



Filer H/W - 002.3 (26)
- Dry Storage
- Transportation

POLICY ISSUE
(Information)

August 25, 1989

SECY-89-261

For: The Commissioners

From: James M. Taylor
Acting Executive Director for Operations

Subject: PROGRESS MADE BY U.S. DEPARTMENT OF ENERGY (DOE) AND THE INDUSTRY TO DEVELOP CASK DESIGNS TO ACHIEVE COMPATIBILITY FOR DRY STORAGE AND TRANSPORTATION PURPOSES

Purpose: To inform the Commission of progress made by DOE and the industry in addressing potential compatibility problems between dry spent fuel storage system designs and offsite transportation of spent fuel from such systems, without need to return fuel to reactor basins.

Summary: In its March 1, 1989, letter to DOE, which commented on DOE's "Final Version Dry Cask Storage Study" (DOE/RW-0220), the Commission indicated that it was pleased with DOE's positive response to the Commission's concern about the need to ensure compatibility of various steps in the storage, transportation, and disposal of spent fuel to enhance the safety and efficiency of fuel handling. The Commission encouraged DOE to actively pursue the commitment that it made in its final study to accomplish resolution of this matter, both through its own actions and in concert with industry.

Since issuance of the final study, DOE's Office of Civilian Radioactive Waste Management (OCRWM) has taken actions in response to the Commission's comments. As DOE committed, DOE/OCRWM has raised this matter with utilities in its Annual Capacity Report issue-resolution process. One part of DOE's transportation cask development initiative is to develop "Specialty Casks" to handle a variety of atypical fuel, hardware, components, etc. DOE has decided that, when this activity is begun, consideration of canistered fuel (as in the NUHOMS design proposed for use at Duke Power Company's Oconee Nuclear Station) will be included in setting requirements for "Specialty Casks." DOE/OCRWM is continuing its interactions with utilities and their representative organizations on this issue and related storage/transportation cask matters. The leading technical matter is allowance for burnup credit in criticality design.

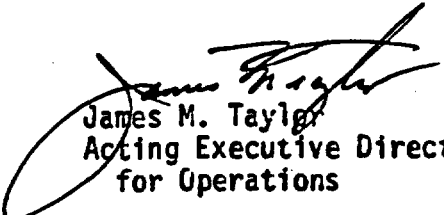
Contact:
J. P. Roberts, NMSS/IMSB
49-20608

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The industry is also seeking to address this issue directly by meeting with U.S. Nuclear Regulatory Commission (NRC) staff to discuss potential initiatives. In 1988 and early 1989, Nuclear Assurance Corporation met with Office of Nuclear Material Safety and Safeguards (NMSS) staff to discuss submittal of a dual-purpose spent fuel storage/transportation cask design.

A topical report for this cask design is expected to be submitted this fall to NMSS' Fuel Cycle Safety Branch for its review for storage in conjunction with an application to NMSS' Transportation Branch for certification as a shipping cask under 10 CFR Part 71. Given successful Parts 71 and 72 reviews and assuming issuance in final form in 1990 of the Commission's dry spent fuel storage cask certification rulemaking [PR 50, 72, 170 (54 FR 19379)], this dual-purpose cask design could represent the first dry-storage technology design to fully address and meet the Commission's direction.

NMSS staff members continue to respond to other industry queries on meeting the Commission's concern for a safe and efficient back end of the fuel cycle. NRC staff expects to see continued progress on this issue in the coming year.



James M. Taylor
Acting Executive Director
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NUCLEAR REGULATORY COMMISSION

10 CFR Parts 50, 72, and 170

RIN: 3150-AC76

Storage of Spent Nuclear Fuel in NRC-Approved Storage Casks at Nuclear Power Reactor Sites

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AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is proposing to amend its regulations to provide, as directed by the Nuclear Waste Policy Act of 1982, for the storage of spent fuel at the sites of power reactors without, to the maximum extent practicable, the need for additional site-specific approvals. Holders of power reactor operating licenses would be permitted to store spent fuel, in casks approved by NRC, under a general license. The proposed rule contains criteria for obtaining an NRC Certificate of Compliance for spent fuel storage casks.

DATE: Submit comments by June 19, 1989. Comments received after this date will be considered if it is practical to do so, but the Commission is able to assure consideration only for comments received on or before this date.

ADDRESSES: Mail written comments to Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555 ATTN: Docketing and Service Branch. Deliver comments to One White Flint North, 11555 Rockville Pike, Rockville, MD between 7:30 a.m. and 4:15 p.m. weekdays.

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Copies of NUREG-0459, 0575, 0709, 1092, 1140, and NUREG/CR-1223, reports which are referenced in this notice and the environmental assessment, may be purchased through the U.S. Government Printing Office by calling (202) 275-2060 or by writing to the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013-7082. DOE/RW-0196, referenced in the regulatory analysis, is available from the U.S. Department of Energy, Office of Scientific and Technical Information, Post Office Box 62, Oak Ridge, TN 37831. Copies of DOE/RL-87-11, referenced in the environmental assessment, and the NUREG reports listed above may be purchased from the National Technical Information Service, U.S. Department of Commerce, Springfield, Virginia 22161. Copies of the NUREG reports listed above, the environmental assessment and finding of no significant environmental impact, and comments received on the proposed rule are available for inspection and copying for a fee at the NRC Public Document Room, 2120 L Street NW., Washington, DC, Lower Level.

FOR FURTHER INFORMATION CONTACT: William R. Pearson, Office of Nuclear Regulatory Research, Nuclear Regulatory Commission, Washington, DC 20555. Telephone: (301)492-3764.

SUPPLEMENTARY INFORMATION:

Background

Section 218(a) of the Nuclear Waste Policy Act of 1982 (NWPA) includes the following directive, "The Secretary [of 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 NWRPA states, in part, that "the 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."

Discussion

This proposed rule would allow power reactor licensees to store spent fuel at the reactor site without additional site-specific reviews. A general license would be issued to holders of power reactor licenses for the storage of spent fuel in dry casks approved by the NRC. The reactor licensee would have to show that there are no changes required in the facility technical specifications or unreviewed safety questions related to activities involving storage of spent fuel under the general license. The licensee would also have to show conformance with conditions of the NRC Certificate of Compliance issued for the cask. The licensee would have to establish and maintain records showing compliance, which would have to be made available for inspection by the Commission.

This proposed rule would not limit storage of spent fuel to that which is generated at the reactor site. Transfers of spent fuel from one reactor site to another are authorized under the receiving site's facility operating license pursuant to 10 CFR Part 50. The holder of a reactor operating license would apply for a license amendment, under

§ 50.90 unless already authorized in the operating license, for the receipt and handling of the spent fuel from another reactor. If transfer of spent fuel is authorized, the reactor licensee would also have to make a request for amendment of the Price-Anderson indemnification agreement to provide for coverage of the transferred spent fuel. 10 CFR Part 72 is not germane to such transfers of spent fuel. If the spent fuel has been previously transferred and is currently stored in the reactor spent fuel pool, the only consideration under the general license would be whether or not the spent fuel meets conditions of the cask's Certificate of Compliance.

Although experience with storage of spent fuel under water is greater than with dry storage in casks, experience with storage of spent fuel in dry casks is extensive and widespread. The Canadians have been storing dry CANDU-type spent fuel at Whiteshell in vertical concrete casks called silos since 1975. Although the storage of spent fuel at Whiteshell does not involve light-water-reactor (LWR) fuel, it has contributed to the knowledge and experience of dry spent fuel storage in concrete casks. Dry cask storage has been demonstrated in West Germany. There has also been experience with dry spent fuel storage in the United States. The Department of Energy (DOE) and its predecessors have kept non-LWR spent fuel in dry storage in vaults and dry wells since the 1960s. An NRC survey of the dry storage of spent fuel, in the United States and elsewhere, was presented in NUREG/CR-1223, "Dry Storage of Spent Fuel - A Preliminary Survey of Existing Technology and Experience" (April 1980). NUREG/CR-1223, at Section IV.C, contains a description of DOE demonstration of dry LWR spent fuel storage in sealed storage casks (SSC) and dry wells. The storage of LWR spent fuel in SSC, which is an above ground,

steel-lined, reinforced concrete cylinder or cask, started in 1979. The DOE demonstration program has continued and has been expanded to include dry storage in metal casks and storage of consolidated fuel rods; and storage of spent fuel assemblies. Programs have been conducted by DOE in cooperation with Virginia Power at its Surry plant, with Carolina Power and Light at its H.B. Robinson 2 plant, and with General Electric at its Morris plant for dry storage of LWR spent fuel. Also dry storage of LWR spent fuel assemblies continues at the Idaho National Engineering Laboratory, along with demonstration of their disassembly and storage of the consolidated fuel rods.

The NRC staff has obtained a substantive amount of information from the DOE development programs. It has also gained experience from the issuance of licenses for the onsite storage of spent fuel in nodular cast iron casks at the Surry site of Virginia Power and in stainless steel canisters stored inside concrete modules at the H.B. Robinson 2 site of Carolina Power and Light. The safety of dry storage of spent fuel was considered during development of the Commission's original regulations in 10 CFR Part 72, "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation (ISFSI)," which was promulgated on November 12, 1980 (45 FR 74693). A final rule entitled, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," which replaced the regulations issued on November 12, 1980, was published in the Federal Register on August 19, 1988 (53 FR 31651) and became effective September 19, 1988. This final rule was issued mainly to provide for licensing the storage of spent fuel and high-level waste at a monitored retrievable storage (MRS) facility,

and does not cover the mandates of Sections 133 and 218(a) of the NWA. However, it did specifically address the safety of dry storage of spent fuel.

Activities related to loading and unloading spent fuel casks are routine procedures at power reactors. The procedures for dry storage of spent fuel in casks would be an extension of these procedures. Over the last several years the staff has reviewed and approved four spent fuel storage cask designs. Requests for approval of cask designs are currently submitted in the form of topical safety analysis reports (TSARs). Four dry storage cask TSARs have been approved for referencing, which means that an ISFSI license applicant may reference appropriate parts of these reports in licensing proceedings for the storage of spent fuel. This greatly reduces an ISFSI license applicant's time, effort, and cost. The same reliance on an approved safety analysis is being made for on-site dry cask storage.

Separate topical safety analysis reports have been received for design of casks fabricated using nodular cast iron, thick-walled ferritic steel, concrete, and stainless steel and lead. Four spent fuel storage cask topical safety analysis reports have been approved for referencing in specific license applications, as previously mentioned, and four are still under review at the present time. In particular, the topical safety analysis report (TSAR) for the Castor-V/21, entitled "Topical Safety Analysis Report for the Castor Cask, Independent Spent Fuel Storage Installation (Dry Storage)," was submitted by General Nuclear Systems, Inc. on December 16, 1983. The NRC staff approved the Castor-V/21 TSAR for reference in licensing proceedings on September 30, 1985. In a specific licensing proceeding under Part 72 by Virginia

Electric Power Company (VEPCo) in January of 1985, the use of the Castor-V/21 was approved on July 7, 1986 for storage of spent fuel at their Surry Power Station. Currently, there are seven of these casks filled and stored on the ISFSI pad at the Surry site and an eighth cask is filled and ready to be moved to the storage pad.

Although the Castor-V/21, is the only spent fuel storage cask currently being used, the SARs for the Westinghouse MC-10, and the Nuclear Assurance Corporation's NAC/ST and NAC-C28 S/T casks have been approved for reference. These casks are being proposed for approval under §72.214, "List of approved spent fuel storage casks." While the Certificate of Compliance for each cask may differ in some specifics, e.g., certificate number; operating procedures; training exercises; spent fuel specifications, many of the safety conditions are very similar. Copies of the Certificates of Compliance are being issued for comment, and are available for inspection and copying for a fee at the Commission's Public Document Room at 2120 L Street NW., Washington, DC, Lower Level. Single copies of the proposed certificates may be obtained from J.P. Roberts, Fuel Cycle Safety Branch, Division of Industrial and Medical Nuclear Safety, Office of Nuclear Materials Safety and Safeguards; (Telephone: (301)492-0608).

Storage casks certified in the future will be routinely added to the listing in § 72.214 through rulemaking procedures. Since this type of rulemaking would neither constitute a significant question of policy nor amend 10 CFR Parts 0, 2, 7, 8, 9 Subpart C, or 110, the Commission concludes that such additions to § 72.214 may be made under the rulemaking authority delegated to the Executive Director for Operations. Certificates of Compliance will be exhibited in a NUREG report issued by the NMSS staff, which will be updated as appropriate.

During review for the cask designs that are being proposed for certification in this rulemaking, the NMSS staff considered compatibility with transportation to and disposal at DOE facilities and will continue to do so for future cask approvals. The vendors of these casks have indicated that they will apply for approval of their casks for spent fuel transportation. Currently there is limited knowledge concerning specific design criteria by which to design storage casks for minimizing the handling of spent fuel between the time it is put into casks for storage at a reactor site and the time it will be handled for storage at a monitored retrievable storage facility (MRS) or disposal at a geologic repository. However, the staff will remain in contact with DOE and will assure, to the extent practicable, that cask designs incorporate the latest design criteria available at the time that the cask design is approved or certified.

The NRC experience in the review of cask design and fabrication, and licensing of spent fuel storage installations on the site of operating reactors, has been documented in part by publication of two draft regulatory guides. In April of 1986, two draft regulatory guides entitled "Standard Format and Content for the Safety Analysis Report for Onsite Storage of Spent Fuel Storage Casks" (Task number CE-301) and "Standard Format and Content for a Topical Safety Analysis Report for a Dry Spent Fuel Storage Cask" (Task number CE-306) were issued for public comment. These draft guides are being processed and the guide under task number CE-301 will become Regulatory Guide 3.62 and the one under task number CE-302 will become Regulatory Guide 3.61. Single copies of these draft guides may be obtained from W.R. Pearson, Office of Nuclear Regulatory Research, Nuclear Regulatory Commission, Washington, DC 20555 (Telephone: (301) 492-3764).

The passive nature of dry storage of spent fuel in casks provides operational benefits attractive to potential users. One benefit is that there is no need to provide operating systems to purify and circulate cooling water or other fluid. Another benefit is that the potential for corrosion of the fuel cladding and reaction with the fuel is reduced, because an inert atmosphere is expected to be maintained inside dry spent fuel storage casks. Because cooling of the spent fuel is a passive activity, active mechanisms, such as pumps and fans, are not required. Although Part 72 allows storage of any spent fuel over one year old, (i.e., one year since the fuel was involved in a sustained nuclear chain reaction), it is anticipated that most spent fuel stored in casks will be five years old or more. Because of the passive nature of cask cooling, the storage capacity of a cask is significantly increased as the spent fuel is aged, especially for fuel that is five years old or more. It is probable that reactor licensees will remove the older fuel from their storage pools to take advantage of this additional cask storage capacity.

As a result of the above discussion, the Commission believes that dry storage of spent fuel in casks approved by the Commission will provide adequate protection to public health and safety and the environment.

Proposed Rule

The General License

Under this proposed rule, a general license would be issued to holders of nuclear power reactor licenses to store spent fuel at reactor sites in casks approved by the NRC. The Commission will rely on dry storage of spent fuel in casks for confinement of radioactive material to

provide adequate protection of public health and safety and the environment. A power reactor license holder would have to notify the Commission before storing spent fuel under the general license for the first time and register use of each cask as the spent fuel is stored. A separate record would also be established for each cask by the cask vendor, which would be transferred to and be maintained by cask users.

The reactor license holder would have to ensure that the storage of spent fuel will be in compliance with the conditions of the cask Certificate of Compliance, including assurance that site parameters and other design bases are within the envelope of the values analyzed in the cask safety analysis report. Evaluations would also have to be made to ensure that there will be no changes necessary to the facility technical specifications and that there are no unresolved safety questions in activities involving the storage casks. Procedures and criteria in 10 CFR 50.59 would be used for these evaluations. These types of evaluation are currently done for specific licenses issued under Part 72. Issues related to systems and components used both for reactor operations and spent fuel storage activities would be included. Most concerns to date have been related to control of heavy loads and have been accommodated. If there is a safety problem or a change in technical specifications required, and the reactor license holder wishes to store spent fuel under the general license, the problem would have to be resolved before storing spent fuel under the general license and could include submittal of an application for license amendment under Part 50 if necessary.

Reactor licensees would have to review their quality assurance program, emergency plan, training program, and radiation control program using procedures in §50.59 and modify them as may be necessary to cover the activities

related to spent fuel storage under the general license. These plans and programs are in effect for reactor operations and the appropriate existing plan or program could be modified or amended to cover activities related to the spent fuel storage.

The reactor licensee would have to conform to conditions in the cask's Certificate of Compliance, which includes conducting activities according to written operating procedures. These operating procedures could be developed using the same or a similar system by which the operating procedures for the reactor were developed.

Instances in which significant reductions in the safety effectiveness of or defects in casks are discovered must be reported to the NRC. Initial reports would be submitted under 10 CFR § 50.72 "Immediate notification requirements for operating nuclear power reactors." A new paragraph would be added to § 50.72(b)(2) for this purpose. A complete written report would be submitted within 30 days.

When the power reactor operating license expiration date approaches, the holder of the license must take some actions. Under 10 CFR 50.54(bb) the reactor license holder must submit a program in writing to the Commission, no later than five years prior to the license expiration date, showing how the reactor licensee intends to manage and provide funding for the management of all irradiated fuel on the reactor site. This program would have to include the spent fuel stored under the general license proposed in this rulemaking. The reactor licensee will also have to decide whether to request termination of the reactor operating license under 10 CFR 50.82. If the reactor license holder decides to apply for termination of the license, the plan submitted with the application must show how the spent fuel stored under this general license will be removed from the site. The plan would have to include an explanation of when and

how the spent fuel will be moved, unloaded, and shipped prior to starting decommissioning of the equipment needed for these activities.

In part, the environmental assessment for this rulemaking relies on findings from the Waste Confidence Decision (49 FR 34658, 8/31/84), in which the Commission concluded they had confidence that there would be no significant environmental impacts from the storage of spent fuel for a period of 30 years beyond the expiration date of reactor licenses. Thus, an application for reactor license termination that proposes a decommissioning period beyond this 30-year period would have to contain a discussion of the environmental impacts from storage of the spent fuel beyond the period analyzed by the Commission. The general license would terminate automatically when the spent fuel is removed from storage.

Cask Certification

A spent fuel storage cask will be relied on to provide safe confinement of radioactive material independent of a nuclear power reactor's site, so long as conditions of the Certificate of Compliance are met. The storage cask approval program, in many respects, will be analogous to that now conducted for spent fuel casks approved for transportation under 10 CFR Part 71. A cask vendor will submit a safety analysis report showing how the cask design, fabrication, and testing will ensure adequate protection of public health and safety. Certificates of Compliance will be exhibited in a NUREG report, which will be made available to the public.

Spent fuel is now temporarily stored at reactor sites, ISFSI, and elsewhere until a Department of Energy monitored retrievable storage facility or high-level radioactive waste repository is ready. The spent fuel will then be shipped to one of these facilities. Changes in the law could shift the Commission's policies or cause a change

in its regulations. It appears to be prudent that cask design approvals, i.e., approval of spent fuel cask topical safety analysis reports under current regulations; or issuance of Certificates of Compliance under the proposed regulation, should be for a limited time period. Current regulations limit the storage of spent fuel in an ISFSI to 20 years, after which the license may be renewed. The Commission believes that 20-year increments are appropriate for such cask design approvals, after which designs may be renewed.

The holder of the cask Certificate of Compliance (cask vendor) should apply for re-approval of a storage cask. Submittal of an application would be made 17 years after the initial cask approval date, which is three years prior to the expiration date of the cask certificate, to allow time for the NRC staff to reevaluate the cask safety and reissue the cask certificate. If the holder of a cask certificate goes out of business or will not submit an application for reapproval in a timely manner for any reason, the Commission should be notified and it in turn would notify cask users. In any case, cask users would have to ensure that spent fuel is stored in casks approved by the NRC. Several options would be available to licensees. If a cask design is reapproved under submittals by the vendor, the Commission would notify all users and the only action necessary for the users would be to update the cask's records. If the cask vendor does not apply for reapproval, for whatever reason, the licensee would be notified by the Commission. The licensee would then have to arrange for reapproval or remove casks from service as their 20-year approved storage life expired. This could mean removal of the spent fuel and storing it elsewhere.

The Commission believes that a prudent concern for overall activities related to the back-end of the LWR fuel cycle dictates that consideration should be given to the compatibility of spent fuel storage cask designs with the transportation of the spent fuel to its ultimate disposition at a DOE facility. Cask designers should be aware of DOE developments and plans for transportation of spent fuel offsite and should design spent fuel storage casks, to the extent that is practicable given the information that is available at the time that the cask is designed, for compatibility with future disposition of the spent fuel. The cask designs that are included in this rulemaking comply to the extent practicable at this time. The Commission notes that the vendors of these casks have indicated their intent to pursue certification for their cask as a shipping container for offsite transportation under 10 CFR Part 71. However, spent fuel can be safely off-loaded from storage casks at reactor sites, if necessary, at the end of the storage period. In the interest of minimizing overall fuel cycle impacts the Commission encourages storage cask design developments that would reduce the handling of spent fuel.

The scope of this rule is to allow holders of nuclear power reactor licenses to store spent nuclear fuel at reactor sites under a general license using certified dry storage casks, because use of these casks is essentially independent of site characteristics. The Commission has evaluated and approved, in specific licenses issued under 10 CFR Part 72, other types of dry storage modules. These methods may be approved in the future for use under a general license.

NRC costs related to spent fuel storage cask Certificates of Compliance, cask fabrication inspections, and onsite inspections would be fully recovered. The schedule of fees in 10 CFR 170.31 would be revised to recover these costs.

Safeguards

Spent fuel removed from light water reactors contains low enriched uranium, fission products, plutonium, and other transuranium elements (transuranics). Owing to the special nuclear material in spent fuel, safeguards for an independent spent fuel storage installation must protect against theft and radiological sabotage and must provide for material accountability. The requirements for physical protection are set forth in proposed § 72.212. No specific requirements for material control and accounting are being added because existing requirements in Parts 72 and 50 are adequate.

The theft issue arises mainly from the plutonium component of the spent fuel. Plutonium, when separated from other substances, can be used in the construction of nuclear explosive devices and therefore must be provided with a high level of physical protection. However, the plutonium contained in spent fuel is not readily separable from the highly radioactive fission products and other transuranics and for that reason is not considered a highly attractive material for theft. Moreover, the massive construction of casks significantly complicates theft scenarios. For these reasons no specific safeguards measures to protect against theft are proposed other than maintaining accounting records and conducting periodic inventories of the special nuclear material contained in the spent fuel.

Safeguards measures should be consistent with existing site provisions against potential radiological sabotage. The term "radiological sabotage" is defined in 10 CFR Part 73 and means any deliberate act directed against a plant or transport vehicle and cask in which an activity licensed under NRC regulations is conducted, or against a component of a plant or transport

vehicle and cask which could directly or indirectly endanger the public health and safety by exposure to radiation.

In assessing the probability and consequences of radiological sabotage, the NRC considers: (1) the threat to storage facilities; (2) the response of typical storage casks or vaults and their contained spent fuel to postulated acts of radiological sabotage; and (3) the public health consequences of acts of radiological sabotage.

The NRC has carried out studies to develop information about possible adversary groups which might pose a threat to licensed nuclear facilities. The results of these studies are published in NUREG-0459, "Generic Adversary Characteristics - Summary Report" (March 1979) and NUREG-0703, "Potential Threat to Licensed Nuclear Activities from Insiders" (July 1980). Actions against facilities were found to be limited to a number of low consequence activities and harassments, such as hoax bomb threats, vandalism, radiopharmaceutical thefts, and firearms discharges. The list of actions is updated annually in a NUREG-0525, "Safeguards Summary Event List" (July 1987). None of the actions have affected spent fuel containment and, thus, have not caused any radiological health hazards.

In addition, the NRC staff regularly consults with law enforcement agencies and intelligence-gathering agencies to obtain their views concerning the possible existence of adversary groups interested in radiological sabotage of commercial nuclear facilities. None of the information the staff has collected confirms the presence of an identifiable domestic threat to dry storage facilities or to other components of nuclear facilities.

Despite the absence of an identified domestic threat, the NRC has considered it prudent to study the response of loaded casks to a range of sabotage scenarios. The study is classified. However, an overview of the study is provided in the following paragraphs.

Being highly radioactive, spent fuel requires heavy shielding for safe storage. Typical movable storage casks are of metal or concrete, weigh 100 tons, and have wall thickness from 10 to 16 inches of metal or 30 inches of concrete. The structural materials and dimensions enable the casks and vaults to withstand attack by small arms fire, pyrotechnics, mechanical aids, high velocity objects, and most forms of explosives without release of spent fuel. After considering various technical approaches to radiological sabotage, the NRC concluded that radiological sabotage, to be successful, would have to be carried out with the aid of a large quantity of explosives.

The consequences to the public health and safety would stem almost exclusively from the fraction of the release that is composed of respirable particles. In an NRC study, an experiment was carried out to evaluate the effects of a very severe, perfectly executed explosive sabotage scenario against a simulated storage cask containing spent fuel assemblies. The amount of fuel disrupted was measured. The fraction of disrupted material of respirable dimensions (0.005%) had been determined in a previous experiment. From this information, an estimate of the airborne, respirable release was made, and the dose as a function of range and other variables was calculated. In a typical situation, for an individual at the boundary of the reactor site (taken as 100 meters from the location of the release) and in the center of the airborne plume, the

whole-body dose was calculated to be 1 rem and the 50-year dose commitment (to the lung, which is the most sensitive organ) was calculated to be 2 rem. Doses higher or lower can be obtained depending on the variables used in the calculation. Variables include the meteorological conditions, the age and burn-up of the fuel, the heat-induced buoyancy of the airborne release, the range to the affected individual, and the explosive scenario assumed.

Although the experiment and calculations carried out lead to a conclusion of low public health consequences, there are limitations that must be taken into account. In particular, consequence modeling assumptions more severe than those in the foregoing calculation are possible if unconstrained sabotage resources or protracted loss of control of the storage site are allowed. For that reason, protection requirements are proposed to provide for (1) early detection of malevolent moves against the storage site, and (2) a means to quickly summon response resources to assure against protracted loss of control of the site.

The proposed requirements comprise a subset of the overall protection requirements currently in force at every operating nuclear power reactor. Inasmuch as the security force at each reactor is thoroughly familiar with requirements similar to those proposed and has years of experience in carrying them out, the NRC concludes that the requirements can be successfully imposed through a general license for storage of spent fuel in NRC-approved casks without the need for advanced NRC review and approval of a physical security plan or other site-specific documents before the reactor licensee implements the requirements.

Material control and accounting (MC&A) requirements are designed to protect against the undetected loss of the special nuclear material in

spent fuel by maintaining vigilance over the material, tracking its movement and location, monitoring its inventory status, maintaining records of transactions and movements, and issuing reports of its status at the time of physical inventory. Similar requirements for MC&A have been applied to power reactors, to spent fuel storage at independent spent fuel storage installations, and to operations at certain other classes of fuel cycle facilities without requiring the licensee to submit a plan to document how compliance will be achieved. In these situations the requirements have been found to be sufficient. For these reasons, it is concluded that the MC&A requirements for the dry storage of spent fuel at power reactors can be handled under a general license.

A minor editorial change to § 72.30(b) is also proposed to make clear that a decommissioning funding plan is an integral part of an applicant's proposed decommissioning plan.

Finding of No Significant Environmental Impact: Availability

The Commission has determined under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in Subpart A of 10 CFR Part 51, that this rule, 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. The rule is mainly administrative in nature and would not change safety requirements, which could have significant environmental impacts. The proposed rule would provide for power reactor licensees to store spent fuel in casks approved by NRC at reactor sites without additional site-specific approvals by the Commission. It would set forth

conditions of a general license for the spent fuel storage and procedures and criteria for obtaining storage cask approval. 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., Washington, DC, Lower Level. Single copies of the environmental assessment and the finding of no significant impact are available from W.R. Pearson, Office of Nuclear Regulatory Research, Nuclear Regulatory Commission, Washington, DC 20555; Telephone: (301) 492-3764.

Paperwork Reduction Act Statement

This proposed rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.). This rulemaking has been submitted to the Office of Management and Budget for review and approval of the information collection requirements.

The reporting burden for this collection of information is estimated to average 2,336 hours, which will be primarily for development and submittal of a safety analysis report (SAR) by spent fuel storage cask vendors. Review and approval of an SAR is necessary in order to obtain a Certificate of Compliance for a cask design from NRC. A Certificate of Compliance is required for each cask design before these casks can be used for spent fuel storage under the general license in this rule. Responses required from power reactor licensees under this rule would be initial notification for use of the general license and submittal of a notice

when each cask is stored. Thus, no significant reporting burden is anticipated for these licensees. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for reducing this burden, to the Records and Reports Management Branch, Mail Stop P-530, Division of Information Support Services, Office of Information Resources Management, U.S. Nuclear Regulatory Commission, Washington, DC 20555; and to the Paperwork Reduction Project (3150-0132), Office of Management and Budget, Washington, DC 20503.

Regulatory Analysis

The Commission has prepared a preliminary regulatory analysis on this proposed rule. The analysis examines the benefits and impacts considered by the Commission. The Preliminary Regulatory Analysis is available for inspection in the NRC Public Document Room, 2120 L Street NW., Washington, DC, Lower Level. Single copies may be obtained from W.R. Pearson, Office of Nuclear Regulatory Research, Nuclear Regulatory Commission, Washington, DC 20555; Telephone: (301)492-3764.

The Commission requests public comments on the preliminary regulatory analysis, which may be submitted to the NRC as indicated under the ADDRESSES heading.

Regulatory Flexibility Act Certification

As required by the Regulatory Flexibility Act of 1980 (5 U.S.C. 605(b)), the Commission certifies that this rule, if adopted, will not have a significant economic impact on a substantial number of small

entities. This proposed rule affects only licensees owning and operating nuclear power reactors. The owners of nuclear power plants do not fall within the scope of the definition of "small entities" set forth in Section 601(3) of the Regulatory Flexibility Act, 15 U.S.C. 632, or the Small Business Size Standards set out in regulations issued by the Small Business Administration at 13 CFR Part 121.

Backfit Analysis

The NRC has determined that the backfit rule, 10 CFR 50.109, does not apply to this proposed rule, and, thus, a backfit analysis is not required for this proposed rule, because these amendments do not involve any provisions which would impose backfits as defined in § 50.109(a)(1).

List of Subjects

Part 50: Antitrust, Classified information, Fire protection, Incorporation by reference, Intergovernmental relations, Nuclear power plants and reactors, Penalty, Radiation protection, Reactor siting criteria, and Reporting and recordkeeping requirements.

Part 72: Manpower training programs, Nuclear materials, Occupational safety and health, Reporting and recordkeeping requirements, Security measures, Spent fuel.

Part 170: Byproduct material, Nuclear materials, Nuclear power plants and reactors; Penalty, Source material, Special nuclear material.

For 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, the Nuclear Waste Policy Act of 1982, and 5 U.S.C. 552 and 553, the NRC is proposing to adopt the following revisions to 10 CFR Part 72 and conforming amendments to 10 CFR Parts 50 and 170.

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

1. The authority citation for Part 72 is revised 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 (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, 96 Stat. 2230 (42 U.S.C. 10153) and 218(a), 96 Stat. 2252 (42 U.S.C. 10198).

For the purposes of sec. 223, 68 Stat. 958, as amended (42 U.S.C. 2273); §§ 72.6, 72.22, 72.24, 72.26, 72.28(d), 72.30, 72.32, 72.44(a), (b)(1), (4), (5), (c), (d)(1), (2), (e), (f), 72.48(a), 72.50(a), 72.52(b), 72.72(b), (c), 72.74(a), (b), 72.76, 72.78, 72.104, 72.106, 72.120, 72.122, 72.124, 72.126, 72.128, 72.130, 72.140(b), (c), 72.148, 72.154, 72.156, 72.160, 72.166, 72.168, 72.170, 72.172, 72.176, 72.180, 72.184, 72.186 are issued under sec. 161b, 68 Stat. 948, as amended (42 U.S.C. 2201(b)); §§ 72.10(a), (e), 72.22, 72.24, 72.26, 72.28, 72.30, 72.32, 72.44(a), (b)(1), (4), (5), (c), (d)(1), (2), (e), (f), 72.48(a), 72.50(a), 72.52(b), 72.90(a)-(d), (f), 72.92, 72.94, 72.98, 72.100, 72.102(c), (d), (f), 72.104, 72.106, 72.120, 72.122, 72.124, 72.126, 72.128, 72.130, 72.140(b), (c), 72.142, 72.144, 72.146, 72.148, 72.150, 72.152, 72.154, 72.156, 72.158, 72.160, 72.162, 72.164, 72.166, 72.168, 72.170, 72.172, 72.176, 72.180, 72.182, 72.184, 72.186, 72.190, 72.192, 72.194 are issued under sec. 161i, 68 Stat. 949, as amended (42 U.S.C. 2201(i)); and §§ 72.10(e), 72.11, 72.16, 72.22, 72.24, 72.26, 72.28, 72.30, 72.32, 72.44(b)(3), (c)(5), (d)(3), (e), (f), 72.48(b), (c), 72.50(b), 72.54(a), (b), (c), 72.56, 72.70, 72.72, 72.74(a), (b), 72.76(a), 72.78(a), 72.80, 72.82, 72.92(b), 72.94(b), 72.140(b), (c), (d), 72.144(a), 72.146, 72.148, 72.150, 72.152, 72.154(a), (b), 72.156, 72.160, 72.162, 72.168, 72.170, 72.172, 72.174, 72.176, 72.180, 72.184,

72.186, 72.192, 72.212(b), 72.216, 72.218, 72.230, 72.234(e) and (g) are issued under sec. 161o, 68 Stat. 950, as amended (42 U.S.C. 2201(o)).

2. In § 72.30, paragraph (b) is revised to read as follows:

§ 72.30 Decommissioning planning, including financing and recordkeeping.

* * * * *

(b) The proposed decommissioning plan must also include a decommissioning funding plan containing information on how reasonable assurance will be provided that funds will be available to decommission the ISFSI or MRS. This information must include a cost estimate for decommissioning and a description of the method of assuring funds for decommissioning from paragraph (c) of this section, including means of adjusting cost estimates and associated funding levels periodically over the life of the ISFSI or MRS.

* * * * *

3. New Subpart K and Subpart L are added to read as follows:

**Subpart K - General License for Storage of Spent
Fuel at Power Reactor Sites**

Sec.

72.210 General license issued.

72.212 Conditions of general license issued under § 72.210.

72.214 List of approved spent fuel storage casks.

72.216 Reports.

72.218 Termination of licenses.

72.220 Violations.

Subpart L - Approval of Spent Fuel Storage Casks

- 72.230 Procedures for spent fuel storage cask submittals.
- 72.232 Inspection and tests.
- 72.234 Conditions of approval.
- 72.236 Specific criteria for spent fuel storage cask approval.
- 72.238 Issuance of an NRC Certificate of Compliance.
- 72.240 Conditions for spent fuel storage cask reapproval.

**Subpart K - General License for Storage of Spent Fuel
at Power Reactor Sites****§ 72.210 General license issued.**

A general license is hereby issued for the storage of spent fuel in an independent spent fuel storage installation at power reactor sites to persons authorized to operate nuclear power reactors under Part 50 of this chapter.

§ 72.212 Conditions of general license issued under § 72.210.

(a)(1) The general license is limited to storage of spent fuel in casks approved under the provisions of this part.

(2) The general license for the storage of spent fuel in each cask fabricated under a Certificate of Compliance shall terminate 20 years after the date that the cask is first used by the licensee to store spent fuel, unless the cask model is reapproved in which case the general license shall terminate on the revised certification date. In the event that a cask vendor does not apply for a cask model reapproval

under § 72.240, any user or user representative may apply for cask reapproval.

(b) The general licensee shall:

(1)(i) Notify the Nuclear Regulatory Commission under §72.4 at least 90 days prior to first storage of spent fuel under the general license. The notice may be in the form of a letter, but must contain the licensee's name, address, reactor license number (s), and the name and means of contacting a person for additional information. A copy of the submittal must be sent to the Administrator of the appropriate Nuclear Regulatory Commission regional office listed in Appendix D to Part 20.

(ii) Register use of each cask with the Nuclear Regulatory Commission no later than 30 days after using the cask to store spent fuel. This registration may be accomplished by submitting a letter containing the following information: the licensee's name and address, the licensee's reactor license number(s), the name and title of a person who can be contacted for additional information, the cask certificate or model number, and the cask identification number. Submittals must be in accordance with the instructions contained in § 72.4 of this part. A copy of each submittal must be sent to the Administrator of the appropriate Nuclear Regulatory Commission regional office listed in Appendix D to Part 20.

(2) Perform written evaluations showing that conditions set forth in the Certificate of Compliance are met for the anticipated total number of casks to be used for storage. The licensee shall also show that cask storage pads and areas are designed to adequately support the static load of the stored casks. Evaluations must show that the requirements of § 72.104 of this part are met. A copy of this record must be retained

until spent fuel is no longer stored under the general license issued under § 72.210.

(3) Review the approved Safety Analysis Report (SAR) referenced in the Certificate of Compliance and the related NRC Safety Evaluation Report to determine that the licensee's applicable site parameters are enveloped by the cask design capabilities considered in these reports. The results of this review should be documented in the evaluation made in paragraph (b)(2) of this section.

(4) Pursuant to § 50.59 of this chapter, determine whether activities under this general license involve any unreviewed safety question or change in the facility technical specifications.

(5) Protect the spent fuel against the design basis threat of radiological sabotage in accordance with the licensee's physical security plan approved in accordance with § 73.55 of this chapter, with the following additional conditions and exceptions:

(i) The physical security organization and program must be modified as necessary to assure that activities conducted under this general license do not decrease the effectiveness of the protection of vital equipment in accordance with § 73.55 of this chapter.

(ii) Storage of spent fuel must be within a protected area, in accordance with § 73.55(c) of this chapter, but need not be within a separate vital area. Existing protected areas may be expanded or new protected areas added for the purpose of storage of spent fuel in accordance with this general license.

(iii) Notwithstanding any requirements of the licensee's approved security plan, the observational capability required by § 73.55(h)(6) of this chapter may be provided by a guard or watchman in lieu of closed

circuit television for protection of spent fuel under the provisions of this general license.

(iv) For the purposes of this general license, the licensee is exempt from § 73.55(h)(4)(iii)(A) and (5) of this chapter.

(6) Pursuant to the procedures in §50.59 of this chapter, review the reactor emergency plan, quality assurance program, training program, and radiation protection program and modify them as necessary for activities related to storage of spent fuel under the general license.

(7) Maintain a copy of the Certificate of Compliance and documents referenced in the certificate for each model of cask used for storage of spent fuel, until use of the cask model is discontinued. The licensee shall comply with the terms and conditions of the certificate.

(8)(i) Maintain the record provided by the cask supplier for each cask that shows:

- (A) The NRC Certificate of Compliance number;
- (B) The name and address of the cask vendor/lessor;
- (C) The listing of spent fuel stored in the cask; and
- (D) Any maintenance performed on the cask.

(ii) This record must include sufficient information to furnish documentary evidence that any testing and maintenance of the cask has been conducted under an approved quality assurance program.

(iii) In the event that a cask is sold, leased, loaned, or otherwise transferred, this record must also be transferred to and must be accurately maintained by the new registered user. This record must be maintained by the current cask user during the period that the cask is used for storage of spent fuel and retained by the last user until decommissioning of the cask is complete.

(9) Conduct activities related to storage of spent fuel under this general license in accordance with written procedures.

(10) On reasonable notice, make records available to the Commission for inspection.

§ 72.214 List of approved spent fuel storage casks.

The following casks have been reviewed and evaluated by the Commission and are approved for storage of spent fuel under the conditions specified in their Certificates of Compliance. Certificates of Compliance are available for inspection and copying for a fee at the Commission's Public Document Room at 2120 L Street NW., Washington, DC, Lower Level.

Certificate Number: 1000

SAR Submitted by: General Nuclear Systems, Inc.

SAR Title: "Topical Safety Analysis Report for the Castor V/21 Cask
Independent Spent Fuel Storage Installation (Dry Storage)
(TSAR)"

Docket Number: 72-1000

Certification Expiration Date: , 2009

Model Number: CASTOR V/21

Certificate Number: 1001

SAR Submitted by: Westinghouse Electric Corporation

SAR Title: Topical Safety Analysis Report For The MC-10 Cask
Independent Spent Fuel Storage Installation (Dry Storage)

Docket Number: 72-1001

Certification Expiration Date: , 2009

Model Number: MC-10

Certificate Number: 1002

SAR Submitted by: Nuclear Assurance Corporation

**SAR Title: Topical Safety Analysis Report For The NAC Storage/
Transportation Cask Independent Spent Fuel Storage
Installation (Dry Storage)**

Docket Number: 72-1002

Certificate Expiration Date: , 2009

Model Number: NAC S/T

Certificate Number: 1003

SAR Submitted by: Nuclear Assurance Corporation

**SAR Title: Topical Safety Analysis Report For The NAC Storage/
Transportation Cask Containing Consolidated Fuel For
Use at an Independent Spent Fuel Storage Installation
(Dry Storage)**

Docket Number: 72-1003

Certificate Expiration Date: , 2009

Model Number: NAC-C28 S/T

§ 72.216 Reports.

(a) The licensee shall make an initial report under § 50.72(b)(2)(vii) of this chapter of any:

(1) Defect with safety significance discovered in any spent fuel storage cask system or component important to safety; and

(2) Instance in which there is a significant reduction in the effectiveness of any spent fuel storage cask confinement system during use.

(b) A written report, including a description of the means employed to repair any defects or damage and prevent recurrence, must be submitted in accordance with § 72.4 within 30 days. A copy of the written report must be sent to the Administrator of the appropriate Nuclear Regulatory Commission regional office shown in Appendix D to Part 20 of this Chapter.

§ 72.218 Termination of licenses.

(a) The notification regarding planning for the management of all spent fuel at the reactor required by § 50.54(bb) of this chapter must include a plan for removal of the spent fuel stored under this general license from the reactor site. The plan must show how the spent fuel will be managed before starting to decommission systems and components needed for moving, unloading, and shipping this spent fuel.

(b) Spent fuel previously stored may continue to be stored under this general license during decommissioning after submission of an application for termination of the reactor operating license under § 50.82 of this chapter. An application for termination of the reactor operating license submitted under § 50.82 of this chapter must, however, contain a description of how the spent fuel stored under this general license will be removed from the reactor site. If the decommissioning alternative selected under § 50.82 is likely to extend beyond 30 years after the normal term of the reactor operating license, the licensee shall include in the application a discussion of incremental environmental impacts of the extended spent fuel storage.

(c) The reactor licensee shall send a copy of submittals under § 72.218(a) and (b) to the Administrator of the appropriate Nuclear Regulatory Commission regional office shown in Appendix D to Part 20 of this Chapter.

§ 72.220 Violations.

Storage of spent fuel under a general license may be halted or terminated under § 72.84.

Subpart L - Approval of Spent Fuel Storage Casks

§ 72.230 Procedures for spent fuel storage cask submittals.

(a) An application must be submitted in accordance with the instructions contained in § 72.4. A safety analysis report describing the proposed cask design and how the cask should be used to store spent fuel safely must be included with the application.

(b) Casks that have been certified for transportation of spent fuel under Part 71 of this chapter may be approved for storage of spent fuel under this subpart. An application must be submitted in accordance with the instructions contained in § 72.4. A copy of the Certificate of Compliance issued by the NRC for the cask, and drawings and other documents referenced in the certificate, must be included with the application. A safety analysis report showing that the cask is suitable for storage of spent fuel for a period of at least 20 years must also be included.

(c) Public inspection. An application for the approval of a cask for storage of spent fuel may be made available for public inspection under § 72.20.

(d) Fees. Fees for review and evaluation related to issuance of a spent fuel storage cask Certificate of Compliance, inspections related to spent fuel storage in approved casks on reactor sites, and vendor inspection of dry storage casks are those shown in § 170.31 of this chapter.

§ 72.232 Inspection and tests.

(a) The applicant shall permit, and make provisions for, the Commission to inspect at reasonable times the premises and facilities at which a spent fuel storage cask is fabricated and tested.

(b) The applicant shall perform, and make provisions that permit the Commission to perform, tests that the Commission deems necessary or appropriate for the administration of the regulations in this part.

(c) The applicant shall submit a notification under § 72.4 at least 45 days prior to starting fabrication of the first spent fuel storage cask under a Certificate of Compliance.

§ 72.234 Conditions of approval.

(a) Design, fabrication, testing, and maintenance of a spent fuel storage cask must comply with the technical criteria in § 72.236.

(b) Design, fabrication, testing, and maintenance of spent fuel storage casks must be conducted under a quality assurance program that meets the requirements of Subpart G of this part.

(c) Cask fabrication must not start prior to receipt of the Certificate of Compliance for the cask model.

(d)(1) The cask vendor shall ensure that a record is established and maintained for each cask fabricated under the NRC Certificate of Compliance.

(2) This record must include:

(i) The NRC Certificate of Compliance number;

(ii) The cask model number;

(iii) The cask identification number;

(iv) Date fabrication started;

(v) Date fabrication completed;

(vi) Certification that the cask was designed, fabricated, tested, and repaired in accordance with a quality assurance program accepted by NRC;

(vii) Certification that inspections required by § 72.236(j) were performed and found satisfactory; and

(viii) The name and address of the cask user.

(3) The original of this record must be supplied to the cask user. A copy of the current composite record of all casks, showing the above information, must be retained by the cask vendor for the life of the cask. If the cask vendor permanently ceases production of casks under a Certificate of Compliance, this record must be sent to the Commission using instructions in § 72.4.

(e) The composite record required by paragraph (d) of this section must be made available to the Commission for inspection.

(f) The cask vendor shall ensure that written procedures and appropriate tests are established for use of the casks. A copy of these procedures and tests must be provided to each cask user .

§ 72.236 Specific criteria for spent fuel storage cask approval.

(a) Specifications concerning the spent fuel to be stored in the cask, such as the type of spent fuel (i.e., BWR, PWR, both), enrichment of the unirradiated fuel, burn-up (i.e., megawatt-days/MTU), cooling time of the spent fuel prior to storage in the cask, maximum heat designed to be dissipated (i.e., kw/assembly, kw/rod), the maximum spent fuel loading limit, and condition of the spent fuel (i.e., intact assembly or consolidated fuel rods), inerting atmosphere requirements, must be provided.

(b) Design bases and design criteria must be provided for structural members and systems important to safety.

(c) The cask must be designed and fabricated so that the spent fuel is maintained in a subcritical condition under credible conditions.

(d) Radiation shielding and confinement features must be provided to the extent required to meet the requirements in §§ 72.104 and 72.106 of this part .

(e) Casks must be designed to provide redundant sealing of confinement systems.

(f) Casks must be designed to provide adequate heat removal capacity without active cooling systems.

(g) Casks must be designed to store the spent fuel safely for a minimum of 20 years and permit maintenance as required.

(h) Casks must be compatible with wet or dry spent fuel loading and unloading facilities.

(i) Casks must be designed to facilitate decontamination to the extent practicable.

(j) Casks must be inspected to ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce their confinement effectiveness.

(k) Casks must be conspicuously and durably marked with

(1) A model number;

(2) A unique identification number; and

(3) An empty weight.

(l) Casks and systems important to safety must be evaluated, by subjecting a sample or scale model to tests appropriate to the part being tested, or by other means acceptable to the Commission, demonstrating that they will reasonably maintain confinement of radioactive material under normal, off-normal, and accident conditions.

(m) To the extent practicable, in the design of dry spent fuel storage casks, consideration should be given to the compatibility of the dry storage cask systems and components with transportation and other activities related to the removal of the stored spent fuel from the reactor site for ultimate disposition by the Department of Energy.

§ 72.238 Issuance of an NRC Certificate of Compliance.

A Certificate of Compliance for a cask model will be issued by NRC on a finding that

(a) The criteria in § 72.236(a) through (i) are met; and

(b) The applicant certifies that each cask will be fabricated, inspected, and tested in accordance with § 72.236(j) and (l).

§ 72.240 Conditions for spent fuel storage cask reapproval.

(a) The holder of a cask model Certificate of Compliance, a user of a cask model approved by NRC, and representatives of cask users may apply for a cask model reapproval.

(b) Application for reapproval of a cask model must be submitted 3 years prior to the date that the Certificate of Compliance for that model expires. The application must be accompanied by a safety analysis report (SAR). The new SAR may reference the SAR originally submitted for the cask model.

(c) A cask model will be reapproved if conditions in § 72.238 are met, including demonstration that storage of spent fuel has not significantly, adversely affected systems and components important to safety.

PART 50 - DOMESTIC LICENSING OF PRODUCTION
AND UTILIZATION FACILITIES

4. The authority citation of Part 50 continues to read as follows:
Authority: Secs. 102, 103, 104, 105, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended, 1244, 1246 (42 U.S.C. 5841, 5842, 5846).

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Section 50.10 also issued under secs. 101, 185,

68 Stat. 936, 955, as amended (42 U.S.C. 2131, 2235); sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.23, 50.35, 50.55, and 50.56 also issued under sec. 185, 68 Stat. 955 (42 U.S.C. 2235). Sections 50.33a, 50.55a and Appendix Q also issued under sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.34 and 50.54 also issued under sec. 204, 88 Stat. 1245 (42 U.S.C. 5844). Sections 50.58, 50.91, and 50.92 also issued under Pub. L. 97-415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80-50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Section 50.103 also under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138). Appendix F also issued under sec. 187, 68 Stat. 955 (42 U.S.C. 2237).

For the purposes of sec. 223, Stat. 958, as amended (42 U.S.C. 2273); §§ 50.10(a), (b), and (c), 50.44, 50.46, 50.48, 50.54, and 50.80(a) are issued under sec. 161b, 68 Stat. 948, as amended (42 U.S.C. 2201(b)); §§ 50.10(b) and (c), and 50.54 are issued under sec. 161i, 68 Stat. 949, as amended (42 U.S.C. 2201(i)); and §§ 50.9, 50.55(e), 50.59(b), 50.70, 50.71, 50.72, 50.73, 50.78 are issued under sec. 161o, 68 Stat. 950, as amended (42 U.S.C. 2201(o)).

5. In § 50.72, a new paragraph is added to read as follows:

§ 50.72 Immediate notification requirements for operating nuclear power reactors.

* * * * *

(b) ***

* * * * *

(2) ***

(vii)(A) Any instance in which a significant defect in a system or component important to safety is discovered in, or (B) any instance in which there is a significant reduction in the confinement system effectiveness of, any cask used to store spent fuel under § 72.210 of this chapter.

* * * * *

PART 170 - FEES FOR FACILITIES AND MATERIALS LICENSES AND OTHER REGULATORY SERVICES UNDER THE ATOMIC ENERGY ACT OF 1954, AS AMENDED

6. The authority citation for Part 170 continues to read as follows:
AUTHORITY: 31 U.S.C. 9701, 96 Stat. 1051; sec. 301, Pub. L. 92-314, 86 Stat. 222 (42 U.S.C. 2201w); sec. 201, 88 Stat. 1242, as amended (42 U.S.C. 5841).

7. In § 170.31, a new category 13 is added and Footnotes 1(b), (c), and (d) are amended to read as follows:

§ 170.31 Schedule of fees for materials licenses and other regulatory services, including inspections.

* * * * *

Category of materials licenses and type of fee ¹			Fee ^{2, 3}
*	*	*	*
13.	A.	Spent fuel storage cask Certificate of Compliance	
		Application	\$150
		Approvals	Full Cost
		Amendments, Revisions and Supplements	Full Cost
		Reapproval	Full Cost
	B.	Inspections of spent fuel storage cask Certificate of Compliance	
		Routine	Full Cost
		Nonroutine	Full Cost
	C.	Inspections of storage of spent fuel under §72.210	
		Routine	Full Cost
		Nonroutine	Full Cost

¹Types of fees - ***

* * * * *

(b) License/approval fees - For new licenses and approvals issued in fee Categories 1A and 1B, 2A, 4A, 5B, 10A, 10B, 11, 12, and 13 the recipient shall pay the license or approval fee as determined by the Commission in accordance with § 170.12(b), (e), and (f).

(c) Renewal/reapproval fees - Applications for renewal of materials licenses and approvals must be accompanied by the prescribed renewal fee for each category, except that applications for renewal of licenses and approvals in fee Categories 1A and 1B, 2A, 4A, 5B, 10A, 10B, 11, and 13 must be accompanied by an application fee of \$150, with the balance due

upon notification by the Commission in accordance with the procedures specified in § 170.12(d).

(d) Amendment fees - Applications for amendments must be accompanied by the prescribed amendment fees. An application for an amendment to a license or approval classified in more than one category must be accompanied by the prescribed amendment fee for the category affected by the amendment unless the amendment is applicable to two or more fee categories in which case the amendment fee for the highest fee category would apply, except that applications for amendment of licenses and approvals in fee Categories 1A and 1B, 2A, 4A, 5B, 10A, 10B, 11, 12, and 13 must be accompanied by an application fee of \$150 with the balance due upon notification by the Commission in accordance with § 170.12(c).

An application for amendment to a materials license or approval that would place the license or approval in a higher fee category or add a new fee category must be accompanied by the prescribed application fee for the new category.

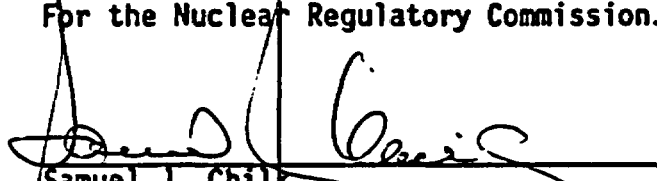
An application for amendment to a license or approval that would reduce the scope of a licensee's program to a lower fee category must be accompanied by the prescribed amendment fee for the lower fee category.

Applications to terminate licenses authorizing small materials programs, when no dismantling or decontamination procedure is required, shall not be subject to fees.

* * * * *

Dated at Rockville, Maryland, this 28th day of April, 1989.

For the Nuclear Regulatory Commission.


Samuel J. Chilk,
Secretary of the Commission.



REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 3.61 (Task CE.306-4)

STANDARD FORMAT AND CONTENT FOR A TOPICAL SAFETY ANALYSIS REPORT FOR A SPENT FUEL DRY STORAGE CASK

USNRC REGULATORY GUIDES

Regulatory Guides are issued to describe and make available to the public methods acceptable to the NRC staff of implementing specific parts of the Commission's regulations, to delineate techniques used by the staff in evaluating specific problems or postulated accidents, or to provide guidance to applicants. Regulatory Guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission.

This guide was issued after consideration of comments received from the public. Comments and suggestions for improvements in these guides are encouraged at all times, and guides will be revised, as appropriate, to accommodate comments and to reflect new information or experience.

Written comments may be submitted to the Regulatory Publications Branch, DFIPS, ARM, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

The guides are issued in the following ten broad divisions:

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INTRODUCTION

Section 72.24, "Contents of Application: Technical Information," of 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," specifies that a safety analysis report (SAR) must be included with an application for a license under Part 72. A safety evaluation specifically for the cask to be used for storing spent fuel must be provided in the SAR for an ISFSI license because the cask is important to safety.

This regulatory guide provides guidance on the format and content of a topical safety analysis report (TSAR) for a spent fuel storage cask. There is no regulation that requires the submittal of a TSAR for spent fuel storage casks. However, if a TSAR on a specific spent fuel storage cask is evaluated by the NRC staff and accepted for referencing in licensing actions, appropriate sections of the TSAR could be referenced in other submittals. Applicants for a specific license under Part 72 could reference the appropriate information in their SAR, thus significantly reducing their time, effort, and costs.

Casks used for storage of spent fuel on a reactor site could be those used for shipping spent fuel or could be those designed for storage only. Casks used for shipping must be licensed under 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," which requires stringent quality assurance and cask testing. Casks used for shipping could also be approved for storage of spent fuel if their safety is demonstrated.

Purpose and Applicability

Not all subjects identified in this regulatory guide may be applicable to a specific cask design, e.g., for casks with solid neutron shield material, guidance related to liquid shielding material and its retention. Additional or different subjects may be applicable to some cask designs. The information identified represents the minimum that should be provided, recognizing that not all information requested necessarily applies to all specific designs.

Additional information may be requested for NRC staff review of the TSAR. If any changes in the cask design are made after submittal of the TSAR but before the NRC has completed its review, the TSAR should be updated. This ensures that the TSAR as accepted for referencing reflects the actual cask design.

The TSAR should serve as the principal technical communication between the cask vendor and the NRC. It establishes the design of the cask and the plans for its use.

The TSAR should contain an analysis of the cask design in terms of potential hazards and the means employed to protect against these hazards, including the associated margins of safety. This includes evaluating:

1. The cask's vulnerability to accidents during operations and from natural phenomena,
2. Radiation shielding,
3. Confinement and control of radioactive materials,

4. Reliability of the systems that are important to safety, and
5. The radiological impact associated with normal operations, off-normal conditions, and accidents.

The TSAR should demonstrate the degree of skill, care, and effort used in planning all aspects of the project. A complete, in-depth analysis of all subjects in the report should be provided.

The TSAR should set forth a description, including all pertinent technical information, and a safety assessment of the design bases of the cask and its components in sufficient detail that the NRC staff can make an independent evaluation of the cask. A detailed description of the quality assurance program associated with the design and fabrication activities, including identification of the components and systems to which it will be applied, should be provided.

An analysis of anticipated operations, including consideration of human error, should be presented in the appropriate sections of the TSAR covering:

1. Preoperational tests,
2. Anticipated operations and maintenance,
3. Potential limiting conditions on the use of the cask, including limiting specifications on the fuel to be stored, and
4. Considerations for facilitating decommissioning.

There are no regulatory requirements for a TSAR on spent fuel storage casks. However, the information in the TSAR is intended to be used in the SAR required of license applicants under 10 CFR Part 72. The information collection requirements of 10 CFR Part 72 have been cleared under OMB Clearance No. 3150-0132.

Supplemental Information

Because of the diversity of design possibilities for a spent fuel dry storage cask, the initial enrichment, burnup, cooling time, condition (e.g., cladding integrity) of the fuels to be stored, and other storage conditions, detailed information not explicitly identified in this Standard Format may be included in the TSAR. The following are examples:

1. Information regarding assumed analytical models or calculational methods for design alternatives used by the vendor or its agents, with particular emphasis on rationale and detailed examples used to develop the bases for criticality safety,
2. Technical information in support of new design features of the cask,
3. Reports furnished by consultants.

Proprietary Information

Proprietary information should be submitted separately. When submitted, it should be clearly identified and accompanied with detailed reasons and justifications for requesting its being withheld from public disclosure as specified by § 2.790, "Public Inspections, Exemptions, Requests for Withholding," of 10 CFR Part 2, "Rules of Practice for Domestic Licensing Proceedings."

Style and Composition

To the extent possible, the TSAR should follow the numbering system of this Standard Format at least down to the level of subsections, e.g., 3.1.2 Design Criteria.

References, including author, date, and page number, should be cited within the text if important to the meaning of the statement. References should appear either as footnotes to the page where referenced or at the end of each chapter.

A table of contents and an index of key items should be included in each volume of the TSAR.

For numerical values, the number of significant figures given should reflect the accuracy and precision to which the number is known. When appropriate, estimated limits of errors or uncertainty should be given.

Abbreviations should be consistent throughout the TSAR and should be consistent with generally accepted usage. Any abbreviations, symbols, or special terms not in general usage or that are unique to the proposed cask design should be defined when they first appear in the TSAR.

Graphic Presentations

Graphic presentations such as drawings, diagrams, sketches, and tables should be employed when the information may be presented more adequately or conveniently by such means. Due concern should be taken to ensure that all information so presented is legible, that symbols are defined, and that drawings are not reduced to the extent that visual aids are necessary to interpret pertinent items of information. These graphic presentations should be located in the section in which they are primarily referenced.

Physical Specifications

Paper size

Text pages: 8-1/2 x 11 inches.

Drawings and graphics: 8-1/2 x 11 inches; however, a larger size is acceptable provided the finished copy when folded does not exceed 8-1/2 x 11 inches.

Paper stock and ink. Suitable quality in substance, paper color, and ink density for handling and reproduction by microfilming or image-copying equipment.

Page margins. A margin of no less than 1 inch should be maintained on the top, bottom, and binding side of all pages submitted.

Printing

Composition: text pages should be single spaced.

Type face and style: should be suitable for microfilming or image-copying equipment.

Reproduction: may be mechanically or photographically reproduced. All pages of text should be printed on both sides with image printed head-to-head.

Binding. Pages should be punched for standard 3-hole loose-leaf binders.

Page numbering. Pages should be numbered with the digits corresponding to the chapter and first-level section numbers followed by a hyphen and a sequential number within the section, e.g., the third page in Section 4.1 of Chapter 4 should be numbered 4.1-3. Do not number the entire report sequentially. (Note that because of the small number of pages in this guide, this Standard Format is numbered sequentially throughout.)

Procedures for Updating or Revising Pages

Data and text should be updated or revised by replacing pages. "Pen and ink" or "cut and paste" changes should not be used.

To avoid confusion between original and updated material, each TSAR supplement should be dated and identified by its supplement number in the lower right-hand corner of the page. Each supplement should be accompanied by a supplement index, also dated and numbered, listing pages to be inserted or removed. The supplement index should identify pages containing new material by page number and the date of the new material.

1. GENERAL DESCRIPTION

Present, in narrative style, the purpose for and a general description of the storage cask. The information in this chapter should enable the reader to obtain a basic understanding of the cask and the protection afforded the public health and safety without having to refer to the subsequent chapters. This general description should enable the reader to follow the detailed chapters with better perspective and to recognize the relative safety importance of each individual item to the overall cask design.

1.1 Introduction

Present briefly the principal design features of the cask. Include a general description of the characteristics of the cask; the nominal capacity of the cask; and the type, form, quantity, and potential sources of the spent fuels to be stored.

1.2 General Description of the Storage Cask

1.2.1 Cask Characteristics

Summarize the principal characteristics of the cask. Include the gross weight, materials of construction, materials used as neutron absorbers and moderators, external dimensions and cavity size, internal and external structures, receptacles, valves, sampling ports, means of passive heat dissipation, volume and type of coolant, outer and inner protrusions, lifting devices, impact limiters if applicable, amount of shielding, pressure relief systems (if applicable), closures, means of confinement, model number, and a description of how individual casks will be identified. The confinement vessel should be clearly identified. Overall and cutaway sketches of the package should be included as part of the description.

If the cask is certified under 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," pertinent information should be provided in this section and details and copies of documents (drawings, etc.) referenced in the cask's Certificate of Compliance should be included in Section 1.5, Supplemental Data.

Drawings and specifications that clearly summarize the safety features considered in the analysis should be included in Section 1.5; for example, material lists, dimensions, and specifications for valves, gaskets, and welds should be included. Detailed construction drawings should not be included.

1.2.2 Operational Features

A discussion of anticipated operations involving the cask should be provided. It should include a schematic diagram showing instrumentation, valves, connections, piping, openings, seals, confinement boundaries, etc. This section should contain a suggested procedure for using the cask. The section should also contain a discussion of the design bases considered for preventing or mitigating the consequences of potential human error.

1.2.3 Cask Contents

State the type and quantity of radionuclides that may be stored in the cask. Include the chemical and physical form, material density, moderator ratios, configurations required for nuclear safety, maximum amount of decay heat, maximum pressure buildup in the inner container, and any other loading restrictions. Estimate the type and quantity of radionuclides available for release.

1.3 Identification of Agents and Contractors

Identify the prime agents or contractors for the design, fabrication, and testing of the cask. All principal consultants and outside service organizations, including those providing quality assurance services, should be identified. The division of responsibility between the designer and fabricator should be delineated.

1.4 Generic Cask Arrays

Identify generic arrays of multiple casks in storage, such as in-line, square, vertical, and horizontal. The information should be sufficient to enable an evaluation of a particular array with regard to thermal and radiological conditions both within the array and at site boundaries.

1.5 Supplemental Data

This section should include detailed information describing the cask and its operational features and contents. Include dimensional drawings, detailed operational schematics, and loading configurations.

2. PRINCIPAL DESIGN CRITERIA

Principal design criteria for the storage cask should be presented in this section. The bases for these criteria should also be discussed. The NRC staff analyzes these design criteria for adequacy in evaluating the cask TSAR. Changes in the criteria are not anticipated after the TSAR is accepted for referencing. Therefore, the criteria selected should encompass all considerations for design alternatives that the vendor may choose.

2.1 Spent Fuel To Be Stored

A detailed description of the physical, thermal, and radiological characteristics of the spent fuels that the cask is designed to store should be provided. Include spent fuel characteristics such as initial enrichment, specific power, burnup, decay time, and heat generation rates.

2.2 Design Criteria for Environmental Conditions and Natural Phenomena

Identify and quantify environmental conditions and natural phenomena used for designing the cask, and identify those components of the cask that are identified as important to safety. Meteorological conditions, flooding, seismicity, ambient temperature range, and peak insolation should be considered, as appropriate. Data and design assumptions should be included.

2.2.1 Tornado and Wind Loadings

2.2.1.1 Applicable Design Parameters. The design parameters applicable to the design tornado such as translational velocity, rotational velocity, and the design pressure differential as well as the associated time interval should be specified. Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," contains information that may be helpful.

2.2.1.2 Determination of Forces on Structures. Describe the methods used to convert the tornado and wind loadings into forces on the cask, including the distribution across the cask and the combination of applied loads. If factored loads are used, the basis for selection of the load factor used for tornado loading should be furnished.

2.2.1.3 Tornado Missiles. The dimensions, energy, velocity, and other parameters should be selected for a potential tornado-driven missile.* An analysis should be presented to show that the cask can withstand the impact of the missile without significantly impairing its confinement ability.

*Paragraph 4 in subsection III of Section 3.5.1.4, "Missiles Generated by Natural Phenomena," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," contains information that may be of value when developing these data. A copy of this section is available for inspection and copying for a fee at the NRC Public Document Room, 2120 L Street NW., Washington, DC, under file CE 306-4.

2.2.2 Water Level (Flood) Design

Discuss the applicability of effects from a probable maximum flood (PMF), and, if applicable, discuss the design loads from forces developed by the PMF, including water height and dynamic phenomena such as velocity. Reference the design criteria to PMF data.

2.2.2.1 Flood Elevations. The flood elevations used in the design of the cask for buoyancy and static water force effects should be provided.

2.2.2.2 Phenomena Considered in Design Load Calculations. The phenomena (e.g., flood current, wind wave, hurricane, or tsunami) considered if dynamic water force is a design load should be identified and discussed.

2.2.2.3 Flood Force Application. Describe the manner in which the forces and other effects resulting from flood loadings are applied.

2.2.2.4 Flood Protection. Describe the flood protection measures for cask components that are important to safety.

2.2.3 Seismic Design

Discuss the applicability of effects from seismic events, and, if applicable, discuss the seismic design bases used in the design and fabrication of the cask to establish the required parameters that envelop credible conditions under which the cask may operate. Sufficient detail should be presented to allow an independent evaluation of the criteria selected. If necessary, the following format is suggested.

2.2.3.1 Input Criteria. This section should discuss the input criteria for seismic design of the cask. If response spectral shapes other than those in Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," are proposed for design of the cask, these should be justified and the earthquake time functions or other data from which these were derived should be presented. For damping values that are used in the design, submit a comparison of the response spectra derived from the time history and the design response spectra. The system period intervals at which the spectra values were calculated should be identified.

2.2.3.2 Seismic-System Analyses. This section should discuss the seismic-system analyses applicable to cask components that are important to safety. The following specific information should be included:

1. Seismic Analysis Methods. For all cask components that are important to safety, the applicable methods of seismic analysis should be identified. Applicable descriptions (or sketches) of typical mathematical models used to determine the response should be specified.

2. Methods to Determine Overturning Moments. A description of the dynamic methods and procedures used to determine cask overturning moments should be provided, including a description of the procedures used to account for vertical earthquake effects. Establish the minimum overturning moment that could cause tipping of the cask.

2.2.4 Snow and Ice Loadings

Describe criteria used to ensure that the effects of snow and ice loads can be accommodated, particularly with respect to thermal and stress transients that may be induced.

2.2.5 Combined Load Criteria

For combined loads, describe the criteria selected to provide mechanical and structural integrity. The loads and loading combinations to which the cask is designed should be defined, including the load factors selected for each load component in which a factored load approach is used. The design approach used with the loading combination and any load factors should be specified. The design loading combinations used to examine the effects on localized areas such as penetrations, structural discontinuities, and local areas of high thermal gradients should be provided, together with time-dependent loading such as thermal effects, effects of creep and shrinkage, and other related effects.

2.3 Safety Protection Systems

2.3.1 General

Identify special considerations in the design that may result from an evaluation of cask operating conditions (e.g., loading, unloading, transport) to ensure the long-term safety and confinement of the stored fuel.

2.3.2 Protection by Multiple Confinement Barriers and Systems

2.3.2.1 Confinement Barriers and Systems. Discuss each method of confinement that will be used to ensure that there will be no uncontrolled release of radioactivity to the environment. Include for each:

1. Criteria for protection against any postulated off-normal operations, internal change, or external natural phenomena,
2. Design criteria selected for backup confinement, and
3. Delineation of the extent to which the design is based on achieving the lowest practical level of radioactive releases from the cask.

2.3.2.2 Cask Cooling. Describe the criteria selected for providing suitable passive cooling of the cask under normal and off-normal conditions.

2.3.3 Protection by Equipment and Instrumentation Selection

2.3.3.1 Equipment. Design criteria for cask equipment that is important to safety should be provided. This would include any equipment used to protect against or mitigate the effects of the release of radioactive material.

2.3.3.2 Instrumentation. Discuss the design bases and design criteria for instrumentation selected with particular emphasis on features to provide reliability and testability.

2.3.4 Nuclear Criticality Safety

Supply pertinent design bases to show the appropriate safety margins that ensure that a subcritical situation exists under all credible conditions.

2.3.4.1 Control Methods for Prevention of Criticality. Present the methods to be used to ensure that subcritical situations are maintained in storage under the worst credible conditions.

2.3.4.2 Error Contingency Criteria. To support the above information, define the error contingency criteria selected.

2.3.4.3 Verification Analyses. Present the criteria for verifying models or computer programs used in criticality analyses. Revision 2 of Regulatory Guide 3.4, "Nuclear Criticality Safety in Operations with Fissionable Materials at Fuels and Materials Facilities," provides information on this subject.

2.3.5 Radiological Protection

Based on anticipated storage system operations, an estimate of collective doses (in person-rem) per year, including estimated collective doses associated with cask operation, maintenance, repair, and decommissioning, should be presented.

2.3.6 Fire and Explosion Protection

Provide the design criteria selected to ensure that all safety functions will successfully withstand credible fire and explosion conditions.

2.4 Decommissioning Considerations

Discuss the consideration given in the design of the cask to decommissioning. Examples of subjects to be covered are (1) discussion of neutron activation of the cask and fuel basket materials, (2) provisions for the decontamination and removal of potentially contaminated components, and (3) discussion of the decommissioning processes.

3. STRUCTURAL EVALUATION

This chapter of the TSAR should identify, describe, discuss, and analyze the principal structural engineering design of the cask's components and systems that are important to safety. The design bases for design criteria should be discussed.

3.1 Structural Design

3.1.1 Discussion

Identify the principal structural members and systems that are important to safety, such as the confinement vessel and closure devices. Reference the location of these items on drawings, and discuss their design and performance.

3.1.2 Design Criteria

Describe the load combinations and factors that serve as design criteria. For each of these criteria, state the maximum allowable stresses and strains (as a percentage of the yield or ultimate values) for ductile failure, and describe how the other structural failure modes (e.g., brittle fracture, fatigue, buckling) are considered. If different design criteria are to be allowed in various parts of the cask for different conditions, the appropriate values for each should be indicated. Include the criteria that will be used for impact evaluation. Identify all codes and standards that are used to determine material properties, design limits, or methods of combining loads and stresses. In cases of deviation from standard codes, or when certain components are not covered by standard codes, provide a detailed description of the design criteria used as substitutes for such codes.

3.2 Weights and Centers of Gravity

Provide the total weight of the cask and contents. Tabulate the weights of major individual subassemblies so that the sum of the parts equals the total of the cask. Locate the center of gravity of the cask and any other centers of gravity referred to in the application. It is not necessary to include the calculations made to determine these values, but a sketch or drawing that clearly shows the individual subassembly referred to and the reference point for locating its center of gravity should be included.

3.3 Mechanical Properties of Materials

Provide mechanical properties of materials used in the structural evaluation. These may include yield stress, ultimate stress, modulus of elasticity, ultimate strain, Poisson's ratio, density, and coefficient of thermal expansion. If impact limiters are used, include either a compression stress-strain curve for the material or the force-deformation relationship for the limiter, as appropriate. For materials that are subjected to dynamic loadings or elevated temperatures, the appropriate mechanical properties under these conditions should be specified to the extent used in the structural evaluation. The source of all information in this section should be clearly and specifically referenced as to publication and page number. If material properties were determined by testing, the test procedure, conditions, and measurements should

be described in sufficient detail to allow the staff to conclude that the results are valid.

3.4 General Standards for Casks

3.4.1 Chemical and Galvanic Reactions

Discuss possible chemical, galvanic, or other reactions in the cask or between the cask and its contents. For each component material of the cask, list all chemically or galvanically dissimilar materials with which it has contact. Indicate any specific measures that have been taken to prevent contact or reaction between materials, and discuss the effectiveness of such measures.

3.4.2 Positive Closure

Describe and discuss the cask closure system in sufficient detail to show that it cannot be inadvertently opened. This demonstration should include covers, valves, or any other access that must be closed during normal operation.

3.4.3 Lifting Devices

Identify all devices and attachments that can be used to lift the cask or its lid. Show by testing or analysis that these devices, if structurally part of the cask, are capable of supporting three times the weight of the loaded cask without generating stress in any part of the cask in excess of its yield strength. Provide drawings or sketches that show the location and construction of these items. Determine the effects of the forces imposed by lifting on vital cask components, including the interfaces between the lifting devices and other cask surfaces. Documented values of the yield stresses of the materials should be used as the criteria to demonstrate compliance with this section.

3.4.4 Heat

The thermal evaluation for the cask should be reported in Section 4.4.

3.4.4.1 Summary of Pressures and Temperatures. Summarize pressures and temperatures (determined in the thermal evaluation in Chapter 4) that will be used to perform the calculations required for Sections 3.4.4.2, 3.4.4.3, and 3.4.4.4.

3.4.4.2 Differential Thermal Expansion. Calculate the circumferential and axial deformations and stresses (if any) that result from differential thermal expansion. Consider steady-state and transient conditions. These calculations must be sufficiently comprehensive to demonstrate cask integrity under normal operating conditions.

3.4.4.3 Stress Calculations. Calculate the stress from the combined effects of thermal gradients, pressure, and mechanical loads. Provide sketches or free body diagrams that show the configuration and dimensions of the members or systems being analyzed, and locate the points at which the stresses are being

calculated. The analysis should consider whether repeated cycles of thermal loadings, together with other loadings, will cause fatigue failure or extensive accumulations of deformation.

3.4.4.4 Comparison with Allowable Stresses. Make the appropriate stress combinations and compare the resulting stresses with the design criteria in Section 3.1.2. Show that all the requirements specified in the regulations have been satisfied.

3.4.5 Cold

Assess the cask for the effects of a steady-state ambient temperature. Consider both material properties and possible freezing of liquids under this condition. For components of the cask that are important to safety, identify the resulting temperatures and the ways they affect the operation of the cask. Brittle fracture should be considered.

3.5 Fuel Rods

When fuel rod cladding is considered in the design criteria for confinement of radioactive material under normal or accident conditions, provide an analysis or test results showing that the cladding will maintain its integrity. Show that fuel rod assemblies can be handled during loading and unloading of the cask without compromising the confinement of radioactive materials.

3.6 Supplemental Data

This section should include information such as justifications of assumptions or analytical procedures; test results; photographs; computer program descriptions, documentation, benchmarks, and input/output; reference lists; and applicable pages from referenced documents.

4. THERMAL EVALUATION

This chapter of the TSAR should identify, describe, discuss, and analyze the thermal engineering design of the cask structures, components, and systems that are important to safety. The bases for the design criteria should be discussed.

4.1 Discussion

Describe the significant thermal design features and operating characteristics of the cask. The operation of all subsystems (e.g., cooling systems, expansion tanks) should be discussed. Summarize the significant results of the thermal analysis or tests and the implication of these results on the overall design. State the minimum and maximum decay heat loads assumed in the thermal evaluation.

4.2 Summary of Thermal Properties of Materials

List the thermal properties of all materials used in the thermal evaluation. References for the data cited should be provided in Section 4.5.

4.3 Specifications for Components

Include the specifications for cask components. For example, in the case of relief devices or rupture discs, the operating pressure range and temperature limits should be included. Data should be supplied in support of technical specifications and should be presented in detail in Section 4.5.

4.4 Thermal Evaluation for Normal Conditions of Storage

4.4.1 Thermal Model

4.4.1.1 Analytical Model. Describe the analytical thermal model in detail. The model should include data on gaskets, valves, fuel assemblies, and the overall containment. Modeling assumptions should be fully justified.

4.4.1.2 Test Model. Describe the tests, models, and procedures used to correlate the test data to the thermal environment for normal conditions. Temperature data should be taken from gaskets, valves, confinement boundaries, and other areas of the cask.

4.4.2 Maximum Temperatures

Provide the maximum temperature distribution for the cask for normal conditions of storage, including the spent fuel, confinement vessel, shielding material, gaskets, valves, etc.

4.4.3 Minimum Temperatures

Provide the minimum temperature distribution for the cask for normal conditions of storage. This evaluation should include the minimum decay heat load that will be experienced. If a decay heat load greater than zero is

required for safe operation, assurance of that heat load must be provided. The temperatures of significant components such as gaskets and valves should be reported.

4.4.4 Maximum Internal Pressures

The conditions within the range of normal conditions of storage that result in the worst internal pressures or the worst combination of thermal loadings should be identified. The internal pressures for the conditions should be determined. The evaluation should consider the effects of phase change, gas generation, chemical decomposition, etc.

4.4.5 Maximum Thermal Stresses

Determine the conditions within the range of normal conditions of storage that result in the worst combination of thermal gradient and isothermal stresses. Provide the resulting temperature distribution.

4.4.6 Evaluation of Cask Performance for Normal Conditions of Storage

Evaluate the cask performance, including system and subsystem operations, for normal conditions of storage with respect to the results of the thermal analyses or tests performed. Take into account significant conditions to be found in the ranges bounded by the minimum and maximum ambient temperatures and minimum and maximum decay heat loads. Compare the results with allowable limits of temperature, pressure, etc., for the cask components. Designate the information that is to be used in other chapters of the TSAR. Present the information in summary tables along with discussions as appropriate.

4.5 Supplemental Data

This section should include data in support of thermal evaluations such as justifications of assumptions or analytical procedures; test results; photographs; computer program descriptions, documentation, benchmarks, and input/output; and applicable pages from referenced documents.

5. SHIELDING EVALUATION

This chapter should identify, describe, discuss, and analyze the shielding design of the cask and its systems that are important to safety. The bases for the design criteria should be discussed.

5.1 Discussion and Results

Discuss the significant shielding design features of the cask and the adequacy of the shielding. Table 5-1 (in Section 5.2.2) should be completed.

5.2 Source Specification

The gamma and neutron source terms used in the shielding analysis and the spent fuel loadings that would produce these values should be stated.

5.2.1 Gamma Source

State the quantity of radioactive material assumed as contents of the cask, and tabulate the gamma decay source strength (MeV/sec and photons/sec) as a function of photon energy. Describe in detail the method used to determine the gamma source strength and distribution.

5.2.2 Neutron Source

State the quantity of radioactive material assumed as contents of the cask, and tabulate the neutron source strength (neutron/sec) as a function of energy. Describe in detail the method used to determine the neutron source strength and distribution.

5.3 Model Specification

In this section, describe the model that was used in the shielding evaluation.

5.3.1 Description of the Radial and Axial Shielding Configurations

Include sketches (to scale) and dimensions of the radial and axial shielding materials. Dose point locations for the various calculations exterior to the package should be shown relative to the source regions in the sketches supplied. Voids or irregularities not taken into account in the model should be discussed in detail, showing that the resultant dose rates are conservative. Differences between the models for normal conditions and accident conditions should be clearly identified.

5.3.2 Shield Regional Densities

The material densities (g/cm^3) and the atomic number densities (atoms/barn-cm) for constituent nuclides of all materials used in the calculational models for the normal and accident analyses should be given in this section. The sources of the data should be referenced; provide a copy of the data for uncommon shielding material in Section 5.5.

TABLE 5-1

**SUMMARY OF MAXIMUM DOSE RATES
(mrem/hr)**

	Cask Surface			1 Meter (3 Feet) from Surface of Cask		
	Sides	Top	Bottom	Sides	Top	Bottom
Normal Conditions						
Gamma						
Neutron						
Total						
Postulated Accident Conditions						
Gamma						
Neutron						
Total						

5.4 Shielding Evaluation

Provide a general description of the basic method used to determine the gamma and neutron dose rates at the selected points outside the cask for both normal conditions of storage and accident conditions. This should include a description of the spatial source distribution and any computer program used, with its referenced documentation. The basic input parameters should be discussed in detail. The basis for selecting the program, attenuation and removal cross sections, and buildup factors should be provided. Flux-to-dose-rate conversion factors as a function of energy should be tabulated. Data are to be supported by appropriate references.

5.5 Supplemental Data

This section should include supplemental data such as justifications of assumptions or analytical procedures; test results; photographs; computer program descriptions, documentation, benchmarks, and input/output; and applicable pages from referenced documents.

6. CRITICALITY EVALUATION

This chapter should identify, describe, discuss, and analyze the criticality safety physics used for design of the cask and its components and systems that are important to safety.

6.1 Discussion and Results

Discuss the significant criticality design features of the cask and the adequacy of the criticality evaluation. A summary of the criticality evaluation should be included in this section.

6.2 Spent Fuel Loading

Provide a summary table showing the maximum spent fuel loading and spent fuel parameters for the cask.

6.3 Model Specification

This section should contain a description of the model used in the criticality evaluation.

6.3.1 Description of Computational Model

Dimensioned sketches (to scale) or the geometric model used in the calculations should be presented. The sketches should identify the materials used in all regions of the model. Differences between the actual cask configuration and the model should be identified, and the model should be shown to be conservative. Differences between the models for normal conditions of storage and accident conditions should be clearly identified.

6.3.2 Cask Regional Densities

The material densities (g/cm^3) and the atomic number densities (atoms/barn-cm) for constituent nuclides of all materials used in the calculational models for the normal and accident analyses are to be given in this section. Fissionable isotopes are to be considered at their most credible reactivity. Masses for materials in all regions should be consistent with atomic number densities and volumes occupied.

6.4 Criticality Calculation

This section should contain descriptions of the calculational or experimental methods used to determine the nuclear reactivity for the maximum fuel loading intended to be stored in the cask.

6.4.1 Calculational or Experimental Method

A description of the method used to calculate the effective multiplication constant of the cask under normal conditions of storage and accident conditions should be provided. This should include a description of the computer program and neutron cross sections used with their referenced documentation. The basis for selecting the program and cross sections should be discussed.

If an experimental method was used to determine the compliance of the cask with criticality requirements, include a complete description of the method and a discussion demonstrating that the method conservatively takes into account both normal and accident conditions of storage for the cask.

6.4.2 Fuel Loading or Other Contents Loading Optimization

Demonstrate that the maximum reactivity for fuel loading or other contents loading has been evaluated for both a single cask and arrays of casks for normal and accident conditions. Approximations, boundary conditions, calculational convergence criteria, and cross-section adjustments should be itemized and discussed.

6.4.3 Criticality Results

Results of the reactivity calculations establishing the most reactive configurations for a single cask and arrays of casks for both normal conditions of storage and accident conditions should be displayed in tabular and graphic form. Justification should be provided for any interpolations and extrapolations. A discussion of the validity and conservatism of the analysis should be provided, including the bias established with the benchmark calculations in Section 6.5.

6.5 Critical Benchmark Experiments

This section should provide justification for and show the validity of the calculational method and neutron cross-section values used in the analyses. Revision 2 of Regulatory Guide 3.4, "Nuclear Criticality Safety in Operations with Fissionable Materials at Fuels and Materials Facilities," provides information on validation of criticality calculations.

6.5.1 Benchmark Experiments and Applicability

Provide a general discussion of selected critical benchmark experiments that are to be analyzed using the method and cross sections given in Section 6.4.1. The applicability of the benchmarks in relation to the cask design and its contents should be shown. References giving documentation on these benchmarks should be provided.

6.5.2 Results of the Benchmark Calculations

Provide the results of the benchmark calculations. Establish and provide a discussion of any calculation bias.

6.6 Supplemental Data

This section should include information such as justifications of assumptions and analytical procedures; test results; photographs; computer program descriptions, documentation, benchmarks, and input/output; and applicable pages from referenced documents.

7. CONFINEMENT

This chapter should identify and discuss cask confinement for normal conditions of storage. The bases for the design criteria should be discussed.

7.1 Confinement Boundary

Identify the confinement boundary of the cask.

7.1.1 Confinement Vessel

A summary of design specifications for the confinement vessel should be provided.

7.1.2 Confinement Penetrations

Identify all penetrations in the primary confinement boundary. Provide a summary of the performance specifications for all components that penetrate the confinement boundary.

7.1.3 Seals and Welds

Identify all seals and welds that affect cask confinement. Provide a summary of the fabrication specifications for these seals and welds, including tests and inspections required for quality assurance.

7.1.4 Closure

Identify the closure devices used for the confinement vessel. Specify the initial bolt torque that will be required to maintain a positive seal during normal conditions of storage and accident conditions.

7.2 Requirements for Normal Conditions of Storage

Summarize the pertinent results of the analyses or tests performed to demonstrate the cask confinement under normal storage conditions.

7.2.1 Release of Radioactive Material

Show that there will be no direct release of particulate radioactive material from the confinement vessel. Describe the means for detecting radioactivity in the confinement vessel without disrupting the sealing system.

7.2.2 Pressurization of Confinement Vessel

Any vapors or gases that could form in the confinement vessel should be identified. Show that any increase in pressure or explosion within the confinement vessel that is caused by these vapors or gases would not result in a radioactive release that exceeds the limits of 10 CFR Part 72.

7.3 Confinement Requirements for Hypothetical Accident Conditions

7.3.1 Fission Gas Products

Estimate the maximum quantity of fission gas products that could be available for release from the confinement vessel under hypothetical accident conditions.

7.3.2 Release of Contents

Show that there can be no significant release of radioactive materials exceeding site boundary requirements.

7.4 Supplemental Data

This section should include supporting information and analyses.

8. OPERATING PROCEDURES

This chapter should describe operating procedures recommended for the preparation for and performance of the processes of loading, testing, storing, unloading, and maintaining the function of the cask. The discussion of these procedures, including appropriate tests, should be presented sequentially in the anticipated order of performance. At a minimum, this chapter should demonstrate that the procedures, if properly followed, will ensure that occupational radiation exposures will be maintained as low as is reasonably achievable and that there is reasonable assurance that the health and safety of the public will be protected. A copy of the recommended procedures and tests should be provided to each user of the cask.

8.1 Procedures for Loading the Cask

The section should include descriptions of recommended procedures for inspections, tests, and special preparations of the cask for loading. If applicable, present a detailed description of the procedures used to ensure that fluids such as shield water and primary coolants fill their respective cavities, in compliance with the design specifications. Also provide details of the procedures used to remove residual moisture from cavities designed to be dry. Provide an evaluation of the effectiveness of such procedures.

8.2 Procedures for Unloading the Cask

This section should include descriptions of recommended procedures for inspections, tests, and special preparations of the cask for unloading. As applicable, provide the procedures used to ensure safe removal of fission gases, contaminated coolant, and solid contaminants. Describe any required cooldown procedure and, if applicable, show that it does not affect reuse of the cask.

8.3 Preparation of the Cask

This section should contain a description of recommended procedures for inspections, tests, and special preparations of the cask necessary to ensure that the cask is properly loaded, closed, decontaminated to prevent the spread of contamination, and delivered to a transport vehicle in such a condition that subsequent transport will not impair the effectiveness of the cask to perform its required safety function.

8.4 Supplemental Data

This section should include supporting documentation, detailed discussions and analyses of procedures, and graphic presentations.

9. ACCEPTANCE CRITERIA AND MAINTENANCE PROGRAM

This chapter should contain a discussion of the cask acceptance criteria and the cask maintenance program. The bases for acceptance criteria should be discussed.

9.1 Acceptance Criteria

Discuss the analyses or tests to be performed prior to the first use of the cask.

9.1.1 Visual Inspection

The visual inspections to be performed and the intended purpose for each inspection should be discussed. The acceptance criteria for each of these inspections, as well as the action to be taken if noncompliance is encountered, should be provided.

9.1.2 Structural

Describe the analyses or tests to be performed for structural acceptance. Present the acceptance criteria and describe the action to be taken when the prescribed criteria are not met. An estimate of the sensitivity of the tests should be provided and the basis for this estimate should be given.

9.1.3 Leak Tests

Describe the leak tests to be performed. Leak tests should be performed on the confinement vessel as well as auxiliary equipment that is important to safety, such as shield tanks. Describe the acceptance criteria and the action to be taken if the criteria are not met. Estimate the sensitivity of these leak tests and give the basis for the estimate.

9.1.4 Components

Analyses and/or tests for components that are important to safety should be discussed. If a characteristic (for instance, longevity) cannot be tested, an upper limit should be justified. Acceptance criteria and actions to be taken if the criteria are not met (e.g., replacement) should be presented.

9.1.4.1 Valves, Rupture Discs, and Fluid Transport Devices. These components should be analyzed or tested under the most severe service conditions for which acceptable performance is assumed for the cask design. When the tests are presumed to adversely affect the continued performance of a component, the results of tests on components of the same model and type may be substituted.

9.1.4.2 Gaskets. Gaskets should be tested under conditions simulating the most severe service conditions under which the gaskets are assumed to perform. Since these acceptance tests may degrade the performance of either the gasket under test or the cask into which it is assembled or both, the tests are not necessarily performed on gaskets or casks to be put into service. The simulation system should ensure adequate representation of those conditions that would prevail if the actual system were used in the test. The manufacturer of the gasket should maintain a quality assurance program adequate to ensure

that acceptance testing of a given gasketing device is equivalent to acceptance testing of all gaskets of that model supplied by that manufacturer.

9.1.4.3 Miscellaneous. Any component not listed in Sections 9.1.4.1 and 9.1.4.2 whose failure would impair cask effectiveness should be analyzed or tested under the most severe conditions for which it was designed. Since acceptance tests may degrade the performance of either the component under test or the system into which it is assembled or both, the tests are not necessarily performed on components or systems to be put into service. The analyses should ensure adequate representation of those conditions that would prevail if the actual system were in use. Furthermore, the manufacturer of the component should maintain a quality assurance program adequate to ensure that acceptance testing of a given component device is equivalent to acceptance testing of all devices of that model supplied by that manufacturer.

9.1.5 Shielding Integrity

Discuss the analyses or tests to be performed to ensure adequate shielding for both gamma and neutron sources. The acceptance criteria as well as the action to be taken if the criteria are not met should be described.

9.1.6 Thermal Acceptance

Discuss the analyses or tests to verify that each cask will perform, within some defined variance, in accordance with the results of the thermal analyses or tests for normal conditions of storage.

9.1.6.1 Discussion of Test Setup. Describe the analysis or test setup. The description should include heat sources, instrumentation, and schematics showing thermocouple and heat source locations as well as the placement of other test equipment. Estimate test sensitivities based on instrumentation, test item, and environmental variations.

9.1.6.2 Test Procedure. Discuss the procedures used in all tests and describe the data-recording method. Report the frequency of data recording during the test. The criteria used to define the steady-state (thermal equilibrium) condition of the test item should also be discussed.

9.1.6.3 Acceptance Criteria. Discuss the thermal acceptance criteria and the method employed to compare any acceptance test results with predicted thermal performance. Discuss the action to be taken if the thermal acceptance criteria are not met.

9.2 Maintenance Program

This section should describe the recommended maintenance program that will ensure continued performance of the storage cask. The program should include recommended testing, inspection, and replacement schedules, as well as criteria for replacement and repair of components and subsystems on an as-needed basis.

9.2.1 Subsystems Maintenance

Describe the tests and replacement schedules recommended for storage cask subsystems (e.g., neutron shield tanks) whose inadequate performance could result in the inability of the cask to perform its safety function. Justify the schedules established, using tests or manufacturers' data.

9.2.2 Valves, Rupture Discs, and Gaskets on Containment Vessel

Specify the test and replacement schedule to be used for these components. Justify the recommended schedules.

10. RADIATION PROTECTION

This chapter of the TSAR should provide information on methods for radiation protection and on estimated radiation exposures to operating personnel during anticipated operation (including maintenance, surveillance, inspections, and instrument calibration). This chapter should also include information on planned procedures and programs and the techniques and practices that should be employed by the applicant in meeting the standards of 10 CFR Part 20 for protection against radiation. Reference to other chapters for information needed in this chapter should be specific.

10.1 Ensuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)

10.1.1 Policy Considerations

Discuss ALARA policies on occupational radiation exposure with respect to cask design, inspections, repair, and maintenance.

10.1.2 Design Considerations

Describe considerations of cask design that are directed toward ensuring that occupational radiation exposure is ALARA. Describe how experience from past designs is used to develop improved design for ensuring that incidents of contamination are minimized. Describe how the design is directed toward reducing (1) the need for maintenance of equipment, (2) radiation levels, and (3) time spent on maintenance.

10.1.3 Operational Considerations

Identify and describe procedures and methods that could be used to ensure that occupational radiation exposure is ALARA.

10.2 Radiation Protection Design Features

Describe cask design features used for ensuring a high degree of integrity for the confinement of radioactive materials.

Provide scale drawings of the cask showing the locations of all sources described in Section 5.2. Include specific activity, physical and chemical characteristics, and expected radioactivity concentrations. Other information provided should include the potential radiation dose rate for the storage area, maintenance and repair activities, and estimates of radioactive materials that might be discharged during storage. Reference may be made to specific sections of the TSAR for this information.

10.3 Estimated Onsite Collective Dose Assessment

Provide the assumed annual occupancy times, including the anticipated maximum total hours per year for any individual and total person-hours per year for all personnel for each radiation area during normal operation and anticipated operational occurrences. Also provide the objectives and criteria for estimated dose rates in various areas and an estimate of the annual collective person-rem

doses associated with major functions such as handling and storage operations, ancillary activities (e.g., offgas handling), maintenance, decontamination, and inservice inspection. Supply the bases, models, and assumptions for the above values. State assumptions made in determining the time-related dose rates.

11. ACCIDENT ANALYSES

The evaluation of cask safety is accomplished in part by analyzing the response of the cask to postulated off-normal and accident events. Consider (1) minimizing the causes of such events, (2) identification and mitigation of the consequences of accidents, and (3) the ability to cope with each situation if it occurs. These analyses are an important aspect of the reviews made by the NRC in evaluating a cask design.

In previous chapters, features important to safety have been identified and discussed. The purpose of this chapter is to identify and analyze a range of credible off-normal and accident occurrences and their causes and potential consequences. For each situation, reference should be made to the appropriate chapter and section that describe the design considerations to prevent or mitigate the accident. The analyses should relate incidents to anticipated cask use at nuclear power reactor sites and spent fuel storage systems.

ANSI/ANS-57.9-1984, "Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type),"* defines four categories of events that provide a means of establishing design requirements to satisfy safety criteria. The first design event is associated with normal operation. The second and third design events apply to events that are expected to occur during the life of the installation. The fourth design event is concerned with natural phenomena or low-probability events. Regulatory Guide 3.60, "Design of an Independent Spent Fuel Storage Installation (Dry Storage)," endorses ANSI/ANS-57.9-1984 for use in the design of an ISFSI that uses a dry environment as a mode of storage subject to certain caveates.

11.1 Off-Normal Operations

In this section, design events pertaining to off-normal operation for expected operational occurrences are considered. They may include equipment malfunctions, radiation leakage, or human error. In general, the consequences of the events discussed in this section would not have a significant effect beyond the cask storage area. The following format should be used to present the desired detail.

11.1.1 Event

Identify the event, including the portion of the cask involved, the type of failure or malfunction, the component, system or systems involved, and the effects, consequences, and corrective actions.

11.1.1.1 Postulated Cause of the Event. Describe the sequence of occurrences that could initiate the event under consideration and the bases upon which credibility or probability of each occurrence in the sequence is determined.

*Copies may be obtained from the American Nuclear Society, 555 N. Kensington Avenue, La Grange Park, IL 60525.

The following should be provided:

1. Starting conditions and assumptions;
2. A step-by-step sequence of the course of each accident, identifying all protection systems required to function at each step; and
3. Identification of any personnel actions necessary.

The discussion should show the extent to which protective systems should function, the effect of failure of protective functions, and the credit taken for cask safety features. The performance of backup protection systems during the entire course of the event should be analyzed. The analysis given should permit an independent evaluation of the adequacy of the protection system as related to the event under study. The results can be used to determine which components, systems, and controls are important to safety and what actions are required under the anticipated operational occurrence.

11.1.1.2 Detection of Event. Discuss the means or methods, such as visual or audible alarms or routine inspections performed on a stated frequency, to be provided to detect the event. Provide for each an assessment of response time.

11.1.1.3 Analysis of Effects and Consequences. Analyze the effects of the event, particularly any radiological consequences. The analysis should:

1. Show the methods, assumptions, and conditions used in estimating the course of events and the consequences,
2. Identify the time-dependent characteristics and release rate of radioactive materials within the confinement system that could escape to the environment, and
3. Describe the margin of protection provided by whatever system is depended on to limit the extent or magnitude of the consequences. Explain how the cask components and their materials of construction provide the needed safety margins. Provide data to support conclusions regarding design assumptions.

11.1.1.4 Corrective Actions. For each event, give the corrective actions necessary to return to a normal situation.

11.1.2 Radiological Impact from Off-Normal Operations

The capability of the cask to operate safely within the range of anticipated operating variations, malfunctions of equipment, and human error should be shown. The information may be presented in tabular form with the situations analyzed listed in one column and other columns that identify:

1. Estimated doses (in person-rem),
2. Method or means available for detecting the situations,
3. Causes of the situation,
4. Corrective actions, and
5. Effects and consequences.

11.2 Accidents

An analysis of potential accidents to the cask (e.g., free fall, overturn, fire) should be presented. Include any credible incident that could potentially result in a dose of >25 mrem beyond a postulated controlled area. If there are no such credible potential accidents, provide the rationale for such a statement. Such analyses should address situations wherein direct radiation or radioactive materials may be released in such quantity as to endanger personnel within the controlled area. Events that could occur during the cask lifetime, e.g., earthquakes or other low-probability events, should be included. Design events of the third and fourth types defined in ANSI/ANS-57.9-1984 should be included in this section.

The following format should be used to provide the desired detail.

11.2.1 Analysis of Accidents

Identify the accident, the portion of the cask involved, and the type of accident. Discuss each accident sequentially (e.g., 11.2.2, 11.2.3 ...).

11.2.1.1 Cause of Accident. For each accident analyzed, describe and list the sequence of events leading to the initiation of the accident. Identify the type of event such as natural phenomenon, human error, component malfunction, or component failure. Include an estimate of probability and how this probability estimate was determined.

11.2.1.2 Accident Analysis. Analyze the effects of each accident, particularly any radiological consequences. Show the methods, assumptions, and conditions used in estimating the consequences, the recovery from the consequences, and the steps used to mitigate each accident. Assess the consequences of the accident to persons and property on the site.

In addition to the assumptions and conditions employed in the course of events and consequences, provide information on the following:

1. The mathematical or physical models employed in accident analyses. Include a description of each simplification introduced to perform the analyses. Identify the bases for the models used with specific reference to:

- a. The distribution and fractions of the radioactive material inventory assumed to be released from the cask,
- b. The concentrations of airborne radioactive materials in the confinement atmosphere and buildup during the postaccident time intervals analyzed, and
- c. The conditions considered in the analyses such as meteorology, topography, and combinations of adverse conditions.

2. Identification of any digital computer program or analog simulation used in the analysis, with principal emphasis on a detailed description of the input data and the extent or range of variables investigated. This information should include figures showing the analytical models, flow path identification, actual computer listings, and complete listings of input data.

3. The time-dependent characteristics, activity, and release rate of transmissible radioactive materials that could escape to the environment via leakages in the confinement boundaries.

4. The considerations of uncertainties in calculational methods, equipment performance, instrumentation response characteristics, or other indeterminate effects that should be taken into account in the evaluation of the results.

5. The conditions and assumptions associated with the events analyzed, including any reference to published data or research and development investigations in substantiation of the assumed or calculated conditions.

6. The extent of system interdependency (confinement systems and other engineered safety features) contributing directly or indirectly to controlling or limiting leakages from the confinement systems.

7. The results and consequences derived from each analysis and the margin of protection provided by whatever system is depended on to limit the extent or magnitude of the consequences.

11.2.1.3 Accident Dose Calculations. For each accident analyzed, provide and discuss the results of conservative calculations of potential integrated whole-body and critical-organ doses to an individual from exposure to radiation as a function of distance and time after the accident. Discuss the results and consequences derived from the analysis and the margin of protection provided by whatever system is depended on (i.e., remains operative) to limit the extent or magnitude of the consequences.

12. OPERATING CONTROLS AND LIMITS

Throughout the previous sections of this regulatory guide, the need to identify safety limits, limiting conditions, and surveillance requirements has been indicated. It is from such information that the cask operating controls, limits, and supporting bases should be developed. These limits should be defined and proposed as the operating controls and limits for the cask in the TSAR.

12.1 Proposed Operating Controls and Limits

Identify and justify the selection of those variable conditions and limits based on the design criteria of the cask or determined, as a result of safety assessment and evaluation, to be probable subjects of operating controls and limits for the cask. The operating controls and limits should be complete; i.e., to the fullest extent possible, numerical values and other pertinent data should be provided, including the support for selection of the technical and operating conditions. For each control or limit, reference the applicable sections and develop, through analysis and evaluation, the details and bases for the control or limit. Operating controls and limits should be proposed in the TSAR and accepted by NRC review and evaluation.

Each cask should have technical specifications, limiting conditions for operation, design features, and surveillance requirements. Operating controls and limits should be proposed in the TSAR along with an analyses of the bases for the technical specifications and a description of anticipated surveillance requirements.

12.1.1 Content of Operating Controls and Limits

Operating controls and limits should include both technical and administrative matters on those features of the cask that are important to safety (e.g., spent fuel loadings, operating variables, or components). In addition, operating controls and limits should address the attainment of ALARA levels of releases and exposures.

12.1.2 Bases for Operating Controls and Limits

When an operating control and limit has been selected, the basis for its selection and its significance to the safety of the operation should be described. This can be done in a summary statement of the technical and operational considerations justifying the selection. The TSAR should fully develop the details of these bases through analysis and evaluation. The format for presenting operating controls and limits assumes importance since the collection of controls and limits and their written bases form a document that delineates those features and actions important to the safety of operation, the reasons for their importance, and their relationships to each other.

12.2 Development of Operating Controls and Limits

Refer to § 72.44, "License Conditions," of 10 CFR Part 72 for guidance on the categories of activities and conditions requiring operating controls and limits.

12.2.1 Functional and Operating Limits, Monitoring Instruments, and Limiting Control Settings

Controls or limits in this category apply to operating variables that are important to safety and that are observable and measurable (e.g., temperatures within the cask or evidence of confinement leakage). Control of such variables is directly related to the performance and integrity of equipment and confinement barriers.

12.2.2 Limiting Conditions for Operation

This category of operating controls and limits covers two general classes, (1) equipment and (2) technical conditions and characteristics of the cask necessary for continued operation.

12.2.2.1 Equipment. Operating controls and limits should establish the lowest acceptable level of performance for a cask system or component and the minimum number of components or the minimum portion of the system that should be operable or available.

12.2.2.2 Technical Conditions and Characteristics. Technical conditions and characteristics should be stated in terms of allowable quantities, e.g., storage temperatures, radioactivity levels in gas samples, area radiation levels, and allowable configurations of equipment and spent fuel assemblies during operations. Specify the allowable quantities associated with limiting conditions. Specific definitions should be provided for limiting conditions even if they appeared in previous chapters.

12.2.3 Surveillance Specifications

Operating limits and technical specifications should be developed and presented for anticipated normal, off-normal, and accident operating conditions. Recommended surveillance procedures, including tests, calibrations, and inspections, should be provided to cask users to verify availability and performance of systems and components that are important to safety. These surveillance specifications should be described in this section.

12.2.4 Design Features

These operating controls and limits should cover design characteristics of special importance to each of the physical barriers and to maintenance of safety margins in the cask design. The principal objective of this category is to control changes in the design of essential equipment.

12.2.5 Suggested Format for Operating Controls and Limits

1. Title:
2. Specification: (e.g., maximum radiation level at any surface)

3. Applicability: The systems or operations to which the control or limit applies should be clearly defined.

4. Objective: The reasons for the control or limit and the specific unsafe conditions it is intended to prevent.

5. Action: What is to be done if the control or limit is exceeded; clearly define specific actions.

6. Surveillance Requirements: What maintenance and tests are to be performed and when.

7. Bases: The TSAR should contain pertinent information and an explicit detailed analysis and assessment supporting the choice of the item and its specific value or characteristics. The basis for each control or limit should contain a summary of the information in sufficient depth to indicate the completeness and validity of the supporting information and to provide justification for the control or limit. The following subjects may be appropriate for discussion in the bases section:

a. Technical Basis. The technical basis is derived from technical knowledge of the process and its characteristics and should support the choice of the particular variable as well as the value of the variable. The results of computations, experiments, or judgments should be stated, and analysis and evaluation should be summarized.

b. Equipment. If a safety limit is protected by or closely related to certain equipment, such a relationship should be noted, and the means by which the variable is monitored and controlled should be stated.

For controls or limits in categories referenced in Sections 12.2.2 and 12.2.3, the bases are particularly important. The function of the equipment and how and why the requirement is selected should be noted here. In addition, the means by which surveillance is accomplished should be noted. If surveillance is required periodically, the basis for frequency of required action should be given.

c. Operation. The margins and the bases that relate to the safety limits and normal operation should be stated. The roles of operating procedures and of protective systems in guarding against exceeding a limit or condition should be stated. Include a brief discussion of such factors as expected system responses, operational transients, and malfunctions. References to related limits should be made.

13. QUALITY ASSURANCE

Subpart G of Part 72 requires that a quality assurance (QA) program be established, maintained, and executed for structures, systems, and components important to safety. Cask systems and components that are important to safety should be identified in the TSAR. The QA program should be applied to design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, and modification of cask systems and components identified as important to safety. The applicable QA criteria should be executed to an extent that is commensurate with their importance to safety.

A QA program that meets the applicable criteria in Appendix B to 10 CFR Part 50 and that has been accepted by the NRC will be acceptable if it is established, maintained, and executed with regard to the design, testing, fabrication, and repair of the spent fuel storage cask. Prior to first use, the applicant should notify the Director, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555, of its intent to apply its previously accepted QA program to spent fuel storage casks. The applicant should identify the program by date of submittal, docket number, and date of NRC acceptance.

A branch technical position entitled "Quality Assurance Programs for Independent Spent Fuel Storage Installations (ISFSI) 10 CFR 72"* has been adopted by the NRC staff for implementing review of quality assurance programs submitted by applicants. This document could also be applied to a QA program for spent fuel storage casks.

*A copy of this branch technical position is available for inspection and copying for a fee at the NRC Public Document Room, 2120 L Street, NW., Washington, DC, under file CE 306-4. Single copies may be obtained by writing to the Fuel Cycle Safety Branch, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

VALUE/IMPACT STATEMENT

A draft value/impact statement was published with the proposed version of this guide (Task CE 306-4) when the draft guide was published for public comment in April 1986. No changes to the value/impact statement were necessary, so a separate value/impact statement for the final guide has not been prepared. A copy of the draft value/impact statement is available for inspection and copying for a fee at the Commission's Public Document Room at 2120 L Street NW., Washington, DC, under Task CE 306-4.

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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

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CHAIRMAN

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

DOE
Dry
Cask

March 1, 1989

The Honorable Donna Fitzpatrick
Acting Secretary of Energy
United States Department of Energy
1000 Independence Avenue, S.W.
Washington, D.C. 20585

Dear Madam Secretary:

I am responding to Mr. Samuel Rousso's (OCRWM) January 23, 1989 request for the Nuclear Regulatory Commission's (NRC's) comments on the Department of Energy's (DOE's) "Final Version Dry Cask Storage Study" (DOE/RW-0196). We have reviewed the final text and find that our November 18, 1988 comments on the draft version of the study have been accommodated. Moreover, we believe that the final version of the study remains a well-balanced presentation of the spent fuel storage requirements, the in-pool consolidated fuel storage and dry storage technologies available to address those requirements in at-reactor storage, and the impacts and costs of such storage.

We are pleased that DOE has responded positively to our concern about the need to ensure the compatibility of various steps in the storage, transport, and disposal of spent fuel to enhance the safety and efficiency of fuel handling. The Commission encourages DOE to actively pursue the commitment that it has made in its final study to accomplish resolution of this matter, both through its own actions and in concert with industry. The Commission itself will continue to support such efforts.

I hope that our comments on this report have been helpful. If you have any questions, please contact me or Mr. Robert M. Bernero, Director of NRC's Office of Nuclear Material Safety and Safeguards (telephone number 492-3352).

Sincerely,

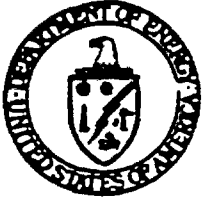
Original signed by
Lando W. Zech, Jr.
Lando W. Zech, Jr.

~~89-3174164~~

cc: Samuel Rousso, OCRWM/DOE

ECY 89-51

OFFICE	SECY	OCM	OCM				
SURNAME	A. Butts	[Signature]	[Signature]				
DATE	2/28/89	3/1/89	3/1/89				



Department of Energy
Washington, DC 20585

JAN 4 1989

Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

In response to your letter of November 18, 1988, which provided comments on the "Initial Version Dry Cask Storage Study" (DOE/RW-0196), I would like to thank you for your overall assessment of the document. We attempted to make it a straightforward, responsible, technical document and were gratified by your response. We will, of course, be responding to all of your comments in the comment response section of the final report. However, in consideration of the significance of the comment in the second paragraph of your letter, concerning compatibility of the various steps in the spent fuel management process, I want to give you an early indication of our current thinking.

The Department agrees that all of the steps in the spent fuel management process should be coordinated to enhance the safety and efficiency of the operations and plans to increase its efforts to ensure that this coordination takes place. This has already been recognized by both the Department and the nuclear utilities, and elements of this general coordination issue have already been identified as topics to be addressed through the process for resolving issues concerning the standard contract for disposal of spent fuel. This contract establishes, among other things, the contractual terms and conditions for the waste acceptance process.

The issue resolution process associated with the contract is a mechanism for identifying and ultimately overcoming obstacles to the effective and efficient implementation of the contract. The issue resolution process was described in the June 1988 issue of the "Annual Capacity Report" (DOE/RW-0191) and is commonly referred to as "the ACR issue resolution process." The Department intends to discuss with the utilities at the next meeting in the ACR issue resolution process the general coordination issue that you have raised, to identify opportunities for and the timing of steps to address any coordination elements that are not already being addressed. Any elements of the general coordination issue that are not appropriate for resolution through the ACR issue resolution process will be taken up separately by the Department working with the utilities, through the auspices of the Edison Electric Institute's Utility Nuclear Waste and Transportation Program (successor to the separate Utility Nuclear Waste Management Group and Transportation Group).

Enclosure

As described in the Department's "Initial Version Dry Cask Storage Study," several different technologies for expanding at-reactor storage are in various stages of development. No single technology is likely to meet the requirements of all the utilities. Furthermore, the utilities believe that they need to retain the flexibility to choose the option that best suits their requirements, while choosing systems that incorporate compatibility elements that are jointly developed based on system requirements.

As more information is developed about each of the technologies, it will be appropriate and natural to consider certain features or interfaces within each of the technology categories for compatibility with the Federal Waste Management System. These features or interfaces could include items such as dimensions, weights, payloads, materials, heat and radiation limits, and handling features.

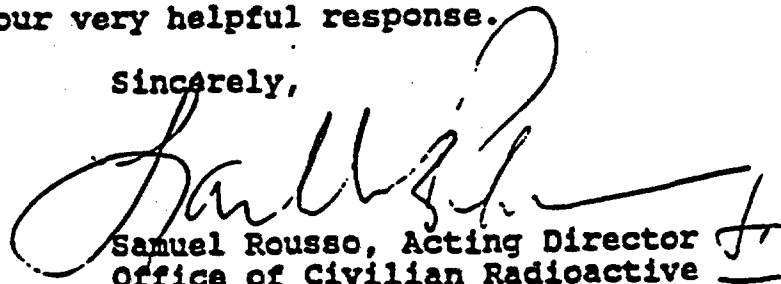
The compatible elements of each of the major types of technology can then serve as focal points for combined Federal and utility efforts to ensure that the various technologies interface satisfactorily with the Federal Waste Management System. Such a process will allow time for major programmatic issues (such as the need for a Monitored Retrievable Storage facility) to be resolved, more information concerning the various at-reactor storage technologies to be developed, and the waste disposal package design and handling requirements to become better defined.

In the meantime, the Department's near-term shipping cask designs will be oriented toward development of the basic designs needed to ship the bulk of the fuel (i.e., maintaining compatibility with the 80 percent that are intact spent fuel assemblies stored in water filled pools). The Department's longer term shipping cask design efforts will consider modifications to these basic designs to maximize the efficiency of handling as much of the remaining 20 percent of the spent fuel as possible, primarily the portion whose storage incorporates the compatibility features discussed above.

Finally, the Department will separately consider how to handle any spent fuel that is stored in ways that do not comply with the compatible techniques established in cooperation with the utility industry, recognizing that such fuel may be subject to delayed acceptance under the terms of the standard disposal contract.

Again, let me thank you for your very helpful response.

Sincerely,



Samuel Rousso, Acting Director
Office of Civilian Radioactive
Waste Management