the assumption that Yucca Mountain is the destination is reasonable for purposes of the staff's analysis. If in the future, conditions change, the assumption made for this analysis may need to be reevaluated.

Comment: Need to consider the intermodal option being considered by Congress for Caliente, Nevada.

Response: The shipment of SNF by rail to Caliente and then transferring it to truck for shipment to Yucca Mountain is one of many options under consideration by DOE. Rather than speculate on which transportation option or options will ultimately be selected, the NRC staff has chosen a mode and routes to Yucca Mountain which in its judgement will have the greatest potential environmental impacts in order to do a bounding analysis for the purpose of this rulemaking.

Comment: The analysis needs to address the impacts of above ground nuclear weapons testing being done at the Nevada Test Site.

Response: For the purposes of considering the environmental impacts of license renewal, there does not appear to be a relevant connection between transportation impacts from civilian SNF and defense related weapons testing at the Nevada test site.

Comment: The analysis relies on assumptions that are 25–30 years old and that have a number of problems including omission of important radionuclides (Iodine-129, Chlorine-36 and Cobalt-60), unrealistic RADTRAN assumptions including inadequate consideration of severe accidents, outdated assumptions from NUREG–0170 and WASH–1238 including the failure to consider the degradation of cladding during extended dry storage, and failure to consider the rail-heavy haul truck option.

Response: With regard to the radionuclides, as indicated in Table 2 of Addendum 1, Cobalt-60 is considered. While both Iodine-129 and Chlorine-36 are long lived, neither is a significant contributor to overall dose. Iodine-129 has a very low specific activity and Chlorine-36 is a beta emitter.

The issue of the severity of accidents considered in the NRC staff's analysis was addressed in an earlier response to comment. The assumptions that are used in the NRC staff's analysis have been periodically reviewed and found adequate. The hypothetical accident conditions of 10 CFR 71.73 have been evaluated against actual conditions encountered in highway and railway accidents and were found to be bounding as documented in NUREG/CR–4829, February 1987, "Shipping Container Response to Severe Highway and Railway Accident Conditions." As noted in Table 3 of Addendum 1, the version of RADTRAN used is updated to March 1999.

Section 3 of Addendum 1 does consider the possible effect of cladding degradation on criticality in the context of increased burnup. That analysis would be equally applicable to any cladding degradation that might occur during prolonged dry storage of the SNF.

With regard to what is asserted to be inadequate consideration of the potential radiological impacts of the rail-heavy haul truck option, the NRC staff has analyzed the radiological impacts of the truck mode along various routes through and around Las Vegas and concludes that they are the limiting scenarios. The largest doses in the incident-free conditions are now to the public. If the rail-heavy haul transport scenario was adopted, a substantial portion of the public exposure would be avoided, since in this scenario, the slow moving heavy haul truck transport would not move through a major population center.

Comment: NRC must consider potential Indian Tribe claims of authority to regulate shipments across reservation lands.

Response: This analysis is a generic study that assumes certain routes for the purpose of evaluating environmental impacts. Because the purpose of this study is neither to propose nor approve routes, the NRC does not need to consider tribal claims of authority to regulate shipments in the context of this analysis.

Comment: The beltway is a county road, not part of the Federal highway system; it is not clear it can be used for shipments.

Response: The DOT regulations do not require that SNF shipments only use federal highways. Therefore, the NRC assumed that the beltway is a possible route around Las Vegas.

Comment: The NRC should address the implications of higher enrichment, higher burnup fuel for consequences of radiological sabotage, as NRC has done so far for the increase in burnup from 33,000 MWd/MTU to 40,000 MWd/MTU (see 49 FR 23867, Proposed Revisions to 10 CFR 73, Modification of Protection Requirements for Spent Fuel Shipments, 6/8/84).

Response: The NRC has not quantified the likelihood of the occurrence of sabotage in this analysis because the likelihood of an individual attack cannot be determined with any degree of certainty. Nonetheless, the NRC has considered, for the purposes of this environmental impact statement and rulemaking, the environmental consequences of such an event. In the determination of the consequences of such an event, higher burnup is only one factor. Based on the staff's study of higher burnup fuel (NUREG–1437, Vol.1, Addendum 1, Table 2), the consequences of a sabotage event involving such fuel could be larger than those in the studies referenced by the commentor. However, given that the consequences of the studies referenced by the commentor were small, even modest increases due to the effects of higher burnup fuel would not result in unacceptably large consequences. Because burnup is not the only factor that could affect the consequences of a sabotage event, the staff continues to study this area. Should new and significant information result from the further study, actions addressing such information will be considered.

Nevertheless, the extensive security measures required by NRC regulations make sabotage events extremely unlikely. Moreover, the casks required to be used to transport spent fuel are designed to withstand very substantial impacts during transport without loss of containment integrity. The cask designs should serve to further reduce the likelihood of release of radioactive material in the extremely unlikely event of sabotage. In view of the fact that NRC safeguards regulations make sabotage events extremely unlikely, and the fact that the cask designs themselves should make a release of radioactive material unlikely even were sabotage to occur, and based on our judgement that, in the extremely unlikely event that sabotage and releases did occur, the consequences from higher burnup fuel would not be unacceptably large, we have concluded that a more extensive study of higher burnup fuel consequences is not warranted for this environmental impact statement and rulemaking.

On June 22, 1999, the Nevada Attorney General filed a petition with the Commission which requested the NRC to amend regulations governing safeguards for shipments of spent nuclear fuel against sabotage and terrorism and to initiate a comprehensive assessment. In particular, the petition indicated that

NRC should factor into its regulations the changing nature of threats posed by domestic terrorists, the increased availability of advanced weaponry and the greater vulnerability of larger shipping casks traveling across the country. If, as a result of reviewing this petition, the NRC reaches conclusions that are inconsistent with the results or assumptions in the present rulemaking, the Commission will need to revisit the analysis presented here.

Finding of No Significant Environmental Impact: Availability

The NRC has determined that this final rule is the type of action described as a categorical exclusion in 10 CFR 51.22(c)(3). Therefore, neither an environmental impact statement nor an environmental assessment has been prepared for this regulation. This action is procedural in nature and pertains only to the type of environmental information to be reviewed.

Paperwork Reduction Act Statement

This final rule decreases unnecessary regulatory burden on licensees by eliminating the requirement that license renewal applicants address the generic and cumulative environmental impacts associated with transportation operation in the vicinity of a HLW repository site (−400 hours, −2 responses), and adds a new requirement to address local traffic impacts attributable to continued operation of the plant during the license renewal term (+20 hours, +2 responses). The public burden for these information collections is estimated to average a reduction of 200 hours for each of 2 responses for the elimination of the above mentioned requirement, and an increase of 10 hours for each of 2 responses for the new requirement, for a net burden reduction of 380 hours. Because the burden for this information collection is insignificant, Office of Management and Budget (OMB) clearance is not required. Existing requirements were approved by the OMB, approval number 3150–0021.

Public Protection Notification

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

Regulatory Analysis

The regulatory analysis prepared for the final rule published on June 5, 1996 (61 FR 28467), and amended on December 18, 1996 (61 FR 66537), to make minor clarifying and conforming changes and add language unintentionally omitted from the June 5, 1996 final rule. The rule is unchanged except for an increase in benefits derived from a reduction in the applicant burden of 190 hours of effort in preparing an application for renewal of a nuclear power plant operating license.

This change increases the substantial cost saving of the final rule estimated in NUREG–1440, "Regulatory Analysis for Amendments to Regulations for the Environmental Review for Renewal of Nuclear Power Plant Operating Licences." NUREG–1440 is available for inspection in the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC. In addition, copies of NRC final documents cited here may be purchased from the Superintendent of Documents, U.S. Government Printing Office, PO Box 37082, Washington, DC 20013–7082. Copies are also available for purchase from the National Technical Information Service, 5285 Port Royal Road, Springfield, Virginia 22161.

Regulatory Flexibility Act Certification

As required by the Regulatory Flexibility Act of 1980 (5 U.S.C. 605(b)), the Commission certifies that this final rule will not have a significant impact on a substantial number of small entities. The final rule will reduce the amount of information to be submitted by nuclear power plant licensees to facilitate NRC's obligations under the National Environmental Policy Act. Nuclear power plant licensees do not fall within the definition of small businesses as defined in Section 3 of the Small Business Act (15 U.S.C. 632) or the Commission's Size Standards, April 11, 1995 (60 FR 18344).

Backfit Analysis

The Commission has determined that these amendments do not involve any provisions that would impose backfits as defined in 10 CFR 50.109(a)(1); therefore, a backfit analysis need not be prepared.

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 of OMB.

National Technology Transfer and Advancement Act

The National Technology Transfer and Advancement Act of 1995, Pub. L 104–113, requires that Federal agencies use technical standards developed by or adopted by voluntary consensus standards bodies unless the use of such a standard is inconsistent with applicable law or otherwise impractical. There are no consensus standards that apply to the analysis and findings process, nor to the requirements imposed by this rule. Thus the provisions of the Act do not apply to this rule.

List of Subjects in 10 CFR Part 51

Administrative practice and procedure, Environmental impact statement, Nuclear materials, Nuclear power plants and reactors, Reporting and recordkeeping requirements.

For the reasons set out in the preamble to this notice and under the authority of the Atomic Energy Act of 1954, as amended; the Energy Reorganization Act of 1974, as amended; the National Environmental Policy Act of 1969, as amended; and 5 U.S.C. 552 and 553, the NRC is adopting the following amendments to 10 CFR part 51.

PART 51—ENVIRONMENTAL PROTECTION REGULATIONS FOR DOMESTIC LICENSING AND RELATED REGULATORY FUNCTIONS

1. The authority citation for part 51 continues to read as follows:

Authority: Sec. 161, 68 Stat. 948, as amended, Sec. 1701, 106 Stat. 2951, 2952, 2953 (42 U.S.C. 2201, 2297f); secs. 201, as amended, 202, 88 Stat. 1242, as amended, 1244 (42 U.S.C. 5841, 5842).

Subpart A also issued under National Environmental Policy Act of 1969, secs. 102, 104, 105, 83 Stat. 853–854, as amended (42 U.S.C. 4332, 4334, 4335); and Pub. L. 95–604, Title II, 92 Stat. 3033–3041; and sec.193, Pub. L. 101–575, 104 Stat. 2835, (42 U.S.C. 2243). Sections 51.20, 51.30, 51.60, 51.61, 51.80, and 51.97 also issued under secs. 135, 141, Pub. L. 97–425, 96 Stat. 2232, 2241, and sec. 148, Pub. L. 100–203, 101 Stat. 1330–223 (42 U.S.C. 10155, 10161, 10168). Section 51.22 also issued under sec. 274, 73 Stat. 688, as amended by 92 Stat. 3036–3038 (42 U.S.C. 2021) and under Nuclear Waste Policy Act of 1982, sec. 121, 96 Stat. 2228 (42 U.S.C. 10141). Sections 51.43, 51.67, and 51.109 also issued under Nuclear Waste Policy Act of 1982, sec. 114(f), 96 Stat. 2216, as amended (42 U.S.C. 10134(f)).

2. In § 51.53, paragraph (c)(3)(ii)(M) is removed and reserved and paragraph (c)(3)(ii)(J) is revised to read as follows:

§ 51.53 Post-construction environmental reports.

* * * * *

(c) * * *

(3) * * *

(ii) * * *

(J) All applicants shall assess the impact of highway traffic generated by the proposed project on the level of service of local highways during periods of license renewal refurbishment activities and during the term of the renewed license.

* * * * *

(M) [Reserved].

* * * * *

3. The "Public services, Transportation" issue under the Socioeconomics Section and the "Transportation" issue under the Uranium Fuel Cycle and Waste Management Section of Table B–1, Appendix B to Subpart A to 10 CFR Part 51 are revised to read as follows:

Appendix B to Subpart A—Environmental Effect of Renewing the Operating License of a Nuclear Power Plant

* * * * *

TABLE B–1.—SUMMARY OF FINDINGS ON NEPA ISSUES FOR LICENSE RENEWAL OF NUCLEAR POWER PLANTS [1]

Issue	Category	Findings
* * *	* *	* *
Socioeconomics		
* * *	* *	* *
Public services, Transportation	2	SMALL, MODERATE, OR LARGE. Transportation impacts (level of service) of highway traffic generated during plant refurbishment and during the term of the renewed license are generally expected to be of small significance. However, the increase in traffic associated with additional workers and the local road and traffic control conditions may lead to impacts of moderate or large significance at some sites. See § 51.53(c)(3)(ii)(J).
* * *	* *	* *
Uranium Fuel Cycle and Waste Management		
* * *	* *	* *
Transportation	1	SMALL. The impacts of transporting spent fuel enriched up to 5 percent uranium-235 with average burnup for the peak rod to current levels approved by NRC up to 62,000 MWd/MTU and the cumulative impacts of transporting high-level waste to a single repository, such as Yucca Mountain, Nevada are found to be consistent with the impact values contained in 10 CFR 51.52(c), Summary Table S–4—Environmental Impact of Transportation of Fuel and Waste to and from One Light-Water-Cooled Nuclear Power Reactor. If fuel enrichment or burnup conditions are not met, the applicant must submit an assessment of the implications for the environmental impact values reported in § 51.52.
* * *	* *	* *

[1] Data supporting this table are contained in NUREG–1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants" (May 1996) and NUREG–1437, Vol. 1, Addendum 1, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Main Report Section 6.3—'Transportation,' Table 9.1 'Summary of findings on NEPA issues for license renewal of nuclear power plants,' Final Report" (August 1999).

Dated at Rockville, Maryland, this 26th day of August, 1999.

For the Nuclear Regulatory Commission.

Annette Vietti-Cook,

Secretary of the Commission.

[FR Doc. 99–22764 Filed 9–2–99; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

10 CFR Part 51

RIN 3150–AG05

Changes to Requirements for Environmental Review for Renewal of Nuclear Power Plant Operating Licenses To Include Consideration of Certain Transportation Impacts, Availability of Supplemental Environmental Impact Statement

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule; Notice of availability of supplemental document.

SUMMARY: The Nuclear Regulatory Commission (NRC) is announcing the completion and availability of NUREG–1437, Vol. 1, Addendum 1, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Main Report Section 6.3—'Transportation,' Table 9.1 'Summary of findings on NEPA issues for license renewal of nuclear power plants,' Final Report" (August 1999).

ADDRESSES: Copies of NUREG–1437, Vol. 1, Addendum 1 may be obtained by writing to the Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402–9328. Copies are also available from the National Technical Information Service, 5285 Port Royal Road, Springfield, Virginia 22161. A copy of the document is also available for inspection and/or copying for a fee in the NRC Public Document Room, 2120

L Street, NW (Lower Level), Washington, DC.

FOR FURTHER INFORMATION CONTACT: Donald P. Cleary, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, telephone: 301–415–3903; e-mail: dpc@nrc.gov.

SUPPLEMENTARY INFORMATION: The report provides the technical basis for the final rule "Changes to Requirements for Environmental Review for Renewal of Nuclear Power Plant Operating Licenses" that amends requirements to the Commission's rule in 10 CFR Part 51—Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions.

The NRC staff has completed the analyses of transportation issues as reported in NUREG–1437, Vol. 1, Addendum 1, which provides the bases for designating the transportation of high level waste as a Category 1 issue. Addendum 1 would supplement the analysis and amend the findings and the Category 2 designation for the issue of Transportation in Section 6.3 and Table 9.1 of NUREG–1437. This report expands the generic findings about the environmental impacts due to transportation of fuel and waste to and from a single nuclear power plant. Specifically, the report adds to findings concerning the cumulative environmental impacts of convergence of spent fuel shipments on a single destination, rather than multiple destinations, and the environmental impact of transportation of higher enriched and higher burnup spent fuel during the renewal term. The report conclusions would permit those findings to be used by incorporation by reference in the environmental review of an application for renewal of an individual nuclear plant operating license. The results are being codified in 10 CFR Part 51.

Electronic Access

NUREG–1437, Vol. 1, Addendum 1, is also available electronically by visiting NRC's Home Page (*http://www.nrc.gov*) and choosing "Nuclear Materials," then "Business Process Redesign Project," then "Library," and then "NUREG–1437, Volume 1, Addendum 1."

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 of OMB.

Dated at Rockville, Maryland, this 26th day of August, 1999.

For the Nuclear Regulatory Commission.

Annette Vietti-Cook,

Secretary of the Commission.

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