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10 CFR Part 72

RIN 3150–AH36

List of Approved Spent Fuel Storage Casks: Standardized NUHOMS®–24P, –52B, –61BT, –24PHB, and –32PT Revision

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is amending its regulations to revise the Transnuclear, Inc. (TN) Standardized NUHOMS®–24P, –52B, –61BT, and –24PHB cask system listing within the ''List of Approved Spent Fuel Storage Casks'' to include Amendment No. 5 to Certificate of Compliance (CoC) Number 1004. Amendment No. 5 will add another dry shielded canister (DSC), designated NUHOMS®–32PT DSC, to the authorized contents of the Standardized NUHOMS®–24P, –52B, –61BT, and –24PHB cask system. This canister is designed to accommodate 32 pressurized water reactor (PWR) assemblies with or without Burnable Poison Rod Assemblies. It is designed for use with the existing NUHOMS® Horizontal Storage Module and NUHOMS® Transfer Cask under a general license.

EFFECTIVE DATE: This final rule is effective on January 7, 2004.

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SUPPLEMENTARY INFORMATION:

Background

Section 218(a) of the Nuclear Waste Policy Act of 1982, as amended (NWPA), requires that ''[t]he Secretary [of the Department of Energy (DOE)] shall establish a demonstration program, in cooperation with the private sector, for the dry storage of spent nuclear fuel at civilian nuclear power reactor sites, with the objective of establishing one or more technologies that the [Nuclear Regulatory] Commission may, by rule, approve for use at the sites of civilian nuclear power reactors without, to the maximum extent practicable, the need for additional site-specific approvals by the Commission.'' Section 133 of the NWPA states, in part, 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 spent fuel storage cask designs. The NRC subsequently issued a final rule on December 22, 1994 (59 FR 65920), that approved the Standardized NUHOMS®–24P and –52B cask design and added it to the list of NRC-approved cask designs in § 72.214 as Certificate of Compliance Number (CoC No.) 1004. Amendments No. 3 and 6 added the –61BT DSC and the –24PHB DSC, respectively, to the system.

Discussion

On June 29, 2001, the certificate holder (TN) submitted an application to the NRC to amend CoC No. 1004 to add another dry shielded canister, designated NUHOMS®–32PT DSC, to the authorized contents of the Standardized NUHOMS®–24P, –52B, –61BT, and –24PHB cask system. This canister is designed to accommodate 32 PWR assemblies with or without Burnable Poison Rod Assemblies. It is designed for use with the existing NUHOMS® Horizontal Storage Module and NUHOMS® Transfer Cask. No other changes to the Standardized NUHOMS®–24P, –52B, –61BT, and –24PHB cask system were requested in this application. The NRC staff performed a detailed safety evaluation of the proposed CoC amendment request and found that an acceptable safety margin is maintained. In addition, the NRC staff has determined that there is still reasonable assurance that public health and safety and the environment will be adequately protected.

This rule revises the Standardized NUHOMS®–24P, –52B, –61BT, and –24PHB cask system listing in § 72.214 by adding Amendment No. 5 to CoC No. 1004. The particular Technical Specifications (TS) which are changed are identified in the NRC staff's Safety Evaluation Report (SER) for Amendment No. 5.

The NRC published a direct final rule (68 FR 49683; August 19, 2003) and the companion proposed rule (68 FR 49726) in the **Federal Register** to revise the TN Standardized NUHOMS®–24P, –52B, –61BT, and –24PHB cask system listing in 10 CFR 72.214 to include Amendment 5 to the CoC. The comment period ended on September 18, 2003. One comment letter was received on the proposed rule. The comments were considered to be significant and adverse and warranted withdrawal of the direct final rule. A notice of withdrawal was published in the **Federal Register** on October 30, 2003; 68 FR 61734.

The NRC finds that the amended TN Standardized NUHOMS®–24P, –52B, –61BT, and –24PHB cask system, as designed and when fabricated and used in accordance with the conditions specified in its CoC, meets the requirements of Part 72. Thus, use of the amended TN Standardized NUHOMS®–24P, –52B, –61BT, and –24PHB 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 TN Standardized NUHOMS®–24P, –52B, –61BT, –24PHB, and –32PT 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 January 7, 2004. Single copies of the CoC and SER are available for public inspection and/or copying for a fee at the NRC Public Document Room, 11555 Rockville Pike, Rockville, MD. Copies of the public comments are available for review in the NRC Public Document Room, 11555 Rockville Pike, Rockville, MD.

Summary of Public Comments on the Proposed Rule

The NRC received one comment letter on the proposed rule. A copy of the comment letter is available for review in the NRC Public Document Room. The NRC's responses to the issues raised by the commenter follow. As stated in the proposed rule (68 FR 49726; August 19, 2003), the NRC considered this amendment to be a noncontroversial and routine action. Therefore, the NRC published a direct final rule (68 FR 49683; August 19, 2003) concurrent with the proposed rule (68 FR 49683; August 19, 2003). The NRC indicated that if it received a ''significant adverse comment'' on the proposed rule, the NRC would publish a document withdrawing the direct final rule and subsequently publish a final rule that addressed comments made on the proposed rule. The NRC believes some of the issues raised by the commenter were ''significant adverse comments.'' Therefore, the NRC published a notice withdrawing the direct final rule (68 FR 61734; October 30, 2003). This subsequent final rule addresses the issues raised by the commenter that were within the scope of the proposed rule.

Comments on Amendment 5 to the TN Standardized NUHOMS®–24P, –52B, –61BT, –24PHB, and –32PT Cask System

The commenter provided specific comments on the Technical Specifications, the SER, and the Final Safety Analysis Report (FSAR). None of these documents were changed as a result of public comments. A review of the comments and the NRC's responses follows:

Comment 1: The commenter stated that TS 1.1.1 set the limits of 0.17g vertical and 0.25g horizontal on seismic accelerations and identified these limits as site-specific parameters. The commenter also stated that the SER was equally ambiguous in paragraph 3.1.2.1.7. The commenter recommended that the TS be corrected to state unequivocally that 0.25g and 0.17g are, respectively, the maximum permitted values of the peak horizontal and vertical accelerations at the NUHOMS/Independent Fuel Storage Installation (ISFSI) pad interface.

To support this recommendation, the commenter referred to an inspection of the FSAR which revealed that 0.25g and 0.17g are applied as peak horizontal and vertical ground accelerations on the NUHOMS system. The commenter stated that it is common knowledge in geomechanics that the free field accelerations at the site can be magnified considerably on the pad due to soil-structure interaction effects. The commenter added that TN's analysis of NUHOMS assumes that 0.25g and 0.17g horizontal and vertical accelerations are applied on the horizontal storage module (HSM) basemat; thus, these are the limiting values of on-the-pad accelerations, not ''site parameters'' as noted in the TS.

Response: Page A–1 of the Technical Specifications states the following. ''* * * site specific parameters and analyses, identified in the SER, that will need verification by the system user, are, as a minimum, as follows: * * *''. Item 3, in that listing, states: ''The horizontal and vertical seismic acceleration levels of 0.25g and 0.17g, respectively.''

The commenter indicates that the SER is ambiguous in addressing when the site-specific seismic parameters are to be taken as design values. In quoting Section 3.1.2.1.7 of the SER, the commenter did not include the second sentence of the SER paragraph. That second sentence of the paragraph states that: ''The location of these accelerations is taken at the top of the concrete pad/basemat of the HSM.'' What the actual values are is a function of the site which includes the ground accelerations and soil structure interaction effects.

No additional clarification is necessary in the Technical Specifications.

Comment 2: The commenter quoted a portion of § 72.130 which mandates that the ISFSI must be designed for decommissioning, particularly it must be designed ''to facilitate the removal of radioactive wastes * * *''.

The commenter stated that, based on the information presented in the FSARs and NRC's SER, one cannot conclude with reasonable confidence that the loaded –32PT dry shielded canisters will be able to be removed by the hydraulic ram after the NUHOMS modules have been on the storage pad for their licensed life (20 years).

To support this view, the commenter presented two main technical reasons for pessimism with regard to the removal of the loaded DSCs after 20 years of storage; namely, potential for long-term settlement of the pad and weathering (corrosion) of the DSC/rail interface under extended exposure (20 years) to the elements.

With respect to long-term settlement, the commenter noted that TS 1.2.9 stipulates that the transfer ''cask must be aligned with respect to the horizontal storage module (HSM) so that the longitudinal centerline of the DSC in the transfer cask is within ± 1⁄8 inch of its true position when the cask is docked with the HSM front access opening.'' Further, this requirement, imposed to enable the DSC to be moved horizontally, is tedious but doable during initial loading. However, calculations performed for typical storage pads loaded with heavy casks show that the long-term differential settlement from soil creep can be several inches over 20 years. The commenter stated that NUHOMS's FSAR makes no special demands on the soil strength to limit long-term settlement of the pad. The commenter further stated that there are no specific strength limits applied on the NUHOMS pad either which, along with the absence of a mandated hard subgrade, would likely lead to several inches of differential settlement of the pad over 20 years of storage, and the user's ability to maintain the alignment specified in TS 1.2.9 will be lost. The commenter claimed that the DSC will be in an irremovable state, in direct violation of § 72.130.

Response: As stated in Section 1.3.1.2 of the FSAR, ''The HSMs are constructed on a load bearing foundation which consists of a reinforced concrete basemat on compacted engineered fill.'' The general licensee is responsible for the design and construction of the HSM load bearing foundations. If a properly designed and constructed foundation system is completed for the basemat, several inches of hypothesized differential settlement should not develop. If differential settlement of a limited magnitude were to develop, the transport trailer is equipped with hydraulic jacks/positioners and an alignment system identified as the support skid positioning system that is normally used for the alignment of the transfer cask. This same system can be used to accommodate effects resulting from limited differential settlement between the basemat and the approach slab. If a situation were to develop where the support skid positioning system could not accommodate the differential settlement, the approach slab can be modified or other measures can be taken. See the following response on corrosion and environment.

Comment 3: The commenter stated that, under the general CoC authority, the NUHOMS system can be installed at any site in the U.S., including coastal sites and marine environments. The potential for surface corrosion, including pitting the DSC and HSM rail surfaces under the ambient environmental conditions and its effect on the removability of the DSC, has not been considered in NUHOMS's August 2000 FSAR for the Standardized NUHOMS System or NRC's SER. This is in violation of § 72.236(m).

Response: The potential for surface corrosion (i.e., pitting corrosion) under the ambient environmental condition and its effect on the retrievability of the DSC has been considered by the selection of corrosion resistant materials. The DSC shell structure is fabricated from ASME SA 240, Type 304 stainless steel. Type 304 stainless steel has excellent corrosion resistance in a wide range of atmospheric environments and many corrosive media. The corrosion resistance is provided by the 18 percent minimum chromium content. The material used as the sliding surface of the DSC is a high-hardness stainless steel plate (Nitronic 60). The Nitronic 60 has similar corrosion resistance as Type 304 stainless steel. This plate is mounted on the HSM rails as shown in Drawing No. NUH–03–6016–SAR contained in FSAR, Appendix E. The surface of the Nitronic 60 is lubricated to minimize friction. Additionally, both the DSC and the DSC support structure are housed inside of the HSM reinforced concrete structure which protects it from direct exposure to the weather. Therefore, staff concludes that none of the DSC and HSM rail materials are expected to degrade or react with each other. Further, staff concludes that the NUHOMS design considers the effects of environmental conditions and retrievability and meets the requirements of 10 CFR 72.236(m).

Comment 4: The commenter claimed that the maximum allowable hydraulic push and pull forces specified in the FSAR are not equal. The commenter stated that the push force is 80 kilopounds (kips); the permitted pull force is only 60 kips. The commenter further stated that it is during the removal of the DSC, when the DSC must be dragged over the corroded HSM rails, that the risk of failure to remove the canister lies. Yet, the allowable pull for the DSC extraction condition is 25 percent less than the available push force during initial insertion. Further, the coefficient of friction during DSC push assumed in the FSAR to be 0.2 is unrealistically low for weathered sliding surfaces.

Response: The commenter is in error in stating that the maximum allowed extraction force for the removal of the DSC from the HSM is 60 kips. It is 60 kips under normal loading and 80 kips for off-normal loadings which is equal to the off-normal insertion loading (FSAR Table 3.2–1 and SER Section 3.1.2.1.2). The permitted loads for insertion and extraction are the same, but there is a difference in the permitted stress allowables. As stated on page 3.1–6 of the FSAR, the hydraulic ram used to exert the insertion or extraction force is sized assuming a coefficient of friction of 1.0.

Comment 5: The commenter noted that, in the FSAR, there was no stress analysis of the DSC bottom cover plate that is being pulled by the hydraulic ram against friction, in conjunction with the internal pressure present in the canister. The commenter stated that internal pressure and the hydraulic ram pull force act in concert to maximize the stress level in the cover plate and its junction with the DSC shell. The commenter believed that neglect of analysis of this scenario leaves the structural adequacy of the bottom outer lid open to question.

Response: Table 8.2–24 of Revision 5 of the FSAR shows that an analysis of the DSC was done for accident unloading conditions that assumed the full force of the ram (80 kips) and an internal pressure of 60 psi. The analysis showed that this situation was bounded by the 75g side drop load at Service Level D. Tables M.2–15 and M.3.7–10 show the same situation for the NUHOMS®–32PT system with the new internal design pressure of 105 psi. Sections 3.1.2.2 and 3.3.2 of the SER address these tables.

Comment 6: The commenter discussed the process of inserting a DSC in the HSM and noted that this requires careful alignment of large fabricated components in open air and that the time duration for such activities can be long. The commenter stated that the NRC imposes seismic requirements on canister transfer outside of Part 50 structures even in vertical operations (see NAC–UMS or HI–STORM FSAR, for example). Yet, for the more tedious horizontal insertion process in NUHOMS, there is no treatment of a concurrent seismic event or even tornado-borne missiles during DSC transfer operations. The commenter stated that this violates a provision in § 72.122(b)(2)(1) which requires that structures, systems, and components must be able to withstand the effects of natural phenomena such as earthquakes.

Response: The FSAR amendment in Section M.3.7.3.6 states that the effects of a seismic event occurring when a loaded DSC is resting inside the transfer cask (TC) have been analyzed. Reference is made to the fact that the conditions for the 32PT are bounded by the conditions used for the 24P analyses described in the original FSAR. The referenced section, Section 8.2.3.2(D), indicates that all conditions existing during loading or transport operations are enveloped by two loading cases that are described in the FSAR, one of which envelops and applies to this condition. TN has performed a stability analysis that shows there is a safety factor of at least 2.0 against overturning the cask/trailer assembly during a seismic event in this bounding case. During the cask transfer operation, the cask/trailer unit is attached to the HSM by the cask restraint devices that are anchored into the front of the HSM and are attached to the trunnions of the TC as shown in FSAR Figure 4.2–13. These restraints are designed for accident conditions and envelop seismic loads. The TC and the HSM are designed for tornado missiles as described in Section 3.2.1 of the FSAR, Revision 5. The NUHOMS system is designed to withstand seismic conditions as well as those produced by tornado-borne missiles.

Comment 7: The commenter stated that the 32PT DSC is the heaviest canister proposed for use thus far in the HSM. The commenter noted that NUHOMS's FSAR asserts that the DSC support structure is braced, presumably to incorporate seismic resistance. A review of the sketches provided in the FSAR showed no bracing. The commenter provided marked up pages from NUHOMS's FSAR for the Standardized NUHOMS System to indicate the missing braces. The commenter stated that, without the braces, the DSC support structure in the HSM is weak against axial or lateral overturning moments, especially the increased g-loads that will accompany the heavier 32PT DSC.

Response: The commenter is correct in stating that the 32PT DSC is the heaviest canister to date proposed for use in the NUHOMS Storage System. As stated by Transnuclear, Inc., on page 1.1–2 of the proposed FSAR revision for Amendment 5, the HSM has been qualified for a DSC weight of 102,000 pounds that envelops the 101,380 pounds for the 32PT in the storage configuration. As stated on page M.1–1 of Amendment 5, there is no change to the HSM required for the 32PT component for the NUHOMS system.

As shown in the FSAR, Revision 5, the DSC is supported on two rails that are supported by a structural steel frame in the cavity of the HSM. The frame structure is anchored to the reinforced concrete floor slab, the side walls, and the front wall. Figures 4.2–6 and 4.2–7 illustrate the longitudinal and transverse sections of the HSM with the DSC support structure inside. Figures 4.2–8 and 4.2–9 provide additional details of the DSC support structure. These drawings show that the structural steel frame is a braced frame in both the transverse and longitudinal directions. A braced frame does not have to be additionally braced with diagonal bracing. Each planar frame or bent of the three dimensional structural frame is braced or restrained from transverse lateral movement, in the plane of the frame or bent, at the top by a structural steel channel section that acts as a strut or tie to the reinforced concrete wall of the HSM. In the longitudinal direction, the entire three-dimensional structural frame is braced through the rail extension plate and base plate that are anchored to reinforced concrete of the throat of the opening of the HSM. Figure 8.1–20 of the FSAR, Revision 5, presents the DSC structural support analytical model showing that this three dimensional (space) frame is considered to be a braced frame. It should be noted that there is another NUHOMS storage system, the Advanced NUHOMS Storage System, that has different features and was developed for higher seismic application areas.

The DSC support structure inside the HSM is adequate for the specified input values to show conformance with § 72.236.

Comment 8: The commenter stated that the consideration of the tornado-borne missile in the FSAR for the Standardized NUHOMS System is oblivious to the real vulnerability of the HSM. The commenter further stated that the entire 3-foot thick top roof is held by a mere 4 anchors about 1½ inches in diameter, and the concrete-filled front door (over 7,000 pounds in weight) is not even held by bolts (rather by 3 straps). The commenter asserted that the FSAR for the Standardized NUHOMS System provides no analysis of the integrity of these weak locations in the HSM under natural environmental phenomena loads.

Response: Although the roof is held to the base by eight 1¼-inch steel bolts and the roof attachment angle assembly which would resist a significant lateral force, these are not the design features provided to resist roof lateral loads and other accident loads. There is a 4-inch key or ledge of concrete which sits in the base that is designed to resist lateral loads of the roof. Downward vertical loads are resisted by shear and bending of the roof with the downward loads carried out at the periphery in bearing to the base unit walls. The key detail can be seen in drawing NUH–03–6015, Rev. 5, Sheet 1 of 2.

Contrary to the assertion of the commenter, the HSM door is held on by bolts, not straps. Analyses of the HSM and the HSM door are presented in FSAR Sections 8.2.2 and 8.2.3 for tornado and seismic conditions. These analyses show that the entire HSM has been qualified for its design basis tornado and wind loads.

The HSM structure is adequately designed to resist the tornado and seismic loading conditions as required by § 72.236.

Comment 9: The commenter stated that how the structural features will resist a larger impact such as a plane should be a matter of concern to the agency in the after-9/11 world.

Response: The Commission believes that the best approach to dealing with threats from aircraft is through strengthening airport and airline security measures. Consequently, we continue to work closely with the appropriate Federal agencies to enhance aviation security and thereby the security of nuclear power plants and other NRC-licensed facilities. Shortly after the September 11, 2001, attacks, the NRC, working with representatives of the Federal Aviation Administration (FAA) and Department of Defense (DOD), determined that a Notice To Airman (NOTAM), issued by the FAA, was the appropriate vehicle to protect the airspace above sensitive sites. This NOTAM strongly urged pilots to not circle or loiter over the following sites: Nuclear/Electrical power plants, power distribution stations, dams, reservoirs, refineries, or military installations, or expect to be interviewed by law enforcement personnel. Further, the NRC issued orders imposing additional physical protection measures for independent spent fuel storage installations using dry storage.

The NRC is conducting a comprehensive evaluation that includes consideration of potential consequences of terrorist attacks using various explosives or other terrorist techniques on dry storage casks. As part of this evaluation, the agency is looking at the structural integrity of dry storage cask systems and will consider the need for additional design requirements to enhance licensee security and public safety.

Comment 10: The commenter noted that, according to the FSARs, the –32PT DSC has purportedly been analyzed for a drop from 80 inches onto an unyielding surface with the added assumption that the transfer cask is rigid. This event is postulated to account for a potential drop of the loaded DSC in the transfer cask during its handling on the basemat. The calculations to compute the g-load, however, use an antiquated method that was determined to be unconservative by the NRC in the mid-1990s.

The commenter stated that, in 1997, the NRC established the acceptable method for reliably and conservatively predicting the g-load in a paper titled ''NRC Staff Technical Approach for Spent Fuel Storage Cask Drop and Tipover Accident Analysis.'' The commenter believed that the method relied on in the FSAR is unconservative and that a much higher value than 75g's will develop if the NUHOMS®–32PT DSC undergoes a free fall of 80 inches on a rigid surface without the benefits of an impact limiter.

Response: The commenter's reference to ''the NRC paper sets down the acceptable method for reliably and conservatively predicting the g-load'' has apparently been misinterpreted to mean that this is the only acceptable method for calculating the impact loads. The referenced paper, in its title, uses the words ''technical approach'' that is intended to imply that the methodology therein is acceptable to the NRC, but that does not mean that it is the only acceptable methodology that could be utilized. Analysis of drops from heights of up to 80 inches were chosen because they were representative of the worst case drops that might be found at an ISFSI, or along the transfer route. There was no assumption that the impacted surface was essentially unyielding or rigid. The methodology adopted by TN considered the stiffness of the impacted surface. As noted on page 3–19 of the NRC staff Safety Evaluation Report dated December 1994 for the Standardized NUHOMS Horizontal Modular Storage System for Irradiated Nuclear Fuel, the NRC staff independently completed calculations to verify that the design deceleration values were conservative.

Comment 11: The commenter stated that TS 1.2.13 permits lift heights of up to 80 inches in cold conditions based on nil ductility transition (NDT) temperature considerations of the transfer cask's materials. The commenter further stated that the underlying documents [Safety Analysis Report (SAR) or SER] do not address the top and bottom shield plugs that are very thick (over 6 inches) and made of a steel that is low-temperature incompetent (A–36). The commenter believed that at –20 F, the A–36 plugs will suffer extensive fracture under a 75-g impact load, perhaps even pulverization.

Response: The shield plugs are fabricated from American Society for Testing and Materials (ASTM) A36 steel, a commonly used steel for structural applications. ASTM A36 was selected because of its high strength and metallurgical stability. However, if this material should experience temperatures below −20°F, its ductility (or fracture toughness) and its ability to be used for structural applications may be insufficient and, thereby, lead to potential fracture of the material. To address this issue, the user is constrained by the TS to ensure that fracture (pulverization, as characterized by the comment) does not occur. TS 1.2.13 prescribes the following limits: (1) No lifts or handling of the TC/DSC at any height are permissible at DSC temperatures below −20°F inside the spent fuel pool building; (2) the maximum lift height of the TC/DSC shall be 80 inches if the basket temperature is below 0°F, but higher than −20°F inside the spent fuel pool building; and (3) the maximum lift height and handling height for all transfer operations outside of the spent fuel pool building shall be 80 inches, and the basket temperature may not be lower than 0°F. Therefore, staff has concluded that the ASTM A36 carbon steel has sufficient fracture toughness (material properties) to remain functional, when operated under the limitations set forth in the TS.

Comment 12: The commenter stated that he was greatly concerned about the clear absence of critical structural welds in the fuel basket in the –32PT DSC. The commenter manually circled areas in the drawing details released to the public that show absence of welds in the fuel basket at critical load transfer locations under a horizontal drop condition.

Response: The commenter is correct in that welds are not shown in the drawing that was marked up and attached to the comments. However, this drawing is not intended to show the weld location and types because this information is contained in proprietary drawing NUH–32PT–1004, Rev 0, Sheet 2 of 2. All required critical locations are welded together. Section M.1.2.1 of Amendment 5 on page M.1–4 of the nonproprietary version provides a verbal description of the basket assembly. The following statement is made in that section: ''The basket structure consists of a grid assembly of welded stainless steel plates or tubes that make up a grid of 32 fuel compartments.''

Comment 13: The commenter stated that TNW's stress analysis of the basket appears to have a serious error, perhaps an erroneous assumption in the finite element model. The commenter stated that critical stress analyses figures were deleted from the nonproprietary copy and he could not offer further help.

Response: The commenter gives no information regarding any specific reference to the related NUHOMS documents and gives no indication as to the origin of the stress such as thermal, seismic, or some other loading condition with respect to the comment. It is assumed that the commenter believes that there are no welds between the various cells of the basket assembly and that the finite element analysis was conducted on a model that represented a continuum or structural integrity across the interfaces among the cells. With regard to the comment that ''critical stress analyses figures are deleted from the non-proprietary copy,'' if the commenter is referring to Figures M.3.6–1 through M.3.6–4, those figures in the proprietary version of Amendment 5 do not identify stresses. Instead, these figures provide the modeling details of the finite elements used in the analyses. The NRC staff has not identified any significant erroneous assumptions in the finite element models utilized.

Comment 14: The commenter quoted from NUREG–1536, Chapter 11, V.1, that ''an event may be analyzed for regulatory purposes even though no credible cause can be identified. Such events should be clearly identified as nonmechanistic.''

The commenter stated that NRC's regulatory practice has been to require a nonmechanistic tipover analysis of casks in long-term storage. According to the NUHOMS FSAR for the NUHOMS Standardized System, each horizontal storage module is freestanding. The height (15 feet) to width radio (9.7 feet wide) of the horizontal storage module is comparable to vertical ventilated systems (that tend to be about 18 feet high by 11 feet diameter) where NRC has always demanded a nonmechanistic tipover analysis. The commenter asked the question why the special dispensation for NUHOMS, with its top heavy structure (a 3-foot thick top roof held in place by slim anchors).

Response: The commenter states that the height to width ratio (15 feet to 9.7 feet) is comparable to vertical ventilated systems. This does not take into account the two side shield walls attached to a single HSM. This would make the limiting dimension 9.7 feet +4 feet = 13.7 feet. Therefore, the height to width ratio is not comparable to vertical ventilated systems ($^{15}/_{13}$.7= 1.09 is considerably less than $^{18}/_{11}$=1.6). The tipover analyses, however, are carried out on a single HSM unit.

The tipover of a single HSM was considered under specific loading conditions, namely the tornado effects as well as the seismic effects. The discussion on these analyses is included in the FSAR, Revision 5, in Sections 8.2.2.2.A.(i) and 8.2.3.2.B.(iii). The factors of safety are 1.38 and 1.24, respectively, against tipover. In the case of the tipover or liftoff of the 32PT DSC from the DSC support structure rails inside the HSM from a seismic event, the factor of safety is 1.20 as identified in Section M.3.7.3.1.2 of FSAR Amendment 5.

The nonmechanistic tipover analysis of a cask system is performed to ascertain that a cask that is handled, lifted, and moved will not suffer a loss of function under a tipover event. In other words, the specific cause or mechanism of that event such as a failed lifting apparatus or human error in the attachment of the lifting device is not identified as a credible cause. In the case of the NUHOMS design concept, the cask storage system that includes the DSC inside the HSM is never handled, lifted, or moved. The nonmechanistic events for this system are those considered when the DSC is in the TC as indicated in Figure 8.2–3 of the FSAR, Revision 5.

The relevant considerations have been made for the nonmechanistic tipover events.

Comment 15: The neutron absorber panels in 32PT DSC appear not to be ''fixed'' as required by § 72.124(b). ≤

Response: The neutron absorber plates are fixed in place. The plates are fixed using screws as shown on Drawing No. NUH–32PT–1003–SAR Sheet 2, Rev. 2.

Comment 16: The commenter stated that the required B–10 loading in the neutron absorber panels is minuscule, merely 0.007 gm/sq.cm., less than even 52BT for BWR fuel (which is 0.016 gm/sq.cm.), and a small fraction of that used in other casks (such as NAC–STC).

Response: The B–10 neutron absorber panels are not solely relied upon for criticality control. The minimum B–10 content of the absorber panels, along with the poison rod assemblies (PRAs) and the borated water, ensures that the 32PT canister will remain subcritical during loading and unloading operations.

Comment 17: The commenter stated that the reliance for reactivity control seems to be based on the so-called Poison Rod Assemblies (PRAs). These PRAs, vital to criticality control, are little more than stainless steel tubes filled with ''B_4C pellets'' (see PSER, Section 3.1.4.2). There are no requirements imposed on the size and integrity of the welds that will join the closure plugs to these thin-walled tubes (as little as 0.018-inch thick per Figure M.1.6–2 in the SAR).

Response: The NUHOMS SAR includes commitments to perform dimensional measurements and visual examination for both the neutron absorber plates and PRAs in Section M.9. The visual examination (per ASME or American Welding Society (AWS)) will identify any weld discontinuities (such as cracks, porosity, blisters, or foreign inclusions) on the end cap of the PRA.

Comment 18: The commenter stated that the so-called nonstructural PRA closure welds, without any regulatory requirements on their NDE, are the sole barrier against leaching out Boron Carbide from the PRAs. The commenter stated that a total reliance on the micro-seal welds to hold B_4C in place to preserve criticality safety appeared to be incredulous, considering that the PRAs will be subject to thermal stresses during fuel loading and be quite hot in long-term storage. The commenter added that there is no requirement to purge air and moisture from the PRA tubes before seal welding its contents. This means entrained air and moisture will be locked in every PRA in the stored fuel.

Response: The temperatures that the PRAs are subjected to are not hot enough to generate a significant pressure from the relative humidity inside of the tube. The NRC staff does not anticipate a loss of the seal welded end cap due to internal pressure build-up. Further, because there is no electrolyte present in the PRAs and since boron carbide is insoluble and inert, there should be no corrosion or chemical interaction between the stainless steel and the boron carbide pellets. It should be noted that if there were any defective weld discontinuities on the end cap of a PRA while the cask is inside the pool, there would be practically no leaching of boron from the defective weld on the closure plug. Boron carbide is virtually insoluble in water. See ASTM Standard Specification for Nuclear-Grade Boron Carbide Powders (C 750–03). Additionally, as stated in Section M.1.2.2.3.1 of the SAR, the PRAs are only necessary during loading and unloading operations. The NRC staff has concluded that the criticality safety is not compromised during loading and unloading operations because there is no mechanism that will cause leaching out of the boron from the PRAs.

Comment 19: The commenter stated that the 32PT DCS is in violation of § 72.236(h) which requires that the ''spent fuel storage cask must be compatible with wet and dry spent fuel loading and unloading facilities.'' To support this view, the commenter stated that the storage slots in the 32PT DSC are 8.7-inch x 8.7-inch (nominal) opening (see PSER). The FSAR for the Standardized NUHOMS System specifies ''the minimum open dimension or each fuel compartment is 8.60 inches x 8.60 inches.'' The commenter stated that, having worked for PWR Nuclear Steam Safety System (NSSS) suppliers for many years, no Westinghouse or B&W plant has fuel storage racks with 8.6-inch (min) or 8.7-inch (nom.) opening dimension. Irradiated fuel tends to bend, bow, and twist in the reactor; for this reason, PWR reactor suppliers require large storage cell openings. The 32PT DSC, with 8.6-inch (min.) opening, would be an engineered stuck fuel event.

Response: The dimensions of the fuel compartment openings are adequate to accommodate the fuel assemblies including the Westinghouse and Babcock & Wilcox types. There is no degradation mechanism that would cause an assembly already in a cask to bow, except for an accident. Therefore, if an assembly is able to be loaded into a cask, it should be able to be unloaded.

Comment 20: In a related matter to Comment 19, above, the commenter expressed deep reservation about the loose aluminum blocks (visible in FSAR Amendment 5) that are assumed to be snugly fitting. The commenter stated that the 32PT DSC will be made from a thinner shell (½-inch) (to hold a heavier basket) than prior NUHOMS DSCs (⅝-inch thick shell). This means that the shell in the 32PT DSC will ovalize more from its dead weight and from full-length butt welds. The commenter further stated that snugly fitted aluminum blocks may appear acceptable on paper, but in real hardware are impossible to manufacture, and told NRC to recall that the lack of fabricability of VSC–24 baskets (cracking of steel plates at the toe of the bend) caused the industry an untold amount of grief.

Response: The commenter referenced Figure M.3.7.3, but it is assumed to have been intended to mean Figure M.3.7–3, ''0-Degree Side Drop Stress Intensity, 32PT Basket With Aluminum Transition Rails (Support Rails at +/−18.5-Degrees),'' in making the comment that ''the loose aluminum blocks * * * that are assumed to be snugly fitting.'' Figure M.3.7–3 is a schematic representation of the transverse cross-section of a DSC that illustrates the stress levels in the materials but does not show details of the configuration. Section M.1.5 of the FSAR contains the drawings that illustrate a configuration of the aluminum transition rail sections with respect to the stainless steel plates they are attached to. Drawing NUH–32PT–1006NP–SAR, Sheet 1 of 1, illustrates that there are attachment connectors between the aluminum transition rails, the rail plates, and the basket assembly. The connectors are stainless steel studs welded to the outside of the basket assembly. The studs and the basket assembly are shown on Drawing NUH–32PT–1003NP–SAR, Sheets 1 and 2 of 2, as Detail 2. The connection configuration also provides for differential thermal movements. Therefore, the aluminum transition rails are not loose and do not rely on a snug fit for their position.

The commenter indicates that because of the reduced thickness of the cylindrical shell of the 32PT DSC and the full length butt welds, there will be increased ovalization of the DSC shell under dead loads. The implication of the comment is apparently that this increased ovalization could potentially cause the assumed snugly fitting transition rails to become even looser. The DSC was analyzed for dead loads using the ANSYS finite element models shown in Figures 8.1–14a and 8.1–14b in the FSAR. One loading condition considers the fuel loaded DSC in a horizontal position with the dead loads. The fuel-loaded portions of the basket assembly bear on transition rails that then bear on the inner shell of the DSC. Figures M.3.6–3 and M.3.6–4 illustrate the model used with the shell and the basket for a typical support condition of the loaded DSC. Such a model is then analyzed to determine the primary membrane and membrane plus bending stresses as well as for the primary plus secondary stresses. Deformed shapes are also obtained from such analyses.

Figure M.3.6–12 illustrates the stress intensities in the DSC shell and the aluminum transition rails under the dead load of the spent fuel inside the basket assembly as supported in an HSM. This is considered a normal loading condition, and the appropriate stress allowables are 17,500 psi for primary membrane stress, 26,300 psi for membrane plus bending stresses, and 54,300 psi for primary plus secondary stresses. This particular loading condition produces very low stress intensities in the shell material that are 2,650 psi, 6,000 psi, and 7,000 psi, respectively, as identified by stress type above, as shown in Table M.3.6–2. With the worst case thermal effects that can be present under these normal conditions, combined with the dead load, the stress for the primary plus secondary stresses increases to 44,550 psi, still less than the 54,300 psi allowable. Figures M.3.6–12 and M.3.6–13 illustrate the results of the analyses.

With these stress levels that show that the material remains in the elastic behavior range, deformations will remain elastic. Specific comparisons of elastic deformations between a 0.625-inch shell thickness and a 0.500-inch shell thickness under dead load conditions have not been made by the NRC. It is correct that there would be more ovalization with a thinner shell; however, the incremental change has no apparent impact on the capability of the DSC to perform its intended storage function cradled on the pair of support rails within the HSM. The effects of longitudinal butt welds in the cylindrical shell on the tendency of the shell to become oval have been considered and have been determined to be of no safety consequence.

The commenter states that snugly fitting aluminum blocks that are the transition rails will be impossible to manufacture. This comment is assumed to have been related to the difficulty that could arise if the positions of the aluminum transition rails were to rely on a ''snug fit.'' As noted above, the transition rails are positioned controlled via studs attached to the basket assembly. The NRC has no information that would indicate that the solid aluminum transition rails cannot be manufactured by current machining practices to the necessary dimensions and tolerances.

Comment 21: The commenter stated that he was surprised to learn from the supplier's FSAR that a loaded 32PT DSC canister will have no provision to be lifted on its own and must be lifted by the TC. The commenter also stated that if the DSC were to be separated from the TC under an accident event, there would be no means to lift and handle the canister. The commenter considered the lack of ability to separately handle a loaded canister to be a severe weakness that violates the notion of retrievability under § 72.122(l).

Response: Retrievability, with regard to certificates of compliance for spent fuel storage casks, is addressed in § 72.236(m), which states: ''To the extent practicable in the design of the storage casks, consideration should be given to compatibility with removal of the stored spent fuel from the reactor site, transportation, and ultimate disposition by the Department of Energy.'' This refers to retrieval of the fuel assemblies from the canister. This design meets this requirement. The canister is able to be handled and placed into the transfer cask before loading of assemblies. The canister is then handled as one piece with the transfer cask until it is placed within the storage module. There are no postulated accidents when the canister is inadvertently separated from the transfer cask.

Comment 22: The commenter referred to Section 1.2.24 of the TS which states: ''* * * for the NUHOMS–32PT system, the fuel cladding limits are based on Interim Staff Guidance (ISG)–11, Revision 2.'' The commenter disagreed and quoted from page 2 of ISG–11, Rev. 2: ''Accordingly, the materials reviewer should coordinate with the thermal reviewer to assure that the maximum calculated temperatures for normal conditions of storage, and for short-term operations including cask drying and backfilling, do not exceed 400°C (752°F).''

The commenter noted that in direct violation of the above requirement, the Amendment 5 FSAR states in Section 4.1: ''During short-term conditions, the fuel temperature limit is 570°C.''

The commenter further stated that calculated temperature values in Table M4.2 indicate that the ISG–11, Rev. 2, limit is exceeded by wide margins under short-term normal conditions.

Response: The comment is based on an older version of Amendment 5 to FSAR CoC 1004 (Rev. 0, June 2001). The correct version of the SAR corresponds to the following reference: Transnuclear West, Amendment No. 5 to NUHOMS CoC 1004, Addition of 32PT DSC to Standardized NUHOMS System, Rev. 4, January 2003, which complies with ISG–11, Rev. 2.

Comment 23: The commenter stated that use of durable materials that are proven for their intended function must be a basic plank of dry storage system design, and a mandated fact under § 72.122(a), (b), and (c). One objection raised by the commenter to the materials being proposed for the 32PT DSC was that the shield plugs at the two ends of the DSC are made from one of the cheapest carbon steels available (A–36). The commenter noted that the lower plug (along with air) is permanently sandwiched between the two stainless plates. This plug will expand and contract under heat, as will the entrained air in the space, constantly stressing the welds that confine the plug. Thermal differential expansion between carbon and stainless steel will further increase stresses in those same welds. The commenter asked why the plugs could not be made of machined stainless steel, which would eliminate material incompatibility, remove most entrained air, and remove long-term concerns.

Response: The material used for the shield plug is appropriate based on the following: First, the shield plugs are fabricated from ASTM A–36 steel, a commonly used steel for structural applications. Second, brittle fracture of the carbon steel is not expected because the ductile-to-brittle transition temperature is below the expected operating temperatures. Third, the shield plugs are also plated with electroless nickel in response to NRC Bulletin 96–04 to ensure that a chemical reaction does not occur. This coating is not expected to react with the spent fuel pool water to produce unsafe levels of flammable gas. Fourth, there are small radial clearances provided between the carbon steel bottom shield plug and the stainless steel DSC shell. Fifth, Table M.3.3–1, ASME Code Materials Data for SA–240 Type Stainless Steel, and Table M.3.3–2, Materials Data for ASTM A–36 Steel, show that the thermal coefficient of expansion is of the same order of magnitude between 100 to 800°F. Sixth, the residence time of a plug in water is limited to cask loading operations and then vacuum dried. Therefore, any degradation would be minimal. The NRC staff concludes that these material properties are acceptable and appropriate for the expected load conditions (*e.g.*, hot or cold temperature, wet or dry conditions) during the license period and in accordance with regulatory requirements.

Comment 24: Related to Comment 23, above, another objection raised by the commenter with respect to the materials being proposed for the 32PT DSC was the neutron absorber. The commenter was not able to locate any specificity on the brands of neutron absorbers permitted by the CoC. The commenter stated that neutron absorbers use aluminum, which is a most reactive material, and stated that NRC has been wise in controlling the specific make of neutron absorbers that are permitted to be used and felt that this caution is well placed, considering the 1996 hydrogen ignition event in SNC's product. Referring to a section in the PSER that stated that purging of the canister during lid welding is not required, the commenter disagreed and stated that it is unsafe to make purging elective if aluminum-based neutron absorber coated carbon steels are present in the canister. He referred to the lesson learned from the Columbia Generating Station experience.

The commenter recommended that the CoC specify the acceptable neutron absorbers to ensure compliance with the above-cited regulation and not let a CoC holder make the choice of neutron absorber unilaterally.

Response: Technical Specification Table 1–1h imposes requirements on neutron absorbers materials for the boron.

The NRC staff is aware of a slight potential for chemical or galvanic reaction between the aluminum and stainless steel in contact with borated water spent fuel pools. This reaction may produce small amounts of hydrogen, during loading and unloading operations. Further, the NRC staff is aware of hydrogen being generated from prepassivated Boral. This reaction may also produce small amounts of hydrogen, during loading and unloading operations. As stated in M.3.4 of the SAR, small amounts of hydrogen could be produced during loading and unloading operations. The applicant's analysis showed that a hydrogen concentration of 2.39 percent can be generated. However, the NRC staff recognizes that this amount of hydrogen is below the ignition limit of 4 percent. However, to address the potential hazards associated with hydrogen gas, the applicant employs mitigation actions contained in the generic procedures of SAR Sections M.8.1.3 and M.3.4. These sections state that if hydrogen gas is detected at concentrations above 2.4 percent in air at anytime before or during welding operations, the hydrogen gas will be removed by purging the suspect regions with an inert gas. The NRC staff concluded during this review that the guidance in the generic procedures is adequate to prevent formation of any hydrogen gas that may be generated during welding operations. Hence, the potential reaction of the aluminum with the spent fuel pool water will be minimized and not impact the efficacy of the poison material.

Neutron absorber materials such as Metamic and BorAlyn have undergone qualification testing. The qualification testing included an evaluation for hydrogen generation. The qualification test program was reviewed and approved by the NRC for these two materials.

Finally, any neutron absorbers used inside of an approved cask design must have been shown through qualification testing to be effective and durable during the license period. The tests and data are usually submitted along with the license application and are subject to review and questioning by the NRC staff. After the absorber material has been approved at a particular level of B–10 credit by the NRC, the SER discusses the technical basis for approval. It should be noted that the licensee may potentially use any neutron absorber material at that approved level of B–10 credit in its cask provided it meets the requirements in § 72.48. Therefore, there is no reason to reference the manufacturer/brand name of the neutron absorber in the CoC.

Comment 25: Referring to paragraph M.4.6.3 of the FSAR for Amendment 5, the commenter concluded that a fire event in the vicinity of the HSM was ruled out. The commenter stated that this inference is also supported by the text matter in the FSAR for the Standardized NUHOMS® System. The commenter believed that the FSAR statements ruling out fire around the HSM are erroneous because the hydraulic fluid in the ram and the fuel in the heavy-haul trailer are credible sources of fire for a previously loaded HSM located in the vicinity of the HSM being loaded.

The commenter stated that the a priori exclusion of fire analysis at the HSM is inconsistent with NRC's previous certification reviews of other ventilation systems and that it is also unsafe.

Response: The fire event associated with the loading operations and storage within the HSM (including fires in the vicinity of the HSM) is bounded by the analyzed transfer cask fire event. The transfer cask fire analysis was based on very conservative assumptions. Other site-specific fires have to be addressed by the system user planning to use the NUHOMS®–32PT storage cask, as part of the § 72.212 evaluations.

Comment 26: The commenter referred to Section M.3.1.2.1 of the FSAR for Amendment 5 which states that the inner bottom cover plate-to-shell joint is subjected to volumetric and liquid penetrant examination as required by Subsection NB of Section III of the ASME Code. The commenter stated that examination of this weld cannot be radiographed or ultrasonically tested by virtue of its geometry.

Response: The examination of the full penetration weld corner joint used on the inner bottom cover plate-to-shell weld is specifically addressed in paragraph NB–5231(c) of the ASME Boiler and Pressure Vessel Code Section III, Subsection NB. The geometry of the weld in question is in accordance with Figure NB–4243–1(f). As stated by TN, the weld geometry of Figure NB–4243–1(f) is able to be successfully examined ultrasonically in conformance with the ASME Code requirements.

Comment 27: The commenter states that Section 4.8 of the SER accepts sudden quenching of irradiated fuel at 678°F in water during reflooding operation. The commenter stated that quenching would cause a sudden cooling of the fuel, and the 117°F temperature limit would undoubtedly be exceeded, a restriction imposed by ISG–11, Rev. 2, presumably to protect semibrittle irradiated fuel from thermal shock. The commenter urged the NRC to reconsider this unnecessary regulatory leniency.

Response: Section 4.8 of the SER states that the maximum cladding temperature reached during vacuum drying after approximately 33 hours is 678°F (358.88°C). This is below the maximum limit of 752°F (400°C) per ISG–11. The maximum temperature difference for the fuel cladding during drying and backfilling operations is 100°F (55.55°C). This meets the thermal cycling criteria specified by ISG–11, which states that the temperature differences greater than 117°F (65°C) should not be permitted. The maximum fuel cladding temperature during cask reflood operations will be significantly less than the vacuum drying condition because of the presence of water and/or steam in the DSC cavity.

Comment 28: Referencing Section 3.7 in the Amendment 5 FSAR, the commenter stated that the consideration of flood in the FSAR is merely to treat it as a source of hydrostatic load. The commenter believed that a low elevation flood that submerges the bottom duct is far more dangerous. He stated that a partially submerged HSM, heated by the DCS through radiation and convection and chilled by the rising floodwaters, will cause severe thermal stresses in its reinforced concrete structure. The commenter further stated that because the HSM's walls are both structural members and biological shield, a thru-thickness crack from large thermal strains induced by a short-duration flash flood will be unacceptable for public health and safety. The commenter stated that there is no consideration of this scenario in the supporting licensing material provided by TNW and added that it calls for a careful analysis.

Response: As stated in the FSAR, Revision 5, Section 8.2.4, recovery from flooding events has been addressed, and the case of completely blocked inlet and outlet vents has been addressed in Section M.4.6.1 of proposed Amendment 5. The blocked vent condition is assumed to be superimposed concurrently with the extreme off-normal ambient thermal condition of 117°F with insolation. Under these conservative design conditions, there is a 40-hour period at minimum, that must elapse before there are thermal conditions arising that would approach design limits. The Technical Specifications in Attachment A of the CoC on page A–57 address the fact that there is daily (every 24 hours) visual surveillance required of the exterior of the vents as well as a close-up inspection performed to see that there are no vent blockages. If blockage is found, action must be taken to clear the vent(s) within the 40-hour time period because, as shown in Figure 8.2–16, the concrete temperature limit of 350°F will be reached in the concrete roof structure of the HSM.

Additionally, in the situation when only the bottom vent is blocked, the water would begin to evaporate from the heat load. This would provide evaporative cooling to the DSC and the upper volume of the HSM. Such a situation would be bounded by the analysis of blocked circulation vents with ambient temperatures at their extremes (−40°F and 117°F) as noted above. In these situations, the maximum temperature gradients experienced by the HSM are 102°F and 99°F, respectively, as shown in Table 8.1–17 of the FSAR.

Comment 29: The commenter stated he was surprised and disappointed that the CoC uses a product designation name like ''–32PT,'' where the ''T'' stands for transportable; and uses the words, ''* * * and T is to designate that the DSC is intended for transportation in a 10 CFR 71 approved package,'' when this CoC pertains only to storage. The commenter stated that from personal experience, foreign utilities in particular do not always recognize the distinction. The commenter questioned the purpose for using this designation or making this statement.

Response: The use of the term ''transportable'' in the SER, SAR, or CoC is descriptive of the intended function. The use of this terminology in a dry storage cask application or an NRC SER/CoC does not represent a certification under 10 CFR Part 71 for the transport of radioactive materials. This CoC does not authorize transportation under Part 71.

Summary of Final Revisions

Section 72.214 List of Approved Spent Fuel Storage Casks

Certificate No. 1004 is revised by adding the effective date of Amendment Number 5 and adding Model Number NUHOMS®–32PT.

Good Cause To Dispense With Deferred Effective Date Requirement

The NRC finds that good cause exists to waive the 30-day deferred effective date provisions of the Administrative Procedure Act (5 U.S.C. 553(d)). The primary purpose of the delayed effective date requirement is to give affected persons; *e.g.*, licensees, a reasonable time to prepare to comply with or take other action with respect to the rule. In this case, the rule does not require any action to be taken by licensees. The regulation allows, but does not require, use of the amended TN Standardized NUHOMS®–24P, –52B, –61BT, and –24PHB cask system for the storage of spent nuclear fuel. The TN Standardized NUHOMS®–24P, –52B, –61BT, and –24PHB cask system, amended to include the new dry shielded canister designated –32PT, meets the requirements of 10 CFR Part 72 and is ready to be used. A general licensee has made plans to load the NUHOMS®–32PT casks in January 2004 to preserve full core off-load capability at its site. The general licensee is currently in a refueling outage and needs to load fuel into the storage casks once done. The amended TN Standardized NUHOMS®–24P, –52B, –61BT, and –24PHB cask system, as approved by the NRC, will continue to provide adequate protection of public health and safety and the environment.

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.

Voluntary Consensus Standards

The National Technology Transfer Act of 1995 (Pub. L. 104–113) requires that Federal agencies use technical standards that are developed or adopted by voluntary consensus standards bodies unless the use of such a standard is inconsistent with applicable law or otherwise impractical. In this final rule, the NRC is revising the Standardized NUHOMS®–24P, –52B, –61BT, and –24PHB cask system design listed in § 72.214 (List of NRC-approved spent fuel storage cask designs). This action does not constitute the establishment of a standard that establishes generally applicable requirements.

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 is not a major Federal action significantly affecting the quality of the human environment and, therefore, an environmental impact statement is not required. This final rule amends the CoC for the TN Standardized NUHOMS®–24P, –52B, –61BT, and –24PHB cask system within the list of approved spent fuel storage casks that power reactor licensees can use to store spent fuel at reactor sites under a general license. The amendment modifies the present cask system design to add another dry shielded canister, designated NUHOMS®–32PT DSC, to the authorized contents of the Standardized NUHOMS®–24P, –52B, –61BT, and –24PHB cask system. This canister is designed to accommodate 32 PWR assemblies with or without Burnable Poison Rod assemblies. It is designed for use with the existing NUHOMS® Horizontal Storage Module and NUHOMS® Transfer Cask. 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, One White Flint North, 11555 Rockville Pike, Room O–1F23, Rockville, MD. Single copies of the environmental assessment and finding of no significant impact are available from Jayne M. McCausland, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 415–6219, e-mail *jmm2@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.

Regulatory Analysis

On July 18, 1990 (55 FR 29181), the NRC issued an amendment to 10 CFR Part 72 to provide for the storage of spent nuclear fuel under a general license in cask designs approved by the

NRC. Any nuclear power reactor licensee can use NRC-approved cask designs to store spent nuclear fuel if it notifies the NRC in advance, spent fuel is stored under the conditions specified in the cask's CoC, and the conditions of the general license are met. A list of NRC-approved cask designs is contained in § 72.214. On December 22, 1994 (59 FR 65920), the NRC issued an amendment to Part 72 that approved the Standardized NUHOMS®–24P and –52B cask system design by adding it to the list of NRC-approved cask designs in § 72.214. Amendments No. 3 and 6 added the –61BT DSC and the –24PHB DSC, respectively, to the system. On June 29, 2001, the certificate holder, Transnuclear, Inc., submitted an application to the NRC to amend CoC No. 1004 to permit a Part 72 licensee to add another DSC, designated NUHOMS®–32PT DSC, to the authorized contents of the Standardized NUHOMS®–24P, –52B, and –61BT cask system. This canister is designed to accommodate 32 PWR assemblies with or without Burnable Poison Rod Assemblies. It is designed for use with the existing NUHOMS® Horizontal Storage Module and NUHOMS® Transfer Cask.

The alternative to this action is to withhold approval of this amended cask system design and issue an exemption to each general licensee. This alternative would cost both the NRC and the utilities more time and money because each utility would have to submit a request for an exemption, and the NRC would have to review each request.

Approval of this final rule eliminates the problem described and is consistent with previous NRC actions. Further, the direct final rule will have no adverse effect on public health and safety. This direct final rule has no significant identifiable impact or benefit on other Government agencies. On the basis of this discussion of the benefits and impacts of the alternatives, the NRC concludes that the requirements of the final rule are commensurate with the Commission's responsibilities for public health and safety and the common defense and security. No other alternative is believed to be satisfactory. Therefore, this action is recommended.

Regulatory Flexibility Certification

As required by the Regulatory Flexibility Act of 1980 (5 U.S.C. 605(b)), the Commission certifies that this rule does not have a significant economic impact on a substantial number of small entities. The final rule affects only the licensing and operation of nuclear power plants, independent spent fuel storage facilities, and Transnuclear, Inc. These entities do not fall within the scope of the definition of ''small entities'' set forth in the Regulatory Flexibility Act or the NRC's size standards (10 CFR 2.810).

Backfit Analysis

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

Small Business Regulatory Enforcement Fairness Act

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

List of Subjects in 10 CFR Part 72

Administrative practice and procedure, Criminal penalties, Manpower training programs, 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. 552 and 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, HIGH-LEVEL RADIOACTIVE WASTE, AND REACTOR-RELATED GREATER THAN CLASS C 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. 102–486, sec. 7902, 106 Stat. 3123 (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. Section 72.214, Certificate of Compliance 1004 is revised to read as follows:

§ 72.214 List of approved spent fuel storage casks.

* * * * *

Certificate Number: 1004.

Initial Certificate Effective Date: January 23, 1995.

Amendment Number 1 Effective Date: April 27, 2000.

Amendment Number 2 Effective Date: September 5, 2000.

Amendment Number 3 Effective Date: September 12, 2001.

Amendment Number 4 Effective Date: February 12, 2002.

Amendment Number 5 Effective Date: January 7, 2004.

SAR Submitted by: Transnuclear, Inc.

SAR Title: Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel.

Docket Number: 72–1004.

Certificate Expiration Date: January 23, 2015.

Model Number: Standardized NUHOMS®–24P, NUHOMS®–52B, NUHOMS®–61BT, NUHOMS®–24PHB, and NUHOMS®–32PT.

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Dated at Rockville, Maryland, this 19th day of December, 2003.

For the Nuclear Regulatory Commission.

William D. Travers,

Executive Director for Operations.

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