5. In § 46.13, paragraphs (a)(2) and (a)(5) are revised to read as follows:

§ 46.13 Address, ownership, changes in trade name, changes in number of branches, changes in members of partnership, and bankruptcy.

* * * * *

(a) * * *

(2) Any change in officers, directors, members, managers, holders of more than 10 percent of the outstanding stock in a corporation, with the percentage of stock held by such person, and holders of more than 10 percent of the ownership stake in a limited liability company, and the percentage of ownership in the company held by each such person;

* * * * *

(5) When the licensee, or if the licensee is a partnership, any partner is subject to proceedings under the bankruptcy laws. A new license is required in case of a change in the ownership of a firm, the addition or withdrawal of partners in a partnership, or in case business is conducted under a different corporate charter, or in case a limited liability company conducts business under different articles or organization from those under which the license was originally issued.

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Dated: April 21, 2000.

Robert C. Keeney,

Deputy Administrator, Fruit and Vegetable Programs.

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NUCLEAR REGULATORY COMMISSION

10 CFR Part 72

RIN 3150–AG 30

List of Approved Spent Fuel Storage Casks: TN–68 Addition

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is amending its regulations to add the Transnuclear TN–68 cask system to the list of approved spent fuel storage casks. This amendment allows holders of power reactor operating licenses to store spent fuel in the Transnuclear TN–68 cask system under a general license.

DATES: The final rule is effective May 30, 2000.

FOR FURTHER INFORMATION CONTACT: Gordon Gundersen, telephone (301) 415–6195, e-mail, GEG1@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 for the dry storage of spent nuclear fuel at civilian nuclear power reactor sites, with the objective of establishing one or more technologies 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, that ''[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 Transnuclear TN–68 cask system to the list of NRC approved casks for spent fuel storage in 10 CFR 72.214. Following the procedures specified in 10 CFR 72.230 of Subpart L, Transnuclear submitted an application for NRC approval with the Safety Analysis Report (SAR) entitled ''Final Safety Analysis Report for the TN–68 Dry Storage Cask,'' dated January 23, 1998. The NRC evaluated the Transnuclear submittal and issued a preliminary Safety Evaluation Report (SER) and proposed Certificate of Compliance (CoC) for the Transnuclear TN–68 cask system. The NRC published a proposed rule in the **Federal Register** (64 FR 45920; August 23, 1999) to add TN–68 cask system to the listing in 10 CFR 72.214. The comment period ended on November 8, 1999. Three comment letters were received on the proposed rule.

Based on NRC review and analysis of public comments, the NRC staff has modified, as appropriate, its proposed CoC, including its appendices, the Technical Specifications (TSs), and the Approved Contents and Design Features for the Transnuclear TN–68 cask system. The NRC staff has also modified its preliminary SER.

The NRC finds that the Transnuclear TN–68 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 Transnuclear TN–68 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 Transnuclear TN–68 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 May 30, 2000. 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 three comment letters on the proposed rule. The commenters included an industry representative, an individual member of the public, and a utility. Copies of the public documents are available for review in the NRC Public Document Room, 2120 L Street, NW (Lower Level), Washington DC.

Comments on the Transnuclear TN–68 Cask System

The comments and responses have been grouped into eight subject areas: General, materials, crud, miscellaneous issues, technical specifications, comments on applicant's SAR, accidents, and radiation protection. To the extent possible, all of the comments on a particular subject are grouped together. A review of the comments and the NRC staff's responses follow:

A. General Comments

Comment A–1: One commenter requested that the general comments submitted by the commenter on the TN–32 rule apply to this rule as well.

Response: Comments that were general enough to apply to both the TN–32 and the TN–68 casks, were addressed in the response to the comments on the TN–32 rule (65 FR 14790, March 20, 2000). Specific comments are addressed in this rulemaking for the TN–68 cask.

Comment A–2: One commenter stated that the environmental assessment (EA) is ''tiered'' on documents having little to do with the dry casks of today and that an Environmental Impact Statement (EIS) for each generic design should be done.

Response: The NRC disagrees with the comment. The EA and Finding of No Significant Impact (FONSI) for this rule are limited in scope to the TN–68 in a generic setting. The NRC has given specific consideration to environmental impacts of dry storage and has not found any new information affecting the conclusion that these impacts are expected to be extremely small and not environmentally significant. Therefore, the NRC is not convinced that meaningful new environmental insights would be gained by performing an environmental impact analysis for each new cask that is certified. The EA covering the proposed rule, as well as the FONSI prepared and published for this final rule, fully comply with NRC's environmental regulations in 10 CFR part 51. The Commission's environmental regulations in part 51 implement the National Environmental Policy Act (NEPA) and give proper consideration to the guidelines of the Council of Environmental Quality (CEQ). The EA and FONSI prepared for the TN–68, as required by 10 CFR part 51, conform to NEPA procedural requirements. Tiering on past EISs and EAs is a standard process under NEPA. As stated in CEQ's 40 Frequently Asked Questions, the tiering process makes each EIS/EA of greater use and meaning to the public as the plan or program develops, without duplication of the analysis prepared for the previous impact statement.

Comment A–3: One commenter stated that decommissioning, transport, and disposal of fuel from these casks have not been adequately analyzed.

Response: The CoC for the TN–68 is for the storage of spent fuel. Decommissioning, transport, and disposal of fuel from the casks is beyond the scope of this rule.

Comment A–4: One commenter stated that the environmental impacts would not be the same for a general license and a site-specific license.

Response: The NRC disagrees with the comment. Each cask is designed and fabricated to specific design criteria whether it is licensed for site-specific or general use. The process for determining the environmental impact varies, but the cask must satisfy the same technical requirements. There are no significant environmental impacts using a spent fuel dry storage cask under either a site-specific or a general license.

Comment A–5: One commenter stated that previous fabricators of casks have not realized that the casks are made to store nuclear spent fuel and the quality of their work can affect the health and safety of the public. The commenter asked why the NRC is ''opening up'' the approval process to lower standards by fabricators, material suppliers, and inspectors.

Response: The NRC disagrees with this comment. All licensees/CoC holders must have a quality assurance (QA) program that has been approved by the NRC as part of the licensing or CoC issue process. This QA program must meet the requirements of 10 CFR 72.148 and 72.154 in regards to the selection of fabricators. The licensee/CoC holder is required to assure that all regulations and certificate conditions applicable to the cask are met. In addition, the licensee/CoC holders and fabricators are subject to NRC inspections to verify compliance.

Comment A–6: One commenter stated that the design should be built and tested before certification and that NRC approving a design without a test is wrong, and asked if the NRC is going to allow the first cask to be tested by a utility.

Response: The NRC disagrees with the comment. The TN–68 cask 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 only 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. As detailed in the SER, the NRC staff has reviewed the analyses performed by TN and found them acceptable.

Comment A–7: One commenter noted a lack of confidence that the vendor knows what it is doing when it is permitted by the NRC to make a best effort in the realm of testing and verification of weld quality.

Response: In fabrication, the specific nondestructive examination desired or otherwise required for a particular weld sometimes cannot be performed due to joint geometry or part configuration. As used here, the term ''best effort'' means the joint will be examined using other acceptable methods suitable for the application under the American Society of Mechanical Engineers (ASME) code. Specifically, on the weld of the bottom inner plate to the confinement shell where the weld cannot be examined by ultrasonic testing (UT), the weld will be examined by radiographic testing (RT) and either penetrant testing (PT) or magnetic particle testing (MT) under ASME Subsection NB requirements.

Comment A–8: One commenter stated that everything in the cask should be identified on the cask label in case documents are lost or destroyed.

Response: The NRC disagrees with this comment. NRC regulations do not require the identification of cask contents on permanent markings affixed to the cask. The need for labeling was evaluated during the rulemaking that established Subpart L in 10 CFR part 72 entitled ''Approval of Spent Fuel Storage Casks'' (55 FR 29193; July 18, 1990). The NRC notes that § 72.212(b)(8) requires that each general licensee accurately maintain a record for each cask that lists the spent fuel stored in the cask. The record must be maintained by the cask user until decommissioning of the cask is complete. Also, § 72.72 requires that records of spent fuel in storage must be kept in duplicate, with the duplicate set sufficiently remote from the original records that a single event would not destroy both sets of records.

Comment A–9: One commenter asked if the ''less than 1 gram-mole/cask'' recommendation listed on Page 8–2 of the SER came from PNL–6365, ''Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel,'' R.W. Knoll and E.R. Gilbert, Pacific Northwest Laboratory, Richland, Washington, November 1987; what kind of dry storage PNL evaluated; and what dry storage casks were in use before 1987? The commenter then added a recommendation that the reference be updated.

Response: The less than 1 gram-mole/cask limit is from the cited reference. The investigators evaluated four cask designs loaded with spent fuel, the MC–10, TN–24P, Castor-V/21, and MSF IV. Further details are contained in the report. Dry storage casks in use before 1987 were the Castor V/21, the MC–10, and the NUHOMS–7P. The NRC considers this reference material to be acceptable and that it does not need to be updated.

Comments A–10: One commenter recommended that detailed site-specific unloading procedures should not be developed by licensees. Instead, the NRC should fully inspect the procedures and place them in the PDR before any cask loading is done at the plant. The commenter also suggested that contamination control measures should be carefully thought out to adequately address the presence of fuel crud, and suggested that the generic review should pay more attention to a detailed plan for emergency cask unloading including how contamination is controlled, especially crud, and how effluents are released.

Response: The NRC disagrees with this comment. The TN–68 Storage Cask System Design operating descriptions and analysis have been reviewed and accepted by the NRC. The NRC staff concluded in the SER that there was reasonable assurance that the cask unloading operations could be safely performed by qualified personnel using detailed procedures developed by the cask user at an ISFSI site. Cask general licensees must be licensed under 10 CFR 50. These licensees have sufficient infrastructure, experience, and processes in place to develop adequate detailed unloading procedures without prior NRC review. Detailed site-specific procedures for performing unloading operations, including contamination/ effluent control measures, are required to be developed and demonstrated at each facility that uses the TN–68.

Comment A–11: One commenter stated that the use of a proprietary neutron shielding material is not in the interest of the public health and safety, and that the best neutron shielding material should be identified and available for use by all vendors and licensees.

Response: This comment is beyond the scope of this rule. The applicant's proposed materials have been found by the NRC staff to be acceptable. The critical attributes of the material are not proprietary and are specified in the CoC and SER.

Comment A–12: One commenter stated that the public would be better served if one design would be approved for casks rather than the large number that is being approved based on utilities choosing the least expensive designs.

Response: This comment is beyond the scope of this rule. NWPA gives NRC authority to approve multiple cask designs.

Comment A–13: One commenter asked where the decontaminated TN–68 components would be stored and where the remaining low-level waste would be disposed.

Response: This comment is beyond the scope of this rule. Disposal of low-level waste is covered by 10 CFR parts 20 and 61.

Comment A–14: One commenter stated that NRC is approving generic designs which allow site specific changes by utilities that use the casks and that this makes it difficult to establish a standardized, integrated total waste system for the United States. The commenter further stated that approval of generic designs is creating vendor competition to rapidly develop cheap designs with current materials instead of competition to create the best and safest designs. The commenter asked how many designs does the NRC plan to allow in the industry and how will approving a large number affect shipping and final disposal of spent nuclear fuel.

Response: This comment is beyond the scope of this rule. NWPA gives NRC authority to approve multiple cask designs.

Comment A–15: One commenter stated that NRC documents are long, repetitive, and hard to understand. The commenter also stated that the more people who go over these documents and ask questions, the better.

Response: The NRC agrees that documents should be easy to understand. Because the documentation necessary to license a storage cask is tiered and must be comprehensive to document the NRC staff's evaluation and findings, the documentation may be extensive. The NRC documents are available for public comment.

Comment A–16: One commenter disagreed that sabotage scenarios have been fully evaluated, and stated that sabotage evaluation for site-specific parameters should be updated.

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 rule that added Subparts K and L to 10 CFR part 72 (55FR 29181; July 18, 1990). The NRC still finds the results of the 1990 rule current and acceptable. 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 Part 72 licensee is required by § 73.51 or § 73.55 to develop a physical protection plan for the ISFSI and to install a physical protection system that provides high assurance against unauthorized activities that could constitute an unreasonable risk to the public health and safety. Each ISFSI is periodically inspected by NRC, and the licensee conducts periodic patrols and surveillances to ensure that physical protection systems are operating within their design limits.

B. Materials

Comment B–1: One commenter asked what is a torispherical weather cover with elastomeric seals and why all dry cask designs should or should not have them.

Response: The torispherical weather cover is a protective cover that provides weather protection for the closure lid, top neutron shield, and overpressure system. The use of such a cover on other cask designs is beyond the scope of this rule.

Comment B–2: One commenter asked why TN is allowed to use alternative neutron shield materials as discussed in the CoC. The commenter also asked why the current materials of borated wrought aluminum alloy or BorALYN™ have not been approved with no alternative and why the best material is not chosen for the design at this point. The commenter stated a concern about the number and complexity of criteria for BorALYN™ fabrication in that it results in a complicated fabrication process. The commenter recommended that more research be conducted to find a better neutron shield material with less problems. The commenter stated that TN appears not to be satisfied with the current neutron shield materials because they ''envision an alternative candidate'' for which they need to develop appropriate qualification test data, and asked why TN and NRC are not waiting until the improved neutron shielding material is available before certification of the CoC.

Response: The applicant's proposed materials have been found acceptable by the NRC staff. After careful review of this material and its properties under various conditions, the staff is not aware of any problems with this material in its intended service.

Comment B–3: One commenter asked if the casks can be moved with the temperature above freezing.

Response: TS 3.1.6 requires that the loaded cask not be lifted if the outer surface of the cask is below −20°F. There is no other temperature restriction for moving the TN–68 cask.

Comment B–4: One commenter asked if the cask meets ASME code standards, asked if the applicant has adequately justified an exemption from the code requirements, and if the NRC staff has verified this action.

Response: The cask is designed, fabricated, and inspected under the appropriate subsections of the ASME Code. Exceptions to the ASME Code are listed in Table 4.1–1 of the TSs and Section 7 of the SAR. These exceptions and associated justifications and compensatory measures were reviewed by the NRC staff and found to have no adverse effects on the cask integrity. The basis for cask approval is documented in the SER.

Comment B–5: One commenter stated that the CoC specifications about fabricator verification of the quality of the welding of the inner plate to the confinement shell are somewhat vague and do not specify firm requirements. Examples cited were statements in the CoC that ultrasonic testing (UT) of the weld will be performed on a best effort basis, the joint examination can be performed by a number of methods, the joint *may* be welded after shrink fitting of the shells, and that the geometry may not allow for UT examination. The commenter also asked if there had been problems with the shield weld in previous designs.

Response: The NRC disagrees that the specifications are vague. ASME Code, Section III, Division 1, Subsection NB–5231(b) requires either ultrasonic or radiographic examinations and either liquid penetrant or magnetic particle examinations be performed on the full penetration corner welded joints. Therefore, the applicant can choose either ultrasonic or radiographic examinations to inspect the corner weld. The bottom inner plate weld is inspected using ultrasonic examination methods if the weld is applied before the outer and inner shells are assembled. If the weld is applied after assembly, this inspection is done radiographically. Both methods will be supplemented by either liquid penetrant or magnetic particle examinations. The NRC staff is not aware of any problems with the shield weld designs.

Comment B–6: One commenter asked what the shrink fit process is and if it has been used and time tested before, questioned using shrink fit and frictional forces to keep the shells from separating, and asked if the shrink fit will be performed before the welding of the bottom confinement shell.

Response: The shrink fit is established as follows: The gamma shield shell and the confinement shell are fabricated separately. To obtain a close fit between these two shells, the outside diameter of the confinement shell is slightly larger than the inside diameter of the gamma shield shell. The gamma shield shell is preheated which causes it to expand before slipping on the confinement shell. As the gamma shield shell cools, it shrinks and tightly clamps onto the confinement shell. Shrink fit is a common industrial practice that has been used to fabricate various nuclear components including those used successfully in other NRC-approved casks. Fire, tipover, or seismic events would not cause the two shells to separate as demonstrated in Sections 3, 4, and 11 of the SAR. The SAR specifies welding either before or after shell assembly. As long as the confinement barrier is welded to meet ASME Code Section III, Subsection NB requirements, test standards, and acceptance standards, the barrier will conform with a standard that will satisfy all of the safety requirements for this application.

Comment B–7: One commenter stated that 30 days in the pool for a cask is a long time, and asked what happens to neutron absorber material, aluminum paint, etc., during this extended period of time. The commenter stated that assemblies left in a cask cavity in the pool are very different from just being in the pool out of the cask, and asked how fast the hot water is going to be exchanged with cooler pool water when fuel is left in the cask with the cover removed. The commenter also asked if the water in the pool is constantly cooled, how cask walls will affect that bit of pool water in the cask with the 68 assemblies compared to the rest of the pool, how cask materials will affect pool water and pool filters if left in the pool for 30 days, if crud will come off the assemblies that were dried and put back in the pool, if iron oxide will come off the paint, and what chemicals in the pool could be affected by the cask being in the pool for seven versus 30 days.

Response: The effect of the water on these materials is negligible. The reactions with pool water occur very slowly and give rise to only small amounts of hydrogen, ions, and/or precipitates in the pool water that are trapped by filters designed to capture small items from the water. This is true for aluminum, aluminum ''paint,'' and the stainless and coated ferrous materials used in this system. The aluminum is not a paint but it is aluminum and aluminum oxide that when applied as a liquid spray of aluminum to the cask surfaces, becomes tightly adherent to the substrate onto which it is applied. Some of the aluminum becomes an oxide of aluminum during this process. Neither the aluminum oxide nor the iron oxide is expected to come off the paint when exposed to pool water. The system is designed to allow the free movement of pool water into the cask with the lid removed, and systems are in place to constantly cool the pool water. The water in a BWR pool is typically pure water which has no chemical addition unless that chemical is evaluated on a site-specific basis. The questions specific to operation of the spent fuel pool are beyond the scope of this rule.

Comment B–8: One commenter stated that 48 hours without helium seems to be the maximum time for the basket and that even if fuel temperature limits are not reached, there could be basket damage. The commenter stated this should be made clearer in the CoC.

Response: The NRC agrees that there is a potential for exceeding the basket temperature limits after 48 hours. To protect the basket, the TSs require that the licensee initiate and complete a helium backfill procedure at the 42 and 48 hour marks, respectively. This is stated in TS bases B3.1.1 and B3.1.2, SER Section 4.5.2.4, and SAR Section 4.6.2.

Comment B–9: One commenter questioned a CoC statement that flaws in the gamma shield are not examined no matter what is typically observed in the material. The commenter suggested that a large crack could let water in and cause rusting of materials.

Response: The NRC disagrees with the comment. The gamma shield is a forged component. Flaws in forgings are very small. There is no safety related risk or materials problem related to the use of a forging in this application. The allowable flaws for various orientations and locations are stated in Appendix 3E of the SAR. Flaws of these sizes will not propagate under service conditions.

Comment B–10: One commenter asked why there are lower trunnions for rotating the cask from horizontal to vertical.

Response: The unloaded cask may be shipped from the manufacturer to the site in a horizontal orientation. The lower trunnions provide capability to rotate the cask to the vertical orientation before loading of spent fuel. The upper trunnions are the only components used for lifting the loaded cask.

Comment B–11: One commenter had a number of concerns related to the neutron source and neutron shielding. The commenter stated that enrichment, burn up, and fuel cooling time seem to be crucial to avoid having a neutron source too high. The commenter also stated that the neutron shield material choice and structure is flimsy and a better choice of material is needed, and that because in the SER the NRC stated ''all of the fire accident temperatures were below short-term design-basis temperatures with the exception of the neutron shield material,'' the design should use another material. The commenter asked what would be the expected result of a long term fire for the neutron material, why the design includes a neutron shield material that can off-gas during a fire, what gas would be given off by the combustion of the neutron shield, how the gas would react, if the gas is explosive, or if it would react with anything from a plane crash or truck bomb to make the problem worse. The commenter stated that the fire accident should be evaluated to consider the effects of neutron shield resin burnup. The commenter also stated that the KX–277 material in the VSC–24 design and the proposed resin shielding in TN casks can contain voids, is not strong, and is flammable, while alloys being discussed for Yucca Mountain seem much better and more expensive. The commenter further stated that having a multipurpose cask with better shielding would be better in the long run instead of vendors using the cheapest materials.

Response: The NRC concurs with the comment on the parameters important to a neutron source term. These parameters are controlled in Section 2 of the TSs. The NRC staff disagrees that a different neutron shield material is needed. The proposed material was evaluated and found to satisfy the safety requirements for the application. The top neutron shield and the radial neutron shield have not been designed to withstand all of the hypothetical accident conditions. Cask structural analyses have been performed assuming that the neutron shield is completely removed during accident conditions. The results indicate that the cask without the neutron shield is adequately designed to withstand various load combinations of the accident condition as presented in Sections 2, 3, 4, and 11 of the SAR. The cask has been analyzed for the post-fire condition and has been found to meet the dose requirements of 10 CFR 72.106 even without the neutron shielding being present. The question on a long-term fire is beyond the testing/analysis required by Part 72. The radial neutron shield is a polymeric material that includes about 50 weight percent fire-retardant mineral fill, which makes it self-extinguishing. The polymeric neutron shield materials may char or off-gas if directly exposed to fire or high temperatures. The applicant has modified the SAR to address the combustibility of the neutron shield. The off-gas products are formed from a very small fraction of the total neutron shield mass and are not explosive but may burn during the fire. The heat input from this reaction would be insignificant relative to that of the design basis fire. Comments on the VSC–24 material, Yucca Mountain, and multipurpose casks are beyond the scope of this rule.

Comment B–12: One commenter asked about information included on Section 4.5.2.4 of the SER. Specifically, the commenter asked if partial pressure injection of helium had ever been performed for a similar cask, where, and what were the results. The commenter also asked if the air-helium mixture will really work. Further, the commenter stated that the NRC referred to a ''different cask system'' and asked what data is applicable to the different cask system and if it can apply to the TN–68 design.

Response: The purpose of the helium injection is to improve the thermal conductivity of the fill gas as a temporary measure to provide an opportunity to troubleshoot and repair any problems during the drying or helium fill process. ISG 7, ''Potential Generic Issue Concerning Heat Transfer in a Transportation Accident'' dated October 2, 1998, provides NRC staff guidance for mixtures of gases within a spent fuel storage cask. In support of ISG–7, a sensitivity study was performed to evaluate the relative change in cladding temperatures as a result of significant reductions in the thermal conductivity of the fill gas (*e.g.,* 30% that of helium). This evaluation found that the cladding temperature was relatively insensitive to gas thermal conductivity as evidenced by an increase in the fuel cladding and bulk gas temperatures of about 3%. The NRC staff did not review or require any testing of the helium injection process based on the analysis performed for ISG–7 and the restrictions, imposed by the TN–68 TSs, on operations without a full helium environment to maintain the desired protection for the cladding.

Comment B–13: One commenter stated that the SER states the NRC staff projected a peak cladding temperature lower than the long term storage cladding temperature limit if the fabrication results in gaps of 0.05 in. or less between component layers. The commenter asked if the NRC would accept up to a 0.05 inch gap and why the applicant's assumed gap of 0.01 in. should not be the fixed limit.

Response: Gaps between the various cask components were assumed in the analysis to account for fabrication and assembly tolerances and uncertainties. The implemented QA program at the fabricator's facility provides reasonable assurance that the as-built casks will have gaps that are less than or equal to those assumed in the analysis. In the context of the statements referenced by the commenter, the NRC performed a sensitivity analysis to evaluate the response of the cask thermal performance to increased gap sizes. The results of that evaluation found that gaps could be five times that assumed in the analysis and the fuel cladding would remain within temperature limits.

Comment B–14: One commenter expressed concern over the continued efficacy of the neutron absorber plates over 20 years of storage. In addition, the commenter stated that the NRC needs to look more carefully at issues such as unexpected erosion or corrosion, potential explosions, and cracks in welds for the life of the cask. The commenter also stated dislike of materials used in this design including poured resin, borated aluminum, and metal matrix.

Response: The neutron absorber is designed to remain effective in the TN–68 system for a storage period greater than 20 years. Section 6.3.2 of the TN–68 SAR describes the neutron absorber and its environment, and evaluates boron depletion due to neutron absorption. Section 9.1.7 of the SAR describes the testing procedures for the neutron absorber material, which will be manufactured and tested under the control and surveillance of a quality assurance and quality control program that conforms to the requirements of 10 CFR part 72, subpart G. The compositions and densities for the materials in the computer models were reviewed by the NRC staff and determined to be acceptable. The NRC staff notes that these materials are not unique and are commonly used in other spent fuel storage and transportation applications.

The NRC staff disagrees that the stated issues need to be looked at more carefully. The NRC is already looking carefully at the materials that may impact the safe performance of storage systems. As part of this effort, the NRC has participated, over the past several years, in the work of a Task Group of Subcommittee C26.13 of the American Society of Testing and Materials on life extension questions. This Task Group has been developing guidance for components of storage cask systems for periods up to a 100-year service life. This work is taken into account in the reviews that are ongoing for storage systems. Erosion and corrosion are not expected to occur at any level significant enough to affect safe performance of components of the cask. The TN–68 is designed to withstand an external pressure of 25 psi. This would include a nearby explosion, debris falling on the cask, etc. If a credible explosion is identified that would apply more than 25 psi to the outer surface of the cask at a site, the site will have to address this issue in its 10 CFR 72.212 evaluation. Any cracks in welds or other flaws in components are small in relation to what is needed to extend these cracks in service. Fracture mechanics calculations can be used to show them to be stable (will not propagate) for the levels of stress to be sustained in service.

Regarding the commenter's dislike for particular materials, material selection is the applicant's responsibility. The applicant must demonstrate that the materials and the materials' properties satisfy the requirements for a given application.

Comment B–15: One commenter recommended that the installation of a blind flange on the overpressure monitoring system (OMS) to mitigate a latent seal failure event should be tested to verify that it will work.

Response: The NRC disagrees with this comment. The possibility of the occurrence of the events needed to occur concurrently for a latent seal failure event is judged to be very remote. If this unlikely event were to occur, the mitigative action to install a blind flange at the OMS port is straightforward and well within the capability of a nuclear power plant licensee. Therefore, the NRC has reasonable assurance that the action can be taken without additional testing.

Comment B–16: One commenter asked the NRC to explain ''bubble leak tests'' in relation to resin enclosures and leak passages on weld enclosures. The commenter also asked how test failures are rectified and rechecked.

Response: This test is described in ANSI 14.5–97, ''American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment'' February 1998. Deficiencies are evaluated, repaired, and retested under the cask vendor's QA program, as described in SAR Section 13.

Comment B–17: One commenter stated that the following editorial corrections should be made in the TS: On the bottom of page 1.2–1, ''continued'' should be moved above the line; on page 1.3–5, ''Time the'' should be moved from the first column to the second column of information; on the bottom of page 3.0–1, ''continued'' should be added below the line; at the top of page 3.0–2, ''3.0 LCO APPLICABILITY (continued)'' should be added; at the bottom of the page 3.0–2, ''continued'' should be moved above the line; at the top of page 3.0–4, the ''continued'' above the line should be deleted and the ''continued'' below the line should begin with a lower case letter; and on page 3.1.1–1, the double line separating conditions B and C should be changed to a single line.

Response: The NRC agrees with these changes. The TSs have been reformatted accordingly.

Comment B–18: One commenter stated that on drawing 972–70–2 of the SAR, the materials for the protective cover should be changed to SA–516 GR. 70 or SA–105 to allow the cover flange to be made from a forging.

Response: The NRC accepts this change to the protective cover materials because the material properties are the same. This change will not affect the structural analyses and the conclusions reached in the SER. Drawing No. 972–70–2 has been changed accordingly.

Comment B–19: One commenter stated that on drawing 972–70–3 of the SAR, a note should be added to allow the protective cover flange to be made from a one-piece forging.

Response: The NRC accepts this change because it will not affect the structural analyses and the conclusions reached in the SER. Drawing No. 972–70–3 has been changed accordingly.

Comment B–20: One commenter stated that the material of the metallic seals described in Chapters 2 and 7 should be changed to allow a stainless steel or nickel alloy liner.

Response: The NRC agrees with this comment. The use of either stainless steel or nickel alloy is acceptable to the NRC staff. The SAR has been changed to reflect this change.

Comment B–21: One commenter stated that on page 3–5 of the SER, the basis for the allowable stress for the 6061–T6 alloy is in error.

Response: The NRC disagrees with this comment. The basis for the allowable stress for the 6061–T6 alloy is Section III of the ASME Code, as stated in Section 3.1.4 on page 3–5 of the SER.

C. Crud

Comment C–1: One commenter asked what would be done if cask vent flow of saturated steam could not be discharged into the spent fuel pool during reflooding of the cask before unloading. The commenter also asked what conditions could preclude discharge to the spent fuel pool, specifically asking about too much radioactivity, failed fuel, crud, fuel fines, and iron oxide debris.

Response: As shown in SAR Figure 8.2.1, the cask may be vented to the spent fuel pool or to the radwaste system. The reasons suggested by the commenter that may impact the cask vent location are interpreted to be primarily radiological concerns. The procedure descriptions for cask unloading include appropriate reference to development of site-specific procedures and actions that will maintain exposures to workers and radiological releases to the environment as low as reasonably achievable (ALARA). The details of where the cask will be vented are a site-specific matter and beyond the scope of this rule.

Comment C–2: One commenter has a number of concerns about crud on boiling water reactor (BWR) fuel: What material composes the crud and should it be allowed in a cask; how crud is analyzed in all aspects of cask loading, transfer, storage, and unloading, and when fuel is put back in the pool and then loaded in a transport cask or placed in different reactor pools; what happens to the dried crud when it is put back into the pool, and how it affects pool water quality; whether crud covers defects in cladding that may be revealed when it dries and falls off; and if BWR crud is different than pressurized water reactor (PWR) crud.

Response: Crud generally consists of oxides of metals (*e.g.,* Co, Mn, Cr, Fe, Zr, Zn) that are not chemically reactive in the storage cask environment. The crud collects on the exterior of the fuel cladding during reactor operation. The crud particles for BWR fuel are very small with diameters ranging from 0.1 to 10 micrometers as reported in SAND88–1358, ''Estimate of Crud Contribution to Shipping Cask Containment Requirements'' January 1991. SAND88–1358 found that the crud on BWR fuel was less adherent than that found on PWR fuel. Some crud may be dislodged or spall from the fuel cladding during spent fuel dry storage or handling; however, there were no differences reported in the spallation behavior of crud between the two fuel types.

The safety concern associated with crud is its radiological impact. The analysis provided by the applicant uses a bounding assumption for crud activity of 1254 $\mu Ci/cm^2$ of Cobalt-60 (this was the maximum activity level found by actual inspection of BWR fuel) distributed over the entire fuel cladding surface. The analysis demonstrates with reasonable assurance that fuel loading, storage, and unloading can be performed safely. The NRC agrees with the commenter that some crud may be flushed from the cask to the spent fuel pool as a part of the unloading process. The operating procedure descriptions address this possibility and the precautions for handling this situation.

Regarding the impact of crud in the spent fuel pool, there is crud from wet fuel storage already present in a spent fuel pool and the amount of crud from the spent fuel cask is expected to be very small. If any crud is discharged to the spent fuel pool, it would be captured in the spent fuel pool filtration system.

Regarding the concern with crud covering defects in cladding and later being revealed when the crud dries and falls away from the defect, the effects of the dislodged crud were addressed earlier in this comment response. The comment also raises the possibility that a cladding defect may be covered by crud, thus allowing the defect to go undetected during visual inspection of the fuel before loading. Cask users must ensure that the fuel loaded into the cask meets the requirements of TS 2.1.1. This

TS precludes loading fuel that has known cladding defects greater than pinhole leaks or hairline cracks. Cask users may use a variety of screening methods to ensure that the fuel meets the TS requirements. These screening methods include review of operational records, visual inspections, fuel assembly sipping, and ultrasonic examination. Because multiple screening methods are used, the NRC has reasonable assurance that the fuel can be adequately screened for compliance with the TS requirements. Further, if a postulated assembly with a cladding defect not meeting the TS requirements was loaded, the NRC does not expect a significant adverse impact in the radiological consequences because the confinement system remains intact during normal, off-normal, and accident conditions.

The impacts of crud on transportation activities are beyond the scope of this rule.

Comment C–3: One commenter stated in reference to page 9–4 of the SER that during unloading a problem could arise due to precipitates, or second-phase particles, even if titanium decreases their size, and noted that any particle or precipitate in unloading, along with crud, etc., is going to be a big concern.

Response: The NRC interprets the comment as a concern for potential loose particles in the cask cavity and disagrees that the particles and precipitates, discussed on page 9–4 of the SER, are a cause for concern in unloading. The discussion on page 9–4 of the SER refers to boride precipitates that are components of the metal matrix in the borated aluminum plate and will not separate from the plate material during unloading. In response to the commenter's question about other particulates, including crud, Comment C–2 responds to that concern.

D. Miscellaneous Items

Comment D–1: One commenter stated that reference 4 on Page 5–7 of the SER should be revised or updated. Specifically, the commenter stated that more current references than those from the 1970's should be used or the NRC should do new research in the area to develop more recent guidance for design review.

Response: As stated on Page 5–2 of the SER, references 4 and 5 were consulted by the NRC staff to determine the appropriate values for the assumed cobalt impurity levels in the fuel assembly hardware. Reference 5 is more recent and was published in 1993.

Comment D–2: One commenter asked what is the ''potentially oxidizing material'' that must be removed from the cask to protect the fuel cladding during storage.

Response: Potentially oxidizing impurities include oxygen, carbon dioxide, carbon monoxide, and water. Oxidizing impurities, their removal, and their effects are discussed in detail in PNL–6365, ''Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel'' November 1987.

Comment D–3: One commenter requested that ''fuel fines'' be defined.

Response: From NUREG/CR–6487, ''Containment Analysis for Type B Packages Used to Transport Various Contents'' November 1996, fuel fines are particulate material composed of fuel compounds and are produced as a result of mechanical stresses at both the fuel-cladding interface and the fuel pellet-fuel pellet interface. This definition is applicable to both transport and storage of light water reactor spent fuel.

Comment D–4: One commenter recommended that reference 9, in NRC Regulatory Guide 1.25, U.S. Nuclear Regulatory Commission, ''Assumptions Used for Evaluating Accidents in the Fuel Handling and Storage Facilities for Boiling and Pressurized Water Reactors' (March 1972), should be revised by the NRC and updated.

Response: Updating this Regulatory Guide is beyond the scope of this rule.

Comment D–5: One commenter suggested that a berm be used in the design.

Response: Under 10 CFR 72.212(b)(2), each general licensee who uses the TN–68 cask must perform an evaluation to show that the regulatory off-site dose limits are met at the licensee's site. The evaluations are made available for NRC inspection and review. Depending on a number of site specific factors including cask array size and distance to the nearest member of the public, a berm may or may not be needed.

Comment D–6: One commenter suggested that reference 1 listed on Page 10–4 of the SER, dated 1978, be updated.

Response: Updating reference 1 (Regulatory Guide 8.8) is beyond the scope of this rule.

Comment D–7: One commenter stated that on page 3–5 of the SER, the third paragraph ends in an extraneous ''0.''

Response: The NRC agrees with this comment and the SER has been changed accordingly.

Comment D–8: One commenter stated that on page 7–6 of the SER, reference 5 should be updated to reflect issuance of ISG–5, Revision 1.

Response: The NRC agrees with this comment. ISG–5 Revision 1 and the draft of the TN–68 SER were issued at nearly the same time. Because the principles and methods described in the revised ISG were reflected in the SER, it is appropriate to revise the SER to update this reference.

E. Technical Specifications

Comment E–1: One commenter stated that the use of logical connectors makes technical specifications difficult to read. The commenter asked if industry workers have commented on the technical specifications and find them easy to understand.

Response: The NRC disagrees with the comment. The TSs are modeled on the Improved Standard Technical Specifications (ISTS) for power reactors. The ISTS were developed as a result of extensive technical meetings and discussions between the NRC staff and the nuclear power industry in the early 1990's, in an effort to improve clarity and consistency of the power TSs and to make them easier for the operators to use. The most likely users of the TN–68 TSs are power reactor licensees familiar with the format of the ISTS.

Comment E–2: One commenter questioned why there are extensions of time intervals in the surveillance requirements and stated that the surveillance should be done according to schedule. The commenter stated that the 25-percent extension of the specified interval for performance of surveillance in the TS will be confusing and used when not applicable. The commenter also stated the same goes for the delay period of up to 24 hours or up to the limit of the specified frequency when it is discovered a surveillance has not been performed. The commenter suggested that extensions and extra leeway should be the explained exceptions rather than the regular allowance, and that the writeups were too complicated with too many options.

Response: The NRC disagrees that extensions of time should not be allowed. The basis for surveillance requirement (SR) 3.0.2 is discussed in the TN–68 Technical Specification Bases Section B 3.0 ''Surveillance Requirement Applicability.'' This section explains the NRC staff's rationale for allowing a 25-percent extension in the completion of periodic surveillances. The NRC staff believes that the 25-percent extension does not significantly degrade the reliability that results from performing the surveillance at its specified frequency. For those cases where it is necessary to adhere to a strict time frame for completing a surveillance, the specific SR will state that the 25-percent extension of SR 3.0.2 is not applicable. The 25-percent extension is also not applicable in cases when a surveillance frequency is specified by a regulation, because regulatory requirements take precedence over TSs. The NRC staff believes that the provisions of SR 3.0.2 are clear to users of the TSs, and that they will ensure that all required surveillances will be performed within an acceptable time period, consistent with the NRC staff's safety analyses.

Comment E–3: Two commenters requested changes to the maximum rod pitch and minimum rod outside diameter in TS 2.1. One commenter requested removal of these parameters because they cannot be verified by direct means. The other commenter requested that the values be specified as nominal [in the TS].

Response: The NRC disagrees with removing the parameters and changing them to nominal values. This design information is crucial to the conclusions reached by the NRC staff in its SER. The rod pitch and diameter, along with other design parameters, already include any design tolerances considered in the SAR. As stated in the TS bases for TS2.1, that have been modified for clarification, these parameters may be checked by administrative review.

Comment E–4: Two commenters requested changes to the maximum uranium content in TS 2.1. One commenter requested removal of this parameter because it may be overly restrictive. The other commenter requested that the values be specified as nominal.

Response: The NRC staff disagrees that the maximum uranium content parameters should be changed. This design information is crucial to the conclusions reached by the NRC staff in its SER. The TS limits on uranium content are based on the most limiting values used in the criticality and shielding analyses and include any design tolerances considered in the SAR. SAR table 5.2–1 shows that the calculated maximum uranium content used in the shielding analysis is higher than actual values. Although TS Basis 2.1.1 states that the shielding evaluation is based on nominal uranium content, the values used in the SAR evaluation are either greater than or equal to the TS values. TS Basis 2.1.1 has been changed to clarify those values.

Comment E–5: Two commenters stated that the channel thickness in TS 2.1 should be identified as a nominal value instead of a maximum [in the TS].

Response: The NRC staff agrees with this comment. However, the applicant provided the maximum rod channel thickness and the supporting analysis in its submittal, and did not provide analysis to support nominals. Therefore, the TS has not been changed, although the basis has been modified for clarification.

Comment E–6: One commenter asked what are boiling water reactor (BWR) fuel assembly channels.

Response: A fuel channel is the part of the BWR fuel assembly that surrounds the fuel bundle. The channel is located between the upper and lower tie plate and is made of Zircaloy. Channels perform functions that form a flow path for bundle coolant flow, provide surfaces for control rod guidance, provide structural stiffness to the bundle, and provide for in-core fuel sipping.

Comment E–7: Two commenters stated that the parameter labeling of Table 2.1.1–1 of the TS should be revised as Minimum Initial Enrichment and Maximum Burnup.

Response: The NRC agrees with this comment for clarification of values. TS Table 2.1.1–1 has been revised to use the terms Minimum Initial Enrichment and Maximum Burnup. Footnotes clarifying that the actual minimum enrichment is to be rounded down and burnup is to be rounded up were also added to the Table. Additionally, a discussion related to the footnotes was added to the bases for the TSs (B2.1.1) located in Chapter 12 of the SAR.

Comment E–8: One commenter asked for clarification on whether the cask could be put in the pool for 30 days or only 7 days when cask cavity drying pressure could not be established within limits, and if so, why.

Response: TS 3.1.1 provides the requirements for cask cavity vacuum drying. The action statements are to be implemented when a condition requiring entry into the ACTIONS exists. The action statements for this TS provide for interim cooling of the fuel and basket by establishment of a nominal helium environment if vacuum drying was not completed within the specified time. A 7-day limit to unload fuel is applicable if a nominal helium environment is not achieved. A longer, 30-day limit to unload fuel is applicable when a nominal helium environment has been achieved. These time limits provide time to take reasonable measures to complete fuel unloading while minimizing the time duration that the fuel is in a condition other than that required for long term storage. A complete discussion is provided in the bases for this TS.

The time limits do not imply how much time the cask must spend in the pool. The actual amount of time the cask is in the pool is a site-specific issue and beyond the scope of this rule. However, when the cask is returned to the pool and the lid is removed, the water surrounding the fuel will provide adequate cooling.

Comment E–9: One commenter stated that an example 1.4–3 of an ''otherwise stated'' exception to the applicability to the surveillance required by Limiting Condition for Operation (LCO) 3.1.6 should be added to the TS.

Response: NRC disagrees with this comment. The existing examples of Section 1.4 provide sufficient clarification for the correct interpretation of the TSs. These examples were developed as part of the Improved Standard Technical Specifications initiative through extensive interactions between the NRC staff and industry representatives. TS 3.1.6 clearly indicates when the surveillance requirement applies, and no additional explanation is considered necessary.

Comment E–10: One commenter stated that on page 3.1.1–1 of the TS, LCO 3.1.1 requires, ''* * * from pumping station.'' For consistency in terminology, ''pumping'' should be changed to ''vacuum drying''.

Response: The NRC agrees with the comment and the TS has been changed to ''vacuum drying''.

Comment E–11: One commenter stated that on page 3.1.1–2 of the TS, SR 3.1.1.1 should be changed from '' * * * at least 30 minutes'' to read, ''Verify that the equilibrium cask cavity vacuum drying pressure is brought to ≤ 4 mbar absolute for ≥ 30 minutes.''

Response: The NRC agrees that the comment adds clarity and has changed the TS to ''≥ 30 minutes.''

Comment E–12: One commenter stated that on page 3.1.2–1 of the TS, the Required Action and Completion times for LCO 3.1.2 are provided without technical basis and should be revised. The commenter further stated that on page 3.1.2–2 of the TS, the Frequency for SR 3.1.2.1 should be changed from 42 to 48 hours.

Response: The NRC disagrees with this comment. The heatup analysis provided by the cask applicant only supports a 48-hour elapsed time from the completion of cavity draining to completion of helium backfill. The completion time of the SR in 42 hours allows time (6 hours) to implement action A.1 if the SR is unsatisfactory. Action A.2 allows 48 additional hours to troubleshoot/repair and reperform the SR provided A.1 is also completed. The SAR, Section 4.6.2, TS Bases B 3.1.2, and the SER provide the technical basis, which shows that the vacuum drying and helium backfill must be completed within 48 hours to maintain cask component temperatures below their allowable temperature limits. The commenter provided no technical basis supporting additional time for completion of the helium backfill and allowance of time to implement appropriate corrective actions as outlined in the action.

Comment E–13: One commenter stated that on page 3.1.5–1, all conditions and required actions have not been identified.

Response: The NRC disagrees with the comment. It is the intent of the TSs to specify the minimum requirements for safe operations and the required actions if the minimum requirements are not met. A complete discussion on TS use and application is provided in TS 1.0. The bases of TS 3.1.5 addresses investigation of the cause of the low pressure condition. If the investigation finds that the cause of the low pressure condition is leakage above the allowable limit, then the appropriate TS action for this condition would also be implemented.

Comment E–14: One commenter stated that on page 3.1.5–2 of the TS, the Frequency of SR 3.1.5.2 should be changed from ''Once, within 7 days of commencing STORAGE OPERATIONS and every 36 months thereafter'' to read, ''Once, within 7 days of commencing STORAGE OPERATIONS *AND* 36 months thereafter.''

Response: NRC agrees with the comment. To make the format of the surveillance requirements consistent, the Frequency statement has been revised to read, ''Once, within 7 days of commencing STORAGE OPERATIONS *AND* 36 months thereafter.''

Comment E–15: One commenter asked if the cask can weep and has this been verified on a real cask.

Response: No TN–68 casks have been loaded and none have been tested for weepage. However, the TN–32 casks are of very similar design, and these casks have been loaded at two reactor sites. Slight weepage has occurred, but has not caused a problem with cask handling and storage. The TN–68 casks must be below the surface contamination levels in TS 3.2.1 before they can be moved to the storage pad.

Comment E–16: One commenter stated that the frequency for Surveillance Requirement 3.1.3.1 should read, ''Once prior to TRANSPORT OPERATIONS.'' Two commenters stated that the frequency for Surveillance Requirement 3.1.4.1 should read, ''Once prior to TRANSPORT OPERATIONS *OR* Once within 48 hours of commencing STORAGE OPERATIONS.''

Response: TS surveillance requirement 3.1.3.1 currently states ''Once, prior to TRANSPORT OPERATIONS,'' therefore no change is required. For TS surveillance requirement 3.1.4.1, the NRC agrees with the comment to revise the frequency requirement for clarification as follows: ''* * * *OR* Once within 48 hours of commencing STORAGE OPERATIONS.'' The affected TSs have been revised as indicated.

Comment E–17: Two commenters stated that the frequency of Surveillance Requirement 3.1.6.1 of the TS should be revised from ''Once, after lifting cask'' and prior to cask transfer to or from ISFSI'' to ''prior to lifting the cask''.

Response: The NRC agrees with this comment. It is acceptable to perform the surveillance requirement before lifting the cask. The TS frequency requirement of SR 3.1.6.1 has been changed to state ''Once, immediately prior to lifting the cask and prior to cask transfer to or from ISFSI.''.

Comment E–18: One commenter asked why 200 gallons of fuel in the transporter is the limiting factor for fire and explosions in the site-specific parameters. The commenter states a plane crash into a full cask array with a full fuel load should be evaluated.

Response: The NRC disagrees with this comment. The 200 gallons of fuel for the fire accident is based on the amount assumed to be carried by the transporter. The fire duration for 200 gallons of fuel is 15 minutes. The analyzed fire is assumed to burn at 1550° F and is assumed to produce the worse case scenario of fire/heated air for the TN–68. The fire is assumed to fully engulf the cask, thus maximizing the heat input into the cask. Fire of this duration exposed to the outside of the cask would have little effect on the cask or its contents due to the thermal inertia of the cask.

Before using the TN–68 casks, the general licensee must evaluate the site to determine whether or not the chosen site parameters are enveloped by the design bases of the approved cask as required by 10 CFR 72.212(b)(3). Included in this evaluation is the verification that the credible sources of an external explosion do not produce an external pressure above 25 psi and that any cask handling equipment used to move the TN–68 cask to the pad is limited to 200 gallons of fuel (refer to TS 4.3.5—Site Specific Parameters and Analyses). Also, when a general licensee uses the cask design, it will review its emergency plan for effectiveness under 10 CFR 72.212. This review will consider interdiction and remedial actions to address accidents of all types and coordination with local emergency response teams.

Comment E–19: One commenter stated that within LCO 3.2.1b, the values should read 20 dpm/100cm^2 instead of 20 dpm/cm^2.

Response: The NRC agrees with this comment. This was a typographical error and LCO 3.2.1b has been corrected.

Comment E–20: Two commenters stated that LCO 3.2.1 would require entry in the action as soon as loading operations commenced, and that the applicability for LCO 3.2.1 should be changed to ''During TRANSPORT OPERATIONS.'' One commenter stated that if the applicability is not changed, a note should be added to CONDITION A to clarify the intent of the specification. The other commenter stated that the applicability of LCO 3.2.1, the required action, and the completion time do not adequately address the retrieval of a cask from an ISFSI to the spent fuel pool to unload the cask, and that SR 3.2.1.1 should be performed before moving a cask from any restricted area.

Response: Action under LCO 3.2.1 is not necessary until the contamination surveillance has been completed. Transport of the cask to the ISFSI storage pad cannot begin until the cask surface is below the decontamination limit. The surveillance requirement is part of the loading phase. A note has been added to LCO 3.2.1 and to the basis for the TS (B3.2.1) located in Chapter 12 of the SAR which states that CONDITION A is not applicable until after the surveillance for surface contamination has been completed.

Regarding cask retrieval and unloading, the primary focus of LCO 3.2.1 is to maintain radioactive contamination and associated personnel exposures As Low As Reasonably Achievable (ALARA). The timing and nature of specific corrective actions are determined by the cask user under the user's radiation protection programs, other relevant programs, and applicable regulations, including 10 CFR part 20, subpart C, Occupational Dose Limits.

Decisions on unloading a cask will be made on a case-by-case basis if appropriate decontamination can not be achieved.

Comment E–21: One commenter stated that on page 4.0–3 of the TS : the title and first paragraph should be changed from site specific to ISFSI specific for clarity; item 3 should be changed to state, ''Seismic loads on the ISFSI pad * * *''; and engineered features to reduce radiation exposure should be classified as ''not important to safety.''

Response: The NRC agrees with comments 1 and 2. The terminology in TS 4.0.3 has been revised to indicate ''ISFSI * * *'' in the title and the first paragraph since this is a general license that is not site-specific. Item 3 has been revised to state ''Seismic loads on the ISFSI pad * * *'' The third comment on engineered features is addressed in the response to comment E–30.

Comment E–22: One commenter stated that the TS indicates that the cask cavity vacuum drying process evaporates any water that has not drained from fuel or basket surfaces. The commenter expressed concern about water not on the specified surfaces and asked what in the cask, including the cask materials, has or could also contain water.

Response: In preparation for dry storage, the loading process ensures the removal of virtually all moisture and oxidizing gases (less than 1 gram-mole per cask) from the fuel cladding, any fuel that may have pinholes or hairline cracks, and from the cask internals. The cask internals do not provide any locations for significant moisture entrapment. The cask is thoroughly vacuum dried, as prescribed in the TSs and the SAR. The vacuum drying process, which involves two, complete, evacuate-fill cycles, coupled with the heat generation of the fuel, very effectively removes residual moisture that may be present in the fuel pellets and interior components of the cask system and oxygen that is inside the cask. The helium fill gas is very pure and dry and the cask is sealed to prevent entry of water and air during storage. The effectiveness of the vacuum drying process, the sources of residual impurities, and the potential effects of impurities, are reported in PNL–6365, ''Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel'' November 1987.

Comment E–23: The commenter asked what is BorALYN™, borated wrought aluminum, and other envisioned alternate neutron absorber materials, and if NRC has read the manufacturers' descriptions as to what is in these materials, their limitations for use, and their reactions with other materials.

Response: BorALYN™ is a trademark for a ceramic of boron carbide particles, which are produced using natural boron, *e.g.,* boron containing the isotopic mix found in nature. In BorALYN™, these particles are in a matrix (formed mechanically with heat and pressure) of a common and widely used aluminum alloy. NRC has visited the plant where this product is produced to review details on the process used to produce BorALYN™. NRC has required the applicant to do extensive durability testing of the material. NRC has reviewed the results of these tests and found this material to be acceptable for this application.

Borated aluminum is a wrought aluminum alloy (made from the liquid state) that uses an enriched boron as an alloy addition to the alloy. Natural boron contains a high-cross section isotope called ^{10}B, that is many times more effective at capturing thermal neutrons than ^{11}B, the other isotope of boron. The neutron absorber must capture thermal neutrons during loading and unloading operations. Enrichment refers to the concentration of ^{10}B.

Other alternative neutron absorber materials are like the BorALYN™ and the borated aluminum, except that they are made with slight variations, *e.g.,* the base material is stainless steel in one case, the boron carbide particles are a different size in another case, etc. All materials approved for use are materials sufficiently nonreactive as to be suitable for the environments that the materials must tolerate well in service conditions for normal, off-normal, and accident conditions. None of these absorber materials have special limitations in relation to the function that they must perform in the cask systems for which they have been approved.

Comment E–24: The commenter stated that any material encased or welded inside another may either expand or contract with the heat in the cask, or react chemically if residual water remains.

Response: Encased material may expand and contract relative to temperature changes. Thermal expansion/contraction of cask components was evaluated in the TN–68 SAR Section 3.4.4.2. This evaluation was acceptable to the NRC. See the response to comment E–22 regarding moisture in the cask cavity.

Comment E–25: The commenter expressed concern about water leaking into encased areas if a cask is allowed to remain in a pool for seven or more days, and asked if the casks are really leak tight, citing the port vent and drain hole areas specifically. The commenter also asked if leak tightness has been checked and how the cask is checked for water retention after soaking for the seven days.

Response: See the response to comment E–22 regarding moisture in the cask cavity. The remainder of the cask is designed to preclude water intrusion and retention for the purposes of decontamination. For example, the shell that encases the radial neutron shield is sealed and leak tested after fabrication as described in SAR Section 9.1.2. If water contacts the polymeric resins, they are not expected to react with the water, nor are the metals expected to react to any extent that could affect safety of the system. The vent and drain port areas as well as the seal areas are thoroughly dried during preparation for storage.

Comment E–26: One commenter asked why seven days is allowed to reflood the cask and unload the fuel when a nominal helium environment cannot be achieved. The commenter noted that the cask can go into the pool for 30 days when the drying pressure limits cannot be achieved, and also asked why one limit is for seven days and one is for 30 days.

Response: TS 3.1.1 provides the requirements for cask cavity vacuum drying. The action statements are to be implemented when a condition requiring entry into the ACTIONS exists. The action statements for this TS provide for interim cooling of the fuel and basket via establishment of a nominal helium environment if vacuum drying was not completed within the specified time. A 7-day limit to unload fuel is applicable if a nominal helium environment is not achieved. A longer 30-day limit to unload fuel is applicable when a nominal helium environment has been achieved. These time limits provide for reasonable measures to complete fuel unloading while minimizing the time duration that the fuel is not in a suitable long-term storage condition. A complete discussion is provided in the bases for this TS.

The time limits do not imply how much time the cask must spend in the pool. The actual amount of time the cask is in the pool is a site-specific issue and beyond the scope of this rule. However, when the cask is returned to the pool and the lid is removed, the water surrounding the fuel will provide adequate cooling.

Comment E–27: One commenter stated that the cell opening and boron loading should be removed from Section 4.1.1 of the TS.

Response: The NRC disagrees with the comment. Design features that may affect safety if altered or modified are included in the TS. As stated in SAR Section 6.1, the TN–68 cask design parameters relied upon for criticality safety control are the fuel assembly spacing and the use of the neutron absorbing plates. This design information is crucial to the conclusions reached by the NRC staff in its SER. Design tolerances considered in the SAR for the boron loading and the cell opening for the basket are included in the TS limits.

Comment E–28: One commenter stated that Section 4.1.3, Codes and Standards, should be removed from the TSs.

Response: The NRC disagrees with the comment. This information is required under 10 CFR 72.24(c)(4).

Comment E–29: One commenter stated that in the Storage Location for Casks, 4.2.1 of the TS, the 16-foot dimension should be listed as a minimum value or a tolerance should be added.

Response: The NRC disagrees with this comment. As written, the TS states that ''the casks shall be spaced a minimum of 16 feet apart, center-to-center.'' This specification assures that the minimum cask spacing assumed in the analysis is achieved to allow proper dissipation of radiant heat energy.

Comment E–30: One commenter stated that references to consideration as important to safety, be removed from Section 4.3.6 of the TS.

Response: The NRC disagrees with this comment. As defined in 10 CFR 72.3, structures, systems, and components important to safety are those features of the ISFSI or MRS whose function is to maintain the conditions required to store spent fuel safely. Thus, when a berm or other system, structure, or component is installed to meet the normal condition dose limits of 10 CFR 72.104 (*i.e.,* to provide safe storage), it is considered important to safety. However, under 10 CFR 72.122, the quality standards for the feature's design, fabrication, erection, and testing may be at a level commensurate with the safety importance of the function to be performed. In general, features that are not needed to meet the accident conditions will not have to meet as high a standard as those that need to function in an accident.

Comment E–31: One commenter stated that on pages 5.0–3 through 5.0–5 of the TS, describing the cask surface dose rate evaluation program, inconsistent terminology is used regarding the neutron shielding. A single term ''radial neutron shield'' should be used consistently.

Response: The NRC agrees with this comment. In the interest of clarity, TS 5.2.3 has been revised to consistently use the term ''radial neutron shield'' where appropriate.

Comment E–32: One commenter stated that on page 5.0–5 of the TS, the reference to Figure 5.2.3–1 should be deleted.

Response: The NRC disagrees with this comment. Figure 5.2.3–1 is provided as a quick reference for the user and the public to help interpret the measurement locations given in TS 5.2.3.7. The figure is an illustration, not to scale, and the specification wording more exactly defines the location of each measurement.

Comment E–33: One commenter stated that the NRC did not clearly state why the interior cannot be preferentially or unevenly flooded and asked why the NRC did not analyze the scenario of a cask partially filled with unborated water and steam.

Response: As stated in SAR Section 6.1, nonuniform flooding of the basket is not credible because all spaces in the basket are interconnected. The applicant evaluated the failure of the four center basket cavities to drain and showed that this was significantly less reactive than a fully flooded cask. As stated in SER Section 6.3.1, the applicant varied the water density in the cask to bound any possible density changes during loading and unloading operations. The full density water resulted in the highest reactivity in all cases.

Comment E–34: One commenter asked which fuel assembly has the highest reactivity; 7x7 GE2, GE2b, or 10x10. Further, the commenter asked why the NRC does not have a third party verify both the NRC's and applicant's calculations.

Response: As shown in SAR Table 6.4–3, the applicant evaluated both the 7x7 and 10x10 assemblies for all normal, off-normal, and accident conditions. The results in this table show that the 10x10 assembly is the most reactive under the most bounding conditions. Because the NRC staff has reasonable assurance that the cask meets the design criterion for criticality safety, further verification by a third party is not required. s

Comment E–35: One commenter stated that on page 3–17 of the SER, reference 4 should be changed to, ''ANSI N14.6, Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials, 1986.'' The commenter also stated that on page 9–8 of the SER, reference 5 should be changed from ANSI N14.6–1993 to ANSI N14.6–1986.

Response: The NRC disagrees with this comment. ANSI N14.6–1993 was used by the NRC staff in this evaluation.

Comment E–36: One commenter stated that on page 4–9 of the SER, the second sentence in the first paragraph under Section 4.5.2.4 should be changed to, ''Assuming design basis heat load fuel and completion of cask cavity drying, helium backfill should be completed within 48 hours.'' This change is needed to conform to TS 3.1.2.

Response: The NRC disagrees with this comment. The heatup analysis provided by the cask applicant only supports a 48-hour elapsed time from the completion of cavity draining to completion of helium backfill. The commenter did not provide a technical basis supporting an additional 48 hours.

Comment E–37: One commenter stated that on page 5–3 of the SER, the use of spectral shift void history on early design fuel (7x7) by TN provides considerable conservatism and should be reconsidered.

Response: The NRC disagrees with this comment. The analysis provided to support a general license design, which applies to all licensees, needs to bound all variations of cask contents unless compensating factors are present. The operational parameters assumed to determine the source term in the design basis fuel need to cover the range of both current and past operating practices of all authorized users.

Comment E–38: One commenter stated that in Table 7–1 of the SER, the percentage of rods that failed in off-normal and accident conditions are not consistent with industry experience and research. More reasonable values are on the order of 0.0001% and 0.01% for off-normal and accident conditions respectively.

Response: The rod breakage fractions presented in Table 7–1 of the SER were based on those already contained in NUREG–1536, ''Standard Review Plan for Dry Cask Storage Systems'' as discussed on page 2–7. This NUREG was previously subject to public comment. Currently, the NRC is confident that the rod breakage fractions are bounding and provide reasonable assurance of public safety with regard to the confinement analyses of spent fuel storage casks. Further, NRC and industry initiatives to modify assumptions for rod breakage fractions are beyond the scope of this rule.

Comment E–39: One commenter stated that in Table 7–1 of the SER, the meteorological conditions to be used to analyze the offsite dose consequences should be consistent with those used for the power plant.

Response: The NRC disagrees with this comment. Since the meteorological conditions for a specific site are not known, the NRC has made bounding assumptions for meteorological conditions to establish a basis for cask approval. General licensees who use a cask approved under 10 CFR 72, subpart L, must calculate dose equivalents for their ISFSIs, considering site-specific meteorology, other exposure pathways such as ingestion and ground deposition, and actual distances to the site boundary.

Comment E–40: One commenter stated there should only be a TEDE limit in Table 7–2 of the SER and that the calculation of other doses is redundant.

Response: The NRC does not agree with this comment. Whole body (TEDE) and organ dose limits are required in 10 CFR 72.104 and 10 CFR 72.106. Also, 10 CFR 72.106 provides dose limits on skin and the lens of the eye. Therefore, evaluation of these doses is needed for cask approval.

Comment E–41: One commenter stated that on page 8–4 of the SER, the last paragraph in Section 8.3.2 refers to a check valve to restrict cooling water flow if cask pressure exceeds 90 psia. A pressure control valve would provide the desired capability.

Response: The NRC agrees with the comment that either valve will satisfy the requirement to restrict flow. The SER Section 8.3.2 has been changed to reflect that a valve designed to restrict flow will act to restrict cooling water flow if cask pressure exceeds 90 psia, which will allow flexibility by the cask user. The SAR has also been revised by the applicant to reflect this change.

Comment E–42: One commenter stated that on page 10–3 of the SER, the last paragraph under Section 10.3.1 should be deleted.

Response: The NRC disagrees with this comment. As defined in 10 CFR 72.3, structures, systems, and components important to safety are those features of the ISFSI or monitored retrievable storage installation (MRS) whose function is to maintain the conditions required to store spent fuel safely. Thus, when a berm or other system, structure, or component is installed to meet the normal condition dose limits of 10 CFR 72.104 (*i.e.,* to provide safe storage), it is considered important to safety. However, under 10 CFR 72.122, the quality standards for the feature's design, fabrication, erection, and testing may be at a level commensurate with the safety importance of the function to be performed. Therefore, the last paragraph is necessary.

Comment E–43: One commenter stated that on page 11–1 of the SER, the last sentence under Section 11.0 should be changed from SAR Revision 4 to SAR Revision 5.

Response: The NRC agrees with this comment and has updated page 11–1.

F. Comments on Applicant's Topical SAR

Comment F–1: One commenter stated that on page 8.1–3 of the SAR, the first sentence of the description for the cask transporter should be changed to read, ''The cask transporter is generally set to limit the lift height of the cask to ensure that the maximum gravitational loading force limit in the event of a cask drop is met.''

Response: The NRC agrees with the comment with additional clarification. The SAR has been revised to state: ''The cask transporter is set to limit the lift height of the cask to ensure that the loads from a postulated drop accident will be bounded by the maximum analyzed loads given in Technical Specifications 4.1.2 and 5.2.2.''

Comment F–2: One commenter stated that drawing 972–70–1 of the SAR should be revised to add a tolerance of +0/−.25 to 13.25-inch dimension to accommodate variations due to welding.

Response: The NRC accepts this change to the tolerance specified on Drawing No. 972–70–1 because it will not affect the structural analyses and the conclusions reached in the SER. Drawing No. 972–70–1 has been changed accordingly.

Comment F–3: One commenter stated that drawing 972–70–4 of the SAR should be revised to add note 6 to allow the clearance hole in the rail at the end to be optional. The size of the clearance hole should be increased from a 2.00-inch diameter to a 3.56-inch diameter to allow sufficient clearance for a socket wrench.

Response: The NRC accepts these changes to the clearance hole in the rail because they will not affect the structural analyses and the conclusions reached in the SER. Drawing No. 972–70–4 has been changed accordingly.

Comment F–4: One commenter stated that Note 2 on drawing 972–70–5 of the SAR should be revised from ''PT examination per ASME Section III, Subsection NG–5231'' to ''PT examination per ASME Section III, Subsection NG–5233.''

Response: The NRC disagrees with the comment that Note 2 on Drawing 972–70–5 needs to be changed. For thin, one-layer welds without filler material, ASME Section III, Subsection NG–5231 is still applicable. For clarification of the nondestructive examination requirement in NG–5231, Table 4.1–1 of the TSs has been revised.

Comment F–5: One commenter stated that drawing 972–70–6 of the SAR should be revised to add a note to allow alternate plumbing configurations. Also, an additional connection may be required through the protective cover for helium leak testing of the over pressure (OP) system.

Response: The NRC agrees with this comment. Alternate plumbing configurations will add flexibility to the design of the OP system without adversely affecting the structural analyses and the conclusions reached in the SER. The note should also state that the parts and equipment used are equivalent to those specified in the drawing. An adequate level of safety is obtained by the quality assurance process, plus the leak testing and monitoring of the system as required by the TSs. The addition of a test fitting in the protective cover does not affect safety because its purpose is to facilitate leak testing of the overpressure monitoring system. Drawing No. 972–70–6 has been revised to reflect these changes.

Comment F–6: One commenter stated that it is not possible to perform PT on the Plasma-Arc Welding (PAW) part of the weld since the Gas Tungsten-Arc Welding (GTAW) is part of the automatic welding equipment. Transnuclear has proposed a code case to Section III, Subsection NG, on this issue for guidance.

Response: The NRC agrees with the applicant's view that inspection after PAW is not practical and that inspection after GTAW is adequate. The proposed code case is beyond the scope of NRC review.

G. Accidents

Comment G–1: One commenter asked if a cask will slide on the pad and could slide into other casks or other structures in the independent spent fuel storage installation (ISFSI), stated that the pad was described in a site-specific manner instead of generically, and asked what structures or vehicles are permitted to be within the ISFSI fence.

Response: The SAR indicates that the cask may slide 7.3 inches due to a 4,000 lb. missile (in this case, an automobile) impacting below the center of gravity of the cask at 126 mph. This is much smaller than the approximately 94-inch distance between casks. Therefore, impacts between TN–68 casks on the pad would not occur. In the unlikely event that two 4,000 lb missiles were to impact below the center of gravity of two adjacent casks from opposite directions at the same time, the two casks still would not collide with each other. Furthermore, the automobile is conservatively assumed to be rigid and absorbs no energy in the analysis. In an actual impact, the majority of the energy will be absorbed by the crushing of the automobile rather than moving of the cask. The pad is a site-specific issue that needs to be addressed in the cask user's 10 CFR 72.212 evaluation. TS 5.2.1, referenced by the commenter, simply requires the cask user to verify that the coefficient of friction for the concrete pad matches the coefficient of friction used in the SAR's cask sliding analysis. The structures and vehicles permitted within the ISFSI fence is a site-specific issue and is beyond the scope of this rule.

Comment G–2: One commenter stated that all things in loading and unloading areas should be evaluated for a cask drop or tip over accident.

Response: This comment is beyond the scope of this rule. The use of a generally licensed cask by a utility requires that the user ensure that the site is not subject to any potential accident that has not been analyzed for the general license.

Comment G–3: One commenter noted that explosive overpressure is not addressed, stated this should be done now and should have been done before the SER was completed, and asked why it was not addressed. They stated that this evaluation is not suitable for a site-specific evaluation and should be addressed as part of the generic review. The commenter suggested that a sabotage explosion such as a truck bomb ramming the fence or a plane explosion needs evaluation for current cask approval.

Response: NRC disagrees with this comment. The TN–68 is designed to withstand an external pressure of 25 psi. This would include a nearby explosion, debris falling on the cask, etc. If a credible explosion is identified that would apply more than 25 psi to the outer surface of the cask at a site, the site will have to address this issue in its 10 CFR 72.212 evaluation.

Comment G–4: One commenter stated that earthquake analysis should not rely on site analysis for the nuclear power plant because the analysis for the plant does not apply to the pad, and the plant and pad are not on the same soil location.

Response: The NRC disagrees with the recommendation that each ISFSI pad be required to have a specific seismic analysis. This is beyond the scope of this rule. The licensee using a particular cask design has the responsibility under the general license to evaluate the match between reactor site parameters and the range of site conditions (*i.e.,* the envelope) reviewed by the NRC for an approved cask. The licensee should also consider if there are any site conditions associated with the actual pad and cask locations that could affect cask design and that were not evaluated in the NRC SER for the cask.

Comment G–5: One commenter stated that the effects of lightning need to be evaluated.

Response: The effects of lightning are addressed in Section 2.2.5.2.8 of the SAR. Section 3.1.2.1.8 of the SER has been revised to clearly indicate this fact.

Comment G–6: One commenter asked if there is a more recent reference document than the 1974 document referenced in the CoC that addresses tornadoes.

Response: The document referenced in the CoC that addresses tornadoes is a Regulatory Guide entitled ‘‘Design Basis Tornado for Nuclear Power Plants.’’ There has been no revision on this Regulatory Guide after the 1974 publishing date.

Comment G–7: One commenter asked why the lid is not modeled for maximum temperature in storage conditions and the cask bottom is not modeled for peak temperature in a fire accident.

Response: The cask lid will perform its intended safety function (confinement) for the normal conditions of storage. The cask bottom will perform its intended safety function (confinement) for the fire accident. Based on the applicant’s modeling and analysis which demonstrated that there was no challenge to the safety functions of these components, explicit modeling of these components in the conditions specified by the commenter was not required.

Comment G–8: One commenter asked if an emergency plan had been developed to retrieve a buried cask, how a TN–68 cask would be excavated in the most efficient and rapid way, and has this been evaluated. The commenter asked if emergency staff at the site and in the nearby communities are trained to deal with cask fires, how training is administered, and if anyone oversees the training to ensure that it is effective.

Response: Cask general licensees are required by 10 CFR 72.212 (b)(6) to evaluate their emergency plans and revise them accordingly before using a cask certified under 10 CFR 72 subpart L. The details of site specific emergency response are beyond the scope of this rulemaking.

H. Radiation Protection

Comment H–1: One commenter had questions about radiation in a full cask array, particularly how the radiation or skyshine from casks of the same design and casks of different designs affect each other and if research has been done to evaluate the effects. The commenter also asked if surface dose rates should be taken again at the pad after the casks have been moved to the pad. The commenter also asked where most loaded casks are presently located.

Response: The shielding analysis addresses the interaction of radiation between the casks of the same design in a storage array. The interaction between casks of different designs is not a part of this rule, but is not expected to be significantly different than that considered in the original analysis. As a final check, each user of a storage cask must perform a site-specific analysis to show that the regulatory dose limits will be met at the user’s site including the effects of other cask designs if present.

For the purposes of TS 5.2.3, a second dose rate measurement is not needed after the cask has been moved to the storage pad. The normal and accident condition analyses of the cask show that the dose rates are not expected to change during transport to the storage pad. However, the licensee’s radiation protection program will include general area measurements at the pad.

The Oconee reactor site has the largest number of loaded dry storage casks.

Comment H–2: One commenter stated that Figure 5.2.3–1, which shows contact dose rate measurement locations, should be changed to show the cask trunnions.

Response: The NRC disagrees with this comment. Figure 5.2.3–1 is provided as a quick reference for the user and the public to help interpret the measurement locations in TS 5.2.3.7. Measurement locations with respect to the trunnions are contained in the specification. The exact location of the trunnions is shown in the SAR drawings.

Comment H–3: One commenter asked where Hansen couplings, basket key, basket rail shims, security wire and seals, and alignment pins are located on Figure 1–1 of the SER. The commenter also asked why Figure 1–1 of the SER does not show the gamma shield. The commenter stated that the figure also should better depict where the outer neutron shield is installed, and asked if the outer neutron shield stops above the bottom trunnion and below the top trunnion or goes around them. The commenter stated that the outer shell design is very unclear and that a better drawing is required.

Response: The NRC disagrees that a more detailed drawing is required in the SER. Figure 1–1 is only intended to depict the general configuration of the cask. The applicant’s SAR includes drawings and design detail that enable the NRC to make a safety finding. That same level of detail does not need to be repeated in the SER, because the SAR drawings are available on the docket and are retrievable by the NRC staff and the public. The neutron shield runs the full length of the active fuel region of the fuel assemblies which is the location of the neutron source term, extending from below the bottom trunnion to half-way around the top trunnion.

Comment H–4: One commenter stated that a date should be provided for reference 5 on page 4–12 and for reference 3 on page 6–8 of the SER, and that the NRC should add dates to all references as regular practice.

Response: Typically, computer codes are listed by version and not by date (*e.g.,* version 4.3, 4.4, etc.). ANSYS Version 5.4 was released in September, 1997. SCALE Version 4.4 was released in September, 1998. These dates were added to the SER.

Comment H–5: One commenter requested that the NRC clarify why the 1-inch thick steel shell above the radial neutron shield is optional.

Response: As stated in TS 5.2.3, the 1-inch thick shield does not need to be installed if it is not needed to meet the surface dose rate limits in the specification. The surface dose rate limits were taken from the shielding analysis.

Comment H–6: One commenter stated that the discussion on Page 5.2 of the SER concerning cobalt impurities in stainless steel is vague and is based on unrelated documents. Further, the commenter asked how much cobalt impurity can vary based on supplier and date of manufacture and how a fabricator knows what is being provided.

Response: The NRC disagrees that the documents are unrelated. The references are widely used reports produced by national laboratories and are considered to be appropriate sources of information for establishing the assumed cobalt impurity levels. Early on, cobalt impurities in fuel assembly hardware were not as well controlled as today and could vary; therefore, appropriate bounding values were established using the data in the references. After the effect of tramp amounts of cobalt became apparent, fabricators and designers began to specify limits on the cobalt content in materials procurement documents. In the last 10 to 15 years, fabricators typically specify the acceptable impurity limits as part of their procurement process subject to the applicable quality assurance procedures.

Comment H–7: One commenter had a number of concerns related to the cobalt content of stainless steel used in cask fabrication: What are the tolerance specifications for the components in the stainless steel and how varying the tolerance would affect their performance; how cobalt affects cask handling and unloading in any way; what cobalt data on a specific batch of stainless steel is reported by the supplier; and if this should be factored into analysis each time a new batch is used.

Response: Thermal (slow) neutrons are required to activate the cobalt in the components that make up the storage cask system. There are essentially no thermal neutrons that collide with these components in storage systems. Therefore, questions concerning the cobalt in this material are not relevant in relation to activation. As for mechanical properties, many if not all are likely to be enhanced by the addition of cobalt to the alloy, but this is not done for economic reasons. The cobalt might be reported by the supplier if it was at a high enough concentration to be detected by the analytical procedures that are normally used for chemical analyses of these alloys. Tramp elements are not always reported, except by special request. Therefore, the NRC staff is not concerned about cobalt in materials used for these components. See also comment H–6.

Comment H–8: One commenter stated concerns relating to how the neutron source is evaluated taking into account the natural uranium blankets used in the BWR fuel that has changed over the years. The commenter stated that a utility needs to carefully evaluate neutron sources to precisely reflect the fuel age and type that is to be loaded in casks, that TN erred in computing the neutron sources in the SAR table, and asked how an applicant could make such a mistake and how the NRC could accept such a mistake. The SAR neutron source table and its calculations need to be done correctly and the SAR needs to be revised to reflect the correct values before the NRC accepts the document.

Response: Less than 10% of the off-site dose comes from neutrons. Thus, uncertainties in the neutron source strength are not significant. A general license analysis does not need to be bounding in every term as long as the overall result is bounding. The NRC staff's review determined that the small underestimation of the neutron source term was more than compensated for by the applicant's overestimation for the gamma-ray source term. Therefore, the applicant's estimated dose from the cask is bounding. The general license analysis is based on generalized operating assumptions. However, each licensee user must perform a site-specific analysis to show compliance with the regulations. The site-specific analysis is the appropriate place to address the type, age, and operating conditions for the actual fuel to be loaded at the site.

Comment H–9: One commenter asked how the fuel reacts at the top and bottom of the cask when exposed to steam during quenching.

Response: Thermal stress associated with reflooding and quenching is discussed in SAR Sections 3.5.2 and 4.6.1. SER Section 4.1 contains the analysis and NRC acceptance of quenching effects described in the SAR.

Comment H–10: One commenter stated a concern with streaming at the trunnions and asked why detailed confirming calculations were not modeled, asked what is the trunnion material, asked whether the trunnions should be lowered, and stated that workers will have to be around the trunnions adjusting the lifting devices and that the vendor should work to reduce unnecessary doses.

Response: The modeling detail of the trunnions in the shielding analysis is at a level that equals the capability of the analytical code. Further detail in the trunnion calculations is not necessary because radiation streaming around the trunnions is very localized and will have negligible effect on meeting the regulatory limit for the off-site dose. Worker doses are subject to ALARA as discussed in item 4 below. The trunnions are made of steel with a central plug of borated polyester resin. Placement of the trunnions was a design decision made by the applicant and is beyond the scope of this rule. The shielding performance of the trunnion design has been reviewed and found to be adequate. The radiation protection program of the licensee user will have the responsibility to implement measures to keep the dose of workers around the trunnions as low as reasonably achievable. Any streaming points will be monitored and avoided during cask handling operations.

Comment H–11: One commenter asked why the neutron shield does not cover the entire cask and if the design is based on the location of the trunnions.

Response: Radially, except at the trunnions, the neutron shield runs the full length of the active region of the spent fuel assemblies, that are the source of neutron radiation. The design of the neutron shield is based on meeting the regulatory requirements and is acceptable.

Comment H–12: One commenter asked about the ''radiation return from radial neutron shield'' reduction of photon dose from 860 mrem/hr to 749 mrem/hr and why the NRC did not conduct confirmatory calculations to be sure that this reduction is correct. The commenter also recommended that the NRC should not accept expected values and should not leave it up to the licensee to determine how to maintain doses ALARA, but should instead provide guidelines as part of the approval process for this design.

Response: In lieu of performing a separate accident calculation, the NRC staff used the results from the normal conditions calculation to bound the dose rate at the cask surface. The NRC staff's analysis shows good agreement with the applicant's calculations. In addition, the maximum off-site dose from a cask under accident conditions is about one tenth of the regulatory limit. Even with a higher value of 860 mrem/hr, the performance of the cask in the hypothetical accident would be well within regulatory limits. Guidelines for a licensee's ALARA are contained in Regulatory Guide 8.8, ''Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low as Reasonably Achievable.''

Comment H–13: One commenter recommended that an eye lens calculation be added to Table 7–2 of the SER so that the effects of radiation dose to the eye can be known.

Response: The NRC has chosen not to add an eye lens calculation to Table 7–2. As discussed in the TN–68 SER, compliance with the dose-equivalent limit for the lens is achieved by demonstrating compliance with the dose-equivalent limit for the skin and the effective dose-equivalent limit. This approach is consistent with guidance in ICRP–26, International Commission on Radiation Protection, ''Statement from the 1980 Meeting of the ICRP,'' ICRP Publication 26, Pergammon Press, New York, New York, 1980.

Summary of Final Revisions

The NRC staff modified the rule language, the CoC, the TSs, and its SER.

Rule Language Change

The rule language has been modified to clarify that it is the Certificate that expires.

CoC Changes

The CoC has been changed for consistency with other issued certificates.

TN–68 TS Changes and Associated Comments

TSs were reformatted into Corel 8 WordPerfect™ software that addressed the editorial changes in comment B–17.

TS1.1 Definition of Intact fuel was revised based on the NRC staff's initiative.

Table 2.1.1–1 revised labels to add in minimum and maximum, and added three footnotes based on comment E–7 and the NRC staff's initiative.

LCO 3.1.1 was revised to state, ''from the vacuum drying system'' based on comment E–10.

SR 3.1.1.1 was revised to state, ''$\leq$ 4 mbar absolute for $\geq$ 30 minutes'' based on comment E–11.

SR 3.1.4.1 was revised to state, ''Once within 48 hours of commencing STORAGE OPERATIONS'' based on comment E–16.

SR 3.1.5.1 Frequency has been revised to state, ''OPERATIONS *AND* 36 months thereafter'' based on comment E–14.

SR 3.1.6.1 Frequency has been revised to state, ''Once, immediately prior to lifting cask'' based on comment E–17.

LCO 3.2.1 b. was revised to state, ''20dpm/100 cm^2'' based on comment E–19, and a note added ''Not applicable until SR 3.2.1.1 is performed'' based on comment E–20.

Table 4.1–1 has been clarified to address PT examination under ASME Section III, Subsection NG–5231, based on comment F–4.

TS 4.3 has been revised to state, ''ISFSI Specific'' and ''load on the ISFSI pad'' based on comment E–21.

TS 5.2.3 has been revised to use the terminology ''radial neutron shield'' throughout the section based on comment E–31.

Agreement State Compatibility

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

Finding of No Significant Environmental Impact: Availability

Under the National Environmental Policy Act of 1969, as amended, and the NRC regulations in Subpart A of 10 CFR part 51, the NRC has determined that this rule, if adopted, would not be 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 Gordon Gundersen, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 415–6195, email GEG1@nrc.gov.

Paperwork Reduction Act Statement

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

Public Protection Notification

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

Voluntary Consensus Standards

The National Technology Transfer Act of 1995 (Pub. L. 104–113) requires that Federal agencies use technical standards that are developed or adopted by voluntary consensus standards bodies unless the use of such a standard is inconsistent with applicable law or otherwise impractical. In this final rule, the NRC will add the Transnuclear TN–68 cask system to the listing within the list of NRC approved casks for spent fuel storage in § 72.214. This action does not constitute the establishment of a standard that establishes generally-applicable requirements.

Regulatory Analysis

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

The alternative to this action is to withhold approval of this new design and issue a site-specific license to each utility that proposes to use the casks. This alternative would cost both the NRC and utilities more time and money for each site-specific license. Conducting site-specific reviews would ignore the procedures and criteria currently in place for the addition of new cask designs that can be used under a general license, and would be in conflict with NWPA direction to the Commission to approve technologies for the use of spent fuel storage at the sites of civilian nuclear power reactors without, to the maximum extent practicable, the need for additional site reviews. This alternative also would tend to exclude new vendors from the business market without cause and would arbitrarily limit the choice of cask designs available to power reactor licensees. This final rule will eliminate the problems above and is consistent with previous NRC 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 discussion above 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

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

Regulatory Flexibility Certification

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

Backfit Analysis

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

List of Subjects in 10 CFR Part 72

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

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

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 § 72.214, Certificate of Compliance (CoC) 1027 is added to read as follows:

§ 72.214 List of approved spent fuel storage casks.

* * * * *

Certificate Number: 1027.

SAR Submitted by: Transnuclear, Inc.

SAR Title: Final Safety Analysis Report for the TN–68 Dry Storage Cask.

Docket Number: 72–1027.

Certificate Expiration Date: May 28, 2020.

Model Number: TN–68.

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

For the Nuclear Regulatory Commission.

Frank J. Miraglia, Jr.,

Acting Executive Director for Operations.

[FR Doc. 00–10390 Filed 4–27–00; 8:45 am]

BILLING CODE 7590–01–P

DEPARTMENT OF TRANSPORTATION

Federal Aviation Administration

14 CFR Part 39

[Docket No. 2000–NM–85–AD; Amendment 39–11699; AD 2000–08–13]

RIN 2120–AA64

Airworthiness Directives; Learjet Model 45 Airplanes

AGENCY: Federal Aviation Administration, DOT.

ACTION: Final rule; request for comments.

SUMMARY: This amendment adopts a new airworthiness directive (AD) that is