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Mr. Elmer D. Gates
Birdsboro Ferrocast, Inc.
200 North Furnace Street
P.O. Box 490
Birdsboro, PA 19508-0490

Dear Mr. Gates:

I am responding to your letter to Senator Arlen Specter, dated April 22, 1991, in which you discussed the capability of Birdsboro Ferrocast, Inc. to supply casks for the storage and transportation of high-level radioactive waste. Although the U.S. Nuclear Regulatory Commission (NRC) is not authorized to develop technology for high-level waste storage or transportation, it does have responsibility for approving or disapproving storage cask design when a reactor operator or a vendor submits an application for review. NRC issues two types of approval for storage designs for use at the sites of commercial nuclear power reactors, an "approval for reference" and a "certificate of compliance," depending on the type of approval requested.

For an approval for reference, a designer submits a "Topical Safety Analysis Report" (TSAR) for a specific storage design. If NRC grants an approval for reference based on the TSAR, a utility may reference that TSAR in a site-specific license application without further description of the design. We have enclosed the "Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask" for your information (Enclosure 1).

The second type of approval NRC issues for storage casks is the certificate of compliance. This type of approval allows a utility to use a certified cask to store spent fuel at its reactor site without additional site specific approvals. Certificates of compliance are issued under Subpart K of 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (Enclosure 2). To date, NRC has issued certificates of compliance for four cask designs.

The current status of NRC's dry spent fuel storage approvals is indicated in Enclosure 3. You may wish to contact individual cask vendors to inform them of your company's capabilities, since they are ultimately responsible for selecting cask manufacturers.

As with spent fuel storage casks, NRC reviews and certifies the design of casks for transportation of radioactive materials including high-level radioactive waste. The NRC requirements in this area are described in 10 CFR Part 71,

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"Packaging and Transportation of Radioactive Material" (Enclosure 4). The U.S. Department of Energy (DOE) has the responsibility for developing the transportation system for high-level waste. Currently, DOE and its contractors are developing several different transportation cask designs. To date, no cask design has been submitted to NRC for review and certification. For further information regarding DOE's program in the area of transportation, we suggest you contact Dr. John W. Bartlett, Director, Office of Civilian Radioactive Waste Management, U.S. Department of Energy, 1000 Independence Ave. S.W., Washington, D.C. 20585.

I trust that this information is helpful in understanding NRC licensing requirements for storage and transportation casks and in identifying individuals involved in this area.

Sincerely,

(Signed) Robert M. Bernero

Robert M. Bernero, Director
Office of Nuclear Material
Safety and Safeguards

Enclosures: As stated

cc: Sen. Arlen Specter

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U.S. NUCLEAR REGULATORY COMMISSION February 1989

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

ENCLOSURE 1

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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 by 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.

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*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 Calculational 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 situat'on 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 $\geq$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.

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*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.

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

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PART 72

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

55 FR 29181

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)).

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Subpart A—General Provisions

§ 72.1 Purpose.

The regulations in this part establish requirements, procedures, and criteria for the issuance of licenses to receive, transfer, and possess power reactor spent fuel and other radioactive materials associated with spent fuel storage in an independent spent fuel storage installation (ISFSI) and the terms and conditions under which the Commission will issue such licenses, including licenses to the U.S. Department of Energy (DOE) for the provision of not more than 1900 metric tons of spent fuel storage capacity at facilities not owned by the Federal Government on January 7, 1983 for the Federal interim storage program under Subtitle B—Interim Storage Program of the Nuclear Waste Policy Act of 1982 (NWPA). The regulations in this part also establish requirements, procedures, and criteria for the issuance of licenses to DOE to receive, transfer, package, and possess power reactor spent fuel, high-level radioactive waste, and other radioactive materials associated with the spent fuel and high-level radioactive waste storage, in a monitored retrievable storage installation (MRS).

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§ 72.2 Scope.

(a) Except as provided in § 72.6(b), licenses issued under this part are limited to the receipt, transfer, packaging, and possession of:

(1) Power reactor spent fuel to be stored in a complex that is designed and constructed specifically for storage of power reactor spent fuel aged for at least one year, and other radioactive materials associated with spent fuel storage in an independent spent fuel storage installation (ISFSI); or

(2) Power reactor spent fuel to be stored in a monitored retrievable storage installation (MRS) owned by DOE that is designed and constructed specifically for the storage of spent fuel aged for at least one year, high-level radioactive waste that is in a solid form, and other radioactive materials associated with spent fuel or high/level radioactive waste storage.

The term "Monitored Retrievable Storage Installation" or "MRS," as defined § 72.3, is derived from the NWPA and includes any installation that meets this definition.

(b) The regulations in this part pertaining to an independent spent fuel storage installation (ISFSI) apply to all persons in the United States, including persons in Agreement States. The regulations in this part pertaining to a monitored retrievable storage installation (MRS) apply only to DOE.

(c) The requirements of this regulation are applicable, as appropriate, to both wet and dry modes of storage of (1) spent fuel in an independent spent fuel storage installation (ISFSI) and (2) spent fuel and solid high-level radioactive waste in a monitored retrievable storage installation (MRS).

(d) Licenses covering the storage of spent fuel in an existing spent fuel storage installation shall be issued in accordance with the requirements of this part as stated in § 72.40, as applicable.

(e) As provided in section 135 of the Nuclear Waste Policy Act of 1982, Pub. L. 97–425, 96 Stat. 2201 at 2232 (42 U.S.C. 10155) the U.S. Department of Energy is not required to obtain a license under the regulations in this part to use available capacity at one or more facilities owned by the Federal Government on January 7, 1983, including the modification and expansion of any such facilities, for the storage of spent nuclear fuel from civilian nuclear power reactors.

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§ 72.3 Definitions.

As used in this part:

"Act" means the Atomic Energy Act of 1954 (68 Stat. 919) including any amendments thereto.

"Affected Indian tribe" means any Indian tribe—

(1) Within whose reservation boundaries a monitored retrievable storage facility is proposed to be located;

(2) Whose federally defined possessory or usage rights to other lands outside of the reservation's boundaries arising out of congressionally ratified treaties may be substantially and adversely affected by the locating of such a facility: *Provided*, That the Secretary of the Interior finds, upon the petition of the appropriate governmental officials of the tribe, that such effects are both substantial and adverse to the tribe.

"Affected unit of local government" means any unit of local government with jurisdiction over the site where an MRS is proposed to be located.

"As low as is reasonably achievable" (ALARA) means as low as is reasonably achievable taking into account the state of technology, and the economics of improvement in relation to—

(1) Benefits to the public health and safety,

(2) Other societal and socioeconomic considerations, and

(3) The utilization of atomic energy in the public interest.

"Atomic energy" means all forms of energy released in the course of nuclear fission or nuclear transformation.

"Byproduct material" means any radioactive material (except special nuclear material) yielded in or made radioactive by exposure to the radiation incident to the process of producing or utilizing special nuclear material.

"Commencement of construction" means any clearing of land, excavation, or other substantial action that would adversely affect the natural environment of a site, but does not mean:

(1) Changes desirable for the temporary use of the land for public recreational uses, necessary borings or excavations to determine subsurface materials and foundation conditions, or other preconstruction monitoring to establish background information related to the suitability of the site or to the protection of environmental values;

(2) Construction of environmental monitoring facilities;

(3) Procurement or manufacture of components of the installation; or

(4) Construction of means of access to the site as may be necessary to accomplish the objectives of paragraphs (1) and (2) of this definition.

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"Commission" means the Nuclear Regulatory Commission or its duly authorized representatives.

"Confinement systems" means those systems, including ventilation, that act as barriers between areas containing radioactive substances and the environment.

"Controlled area" means that area immediately surrounding an ISFSI or MRS for which the licensee exercises authority over its use and within which ISFSI or MRS operations are performed.

"Decommission" means to remove (as a facility) safely from service and reduce residual radioactivity to a level that permits release of the property for unrestricted use and termination of license.

"Design bases" means that information that identifies the specific functions to be performed by a structure, system, or component of a facility and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be restraints derived from generally accepted "state of the art" practices for achieving functional goals or requirements derived from analysis (based on calculation or experiments) of the effects of a postulated event under which a structure, system, or component must meet its functional goals. The values for controlling parameters for external events include: (1) Estimates of severe natural events to be used for deriving design bases that will be based on consideration of historical data on the associated parameters, physical data, or analysis of upper limits of the physical processes involved and (2) estimates of severe external man-induced events to be used for deriving design bases that will be based on analysis of human activity in the region taking into account the site characteristics and the risks associated with the event.

"Design capacity" means the quantity of spent fuel or high-level radioactive waste, the maximum burnup of the spent fuel in MWD/MTU, the curie content of the waste, and the total heat generation in BTU per hour that the storage installation is designed to accommodate.

"DOE" means the U.S. Department of Energy or its duly authorized representatives.

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"Floodplain" means the lowland and relatively flat areas adjoining inland and coastal waters, including floodprone areas of offshore islands. Areas subject to a one percent or greater chance of flooding in any given year are included.

"High-level radioactive waste" or "HLW" means (1) the highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and (2) other highly radioactive material that the Commission, consistent with existing law, determines by rule requires permanent isolation.

"Historical data" means a compilation of the available published and unpublished information concerning a particular type of event.

"Independent spent fuel storage installation" or "ISFSI" means a complex designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage. An ISFSI which is located on the site of another facility may share common utilities and services with such a facility and be physically connected with such other facility and still be considered independent: Provided, that such sharing of utilities and services or physical connections does not: (1) Increase the probability or consequences of an accident or malfunction of components, structures, or systems that are important to safety; or (2) reduce the margin of safety as defined in the basis for any technical specification of either facility.

"Indian Tribe" means an Indian tribe as defined in the Indian Self Determination and Education Assistance Act (Pub. L. 93-638).

"Monitored Retrievable Storage Installation" or "MRS" means a complex designed, constructed, and operated by DOE for the receipt, transfer, handling, packaging, possession, safeguarding, and storage of spent nuclear fuel aged for at least one year and solidified high-level radioactive waste resulting from civilian nuclear activities, pending shipment to a HLW repository or other disposal.

"NEPA" means the National Environmental Policy Act of 1969 including any amendments thereto.

"NWPA" means the Nuclear Waste Policy Act of 1982 including any amendments thereto.

"Person" means—

(1) Any individual, corporation, partnership, firm, association, trust, estate, public or private institution, group, Government agency other than the Commission or the Department of Energy (DOE) except that the DOE shall be considered a person within the meaning of the regulations in this part to the extent that its facilities and activities are subject to the licensing and related regulatory authority of the Commission pursuant to section 202 of the Energy Reorganization Act of 1974, as amended (88 Stat. 1244), and Sections 131, 132, 133, 135, 137, and 141 of the Nuclear Waste Policy Act of 1982 (96 Stat. 2229, 2230, 2232, 2241);

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(2) Any State, any political subdivision of a State, or any political entity within a State;

(3) Any foreign government or nation, or any political subdivision of any such government or nation, or other entity; and

(4) Any legal successor, representative, agent, or agency of the foregoing.

"Population" means the people that may be affected by the change in environmental conditions due to the construction, operation, or decommissioning of an ISFSI or MRS.

"Region" means the geographical area surrounding and including the site, which is large enough to contain all the features related to a phenomenon or to a particular event that could potentially impact the safe or environmentally sound construction, operation, or decommissioning of an independent spent fuel storage or monitored retrievable storage installation.

"Reservation" means—

(1) Any Indian reservation or dependent Indian community referred to in clause (a) or (b) of section 1151 of title 18, United States Code; or

(2) Any land selected by an Alaska Native village or regional corporation under the provisions of the Alaska Native Claims Settlement Act (43 U.S.C. 1601 *et seq.*).

"Site" means the real property on which the ISFSI or MRS is located.

"Source material" means—

(1) Uranium or thorium, or any combination thereof, in any physical or chemical form or

(2) Ores that contain by weight one-twentieth of one percent (0.05%) or more of:

(i) Uranium,

(ii) Thorium, or

(iii) Any combination thereof.

Source material does not include special nuclear material.

"Special nuclear material" means—

(1) Plutonium, uranium-233, uranium enriched in the isotope 233 or in the isotope 235, and any other material which the Commission, pursuant to the provisions of section 51 of the Act, determines to be special nuclear material, but does not include source material; or

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(2) Any material artificially enriched by any of the foregoing but does not include source material.

"Spent Nuclear Fuel" or "Spent Fuel" means fuel that has been withdrawn from a nuclear reactor following irradiation, has undergone at least one year's decay since being used as a source of energy in a power reactor, and has not been chemically separated into its constituent elements by reprocessing. Spent fuel includes the special nuclear material, byproduct material, source material, and other radioactive materials associated with fuel assemblies.

"Structures, systems, and components important to safety" mean those features of the ISFSI or MRS whose function is:

(1) To maintain the conditions required to store spent fuel or high-level radioactive waste safely,

(2) To prevent damage to the spent fuel or the high-level radioactive waste container during handling and storage, or

(3) To provide reasonable assurance that spent fuel or high-level radioactive waste can be received, handled,

packaged, stored, and retrieved without undue risk to the health and safety of the public.

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§ 72.4 Communications.

Except where otherwise specified, all communications and reports concerning the regulations in this part and applications filed under them should be addressed to the Director, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Communications, reports, and applications may be delivered in person at the Commission's Offices at 11555 Rockville Pike, Rockville, Maryland, or at 2120 L Street NW., Washington, DC.

§ 72.5 Interpretations.

Except as specifically authorized by the Commission in writing, no interpretation of the meaning of the regulations in this part by an officer or employee of the Commission, other than a written interpretation by the General Counsel, will be recognized to be binding upon the Commission.

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§ 72.6 License required; types of licenses.

(a) Licenses for the receipt, handling, storage, and transfer of spent fuel or high-level radioactive waste are of two types: general and specific. Any general license provided in this part is effective without the filing of an application with the Commission or the issuance of a licensing document to a particular person. A specific license is issued to a named person upon application filed pursuant to regulations in this part.

(b) A general license is hereby issued to receive title to and own spent fuel or high-level radioactive waste without regard to quantity. Notwithstanding any other provision of this chapter, a general licensee under this paragraph is not authorized to acquire, deliver, receive, possess, use, or transfer spent fuel or high-level radioactive waste except as authorized in a specific license.

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(c) Except as authorized in a specific license and in a general license under subpart K of this part issued by the Commission in accordance with the regulations in this part, no person may acquire, receive, or possess—

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(1) Spent fuel for the purpose of storage in an ISFSI; or

(2) Spent fuel, high-level radioactive waste, or radioactive material associated with high-level radioactive waste for the purpose of storage in an MRS.

§ 72.7 Specific exemptions.

The Commission may, upon application by any interested person or upon its own initiative, grant such exemptions from the requirements of the regulations in this part as it determines are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest.

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§ 72.8 Denial of licensing by Agreement States.

Agreement States may not issue licenses covering the storage of spent fuel in an ISFSI or the storage of spent fuel and high-level radioactive waste in an MRS.

§ 72.9 Information collection requirements: OMB approval.

(a) The Nuclear Regulatory Commission has submitted the information collection requirements contained in this part to the Office of Management and Budget (OMB) for approval as required by the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 *et seq.*). OMB has approved the information collection requirements contained in this part under control number 3150–0132.

(b) The approved information collection requirements contained in this part appear in §§ 72.16, 72.22 through 72.34, 72.42, 72.44, 72.48 through 72.56, 72.62, 72.70 through 72.82, 72.90, 72.92, 72.94, 72.98, 72.100, 72.102, 72.104, 72.108, 72.120, 72.126, 72.140 through 72.176, 72.180 through 72.186, and 72.192.

§ 72.10 Employee protection.

(a) Discrimination by a Commission licensee, an applicant for a Commission license, or a contractor or subcontractor of a Commission licensee or applicant against an employee for engaging in certain protected activities is prohibited. Discrimination includes discharge and other actions that relate to compensation, terms, conditions, and privileges of employment. The protected activities are established in section 210 of the Energy Reorganization Act of 1974, as amended, and in general are related to the administration or enforcement of a requirement imposed under the Atomic Energy Act of 1954, as amended, or the Energy Reorganization Act.

(1) The protected activities include but are not limited to—

(i) Providing the Commission information about possible violations of requirements imposed under either of the above statutes;

(ii) Requesting the Commission to institute action against his or her employer for the administration or enforcement of these requirements; or

(iii) Testifying in any Commission proceeding.

(2) These activities are protected even if no formal proceeding is actually initiated as a result of the employee assistance or participation.

(3) This section has no application to any employee alleging discrimination prohibited by this section who, acting without direction from his or her employer (or the employer's agent), deliberately causes a violation of any requirement of the Energy Reorganization Act of 1974, as amended, or the Atomic Energy Act of 1954, as amended.

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(b) Any employee who believes that he or she has been discharged or otherwise discriminated against by any person for engaging in the protected activities specified in paragraph (a)(1) of this section may seek a remedy for the discharge or discrimination through an administrative proceeding in the Department of Labor. The administrative proceeding must be initiated within 30 days after an alleged violation occurs by filing a complaint alleging the violation with the Department of Labor, Employment Standards Administration, Wage and Hour Division. The Department of Labor may order reinstatement, back pay, and compensatory damages.

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(c) A violation of paragraph (a) or paragraph (g) of this section by a Commission licensee, an applicant for a Commission license, or a contractor or subcontractor of a Commission licensee or applicant may be grounds for—

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(1) Denial, revocation, or suspension of the license.

(2) Imposition of a civil penalty on the licensee or applicant.

(3) Other enforcement action.

(d) Actions taken by an employer, or others, which adversely affect an employee may be predicated upon nondiscriminatory grounds. The prohibition applies when the adverse action occurs because the employee has engaged in protected activities. An employee's engagement in protected activities does not automatically render him or her immune from discharge or discipline for legitimate reasons or from adverse action dictated by nonprohibited considerations.

(e)(1) Each licensee and each applicant shall post Form NRC-3, "Notice to Employees," on its premises. Posting must be at location sufficient to permit employees protected by this section to observe all copy on the way to or from their place of work. Premises must be posted no later than 30 days after an application is docketed and remain posted while the application is pending before the Commission, during the term of the license, and for 30 days following license termination.

(2) Copies of Form NRC-3 may be obtained by writing to the Regional

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Administrator of the appropriate U.S. Nuclear Regulatory Commission Regional Office listed in Appendix A, Part 73 of this chapter or the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

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(f) No agreement affecting the compensation, terms, conditions and privileges of employment, including an agreement to settle a complaint filed by an employee with the Department of Labor pursuant to section 210 of the Energy Reorganization Act of 1974, may contain any provision which would prohibit, restrict, or otherwise discourage, an employee from participating in protected activity as defined in paragraph (a)(1) of this section, including, but not limited to, providing information to the NRC on potential violations or other matters within NRC's regulatory responsibilities.

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§ 72.11 Completeness and accuracy of information.

(a) Information provided to the Commission by an applicant for a license or by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the applicant or the licensee shall be complete and accurate in all material respects.

(b) Each applicant or licensee shall notify the Commission of information identified by the applicant or licensee as having for the regulated activity a significant implication for public health and safety or common defense and security. An applicant or licensee violates this paragraph only if the applicant or licensee fails to notify the Commission of information that the applicant or licensee has identified as having a significant implication for public health and safety or common defense and security. Notification shall be provided to the Administrator of the appropriate Regional Office within two working days of identifying the information. This requirement is not applicable to information which is already required to be provided to the Commission by other reporting or updating requirements.

Subpart B—License Application, Form, and Contents

53 FR 43419

§ 72.16 Filing of application for specific license.

(a) *Place of filing.* Each application for a license, or amendment thereof, under this part should be filed with the Director, Division of Industrial and Medical Nuclear Safety, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Applications, communications, reports, and correspondence may also be delivered in person at the Commission's offices at 11555 Rockville Pike, Rockville, Maryland, or at the NRC Public Document Room, 2120 L Street NW., Washington, DC.

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(b) *Oath or affirmation.* Each application for a license or license amendment (including amendments to such applications), except for those filed by DOE, must be executed in an original signed by the applicant or duly authorized officer thereof under oath or affirmation. Each application for a license or license amendment (including amendments to such applications) filed by DOE must be signed by the Secretary of Energy or the Secretary's authorized representative.

(c) *Number of copies of application.* Each filing of an application for a license or license amendment under this part (including amendments to such applications) must include, in addition to a signed original, 15 copies of each portion of such application, safety analysis report, environmental report, and any amendments. Another 125 copies shall be retained by the applicant for distribution in accordance with instruction from the Director or the Director's designee.

(d) *Fees.* The application, amendment, and renewal fees applicable to a license covering the storage of spent fuel in an ISFSI are those shown in § 170.31 of this chapter.

(e) *Notice of docketing.* Upon receipt of an application for a license or license amendment under this part, the Director, Office of Nuclear Material Safety and Safeguards or the Director's designee will assign a docket number to the application, notify the applicant of the docket number, instruct the applicant to distribute copies retained by the applicant in accordance with paragraph (c) of this section, and cause a notice of docketing to be published in the Federal Register. The notice of docketing shall identify the site of the ISFSI or the MRS by locality and State and may include a notice of hearing or a notice of proposed action and opportunity for hearing as provided by § 72.46 of this part. In the case of an application for a license or an amendment to a license for an MRS, the Director, Office of Nuclear Material Safety and Safeguards, or the Director's designee, in accordance with § 72.200 of this part, shall send a copy of the notice of docketing to the Governor and legislature of any State in which an MRS is or may be located, to the Chief Executive of the local municipality, to the Governors of any contiguous States and to the governing body of any affected Indian tribe.

§ 72.18 Elimination of repetition.

In any application under this part, the applicant may incorporate by reference information contained in previous applications, statements, or reports filed with the Commission: *Provided*, That such references are clear and specific.

§ 72.20 Public inspection of application.

Applications and documents submitted to the Commission in connection with applications may be made available for public inspection in accordance with provisions of the regulations contained in Parts 2 and 9 of this chapter.

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§ 72.22 Contents of application: General and financial information.

Each application must state:

(a) Full name of applicant;

(b) Address of applicant;

(c) Description of business or occupation of applicant;

(d) If applicant is:

(1) An individual: Citizenship and age;

(2) A partnership: Name, citizenship, and address of each partner and the principal location at which the partnership does business;

(3) A corporation or an unincorporated association:

(i) The State in which it is incorporated or organized and the principal location at which it does business; and

(ii) The names, addresses, and citizenship of its directors and principal officers;

(4) Acting as an agent or representative of another person in filing the application: The identification of the principal and the information required under this paragraph with respect to such principal.

(5) The Department of Energy:

(i) The identification of the DOE organization responsible for the construction and operation of the ISFSI or MRS, including a description of any delegations of authority and assignments of responsibilities.

(ii) For each application for a license for an MRS, the provisions of the public law authorizing the construction and operation of the MRS.

(e) Except for DOE, information sufficient to demonstrate to the Commission the financial qualifications of the applicant to carry out, in accordance with the regulations in this chapter, the activities for which the license is sought. The information must state the place at which the activity is to be performed, the general plan for carrying out the activity, and the period of time for which the license is requested. The information must show that the applicant either possesses the necessary funds, or that the applicant has reasonable assurance of obtaining the necessary funds or that by a combination of the two, the applicant will have the necessary funds available to cover the following:

(1) Estimated construction costs;

(2) Estimated operating costs over the planned life of the ISFSI; and

(3) Estimated decommissioning costs, and the necessary financial arrangements to provide reasonable assurance prior to licensing that decommissioning will be carried out after the removal of spent fuel and/or high-level radioactive waste from storage.

§ 72.24 Contents of application: Technical information.

Each application for a license under this part must include a Safety Analysis Report describing the proposed ISFSI or MRS for the receipt, handling, packaging, and storage of spent fuel or high-level radioactive waste, including how the ISFSI or MRS will be operated. The minimum information to be included in this report must consist of the following:

(a) A description and safety assessment of the site on which the ISFSI or MRS is to be located, with appropriate attention to the design bases for external events. Such assessment must contain an analysis and evaluation of the major structures, systems, and components of the ISFSI or MRS that bear on the suitability of the site when the ISFSI or MRS is operated at its design capacity. If the proposed ISFSI or MRS is to be located on the site of a nuclear power plant or other licensed facility, the potential interactions between the ISFSI or MRS and such other facility must be evaluated.

(b) A description and discussion of the ISFSI or MRS structures with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.

(c) The design of the ISFSI or MRS in sufficient detail to support the findings in § 72.40, including:

(1) The design criteria for the ISFSI or MRS pursuant to Subpart F of this part, with identification and justification for any additions to or departures from the general design criteria;

(2) The design bases and the relation of the design bases to the design criteria;

(3) Information relative to materials of construction, general arrangement, dimensions of principal structures, and descriptions of all structures, systems, and components important to safety, in sufficient detail to support a finding that the ISFSI or MRS will satisfy the design bases with an adequate margin for safety, and

(4) Applicable codes and standards.

(d) An analysis and evaluation of the design and performance of structures, systems, and components important to safety, with the objective of assessing the impact on public health and safety resulting from operation of the ISFSI or MRS and including determination of:

(1) The margins of safety during normal operations and expected operational occurrences during the life of the ISFSI or MRS, and

(2) The adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents, including natural and manmade phenomena and events.

(e) The means for controlling and limiting occupational radiation exposures within the limits given in Part 20 of this chapter, and for meeting the objective of maintaining exposures as low as is reasonably achievable.

(f) The features of ISFSI or MRS design and operating modes to reduce to the extent practicable radioactive waste volumes generated at the installation.

(g) An identification and justification for the selection of those subjects that will be probable license conditions and technical specifications. These subjects must cover the design, construction, preoperational testing, operation, and decommissioning of the ISFSI or MRS.

(h) A plan for the conduct of operations, including the planned managerial and administrative controls system, and the applicant's organization, and program for training of personnel pursuant to Subpart I.

(i) If the proposed ISFSI or MRS incorporates structures, systems, or components important to safety whose functional adequacy or reliability have not been demonstrated by prior use for that purpose or cannot be demonstrated by reference to performance data in related applications or to widely accepted engineering principles, an identification of these structures, systems, or components along with a schedule showing how safety questions will be resolved prior to the initial receipt of spent fuel or high-level radioactive waste for storage at the ISFSI or MRS.

(j) The technical qualifications of the applicant to engage in the proposed activities, as required by § 72.28.

(k) A description of the applicant's plans for coping with emergencies, as required by § 72.32.

(l) A description of the equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal operations and expected operational occurrences. The description must identify the design objectives and the means to be used for keeping levels of radioactive material in effluents to the environment as low as is reasonably achievable and within the exposure limits stated in § 72.104. The description must include:

(1) An estimate of the quantity of each of the principal radionuclides expected to be released annually to the environment in liquid and gaseous effluents produced during normal ISFSI or MRS operations;

(2) A description of the equipment and processes used in radioactive waste systems; and

(3) A general description of the provisions for packaging, storage, and disposal of solid wastes containing radioactive materials resulting from treatment of gaseous and liquid effluents and from other sources.

(m) An analysis of the potential dose equivalent or committed dose equivalent to an individual outside the controlled area from accidents or natural phenomena events that result in the release of radioactive material to the environment or direct radiation from the ISFSI or MRS. The calculations of individual dose equivalent or committed dose equivalent must be performed for direct exposure, inhalation, and ingestion occurring as a result of the postulated design basis event.

(n) A description of the quality assurance program that satisfies the requirements of Subpart G to be applied to the design, fabrication, construction, testing, operation, modification, and decommissioning of the structures, systems, and components of the ISFSI or MRS important to safety. The description must identify the structures, systems, and components important to safety. The program must also apply to managerial and administrative controls used to ensure safe operation of the ISFSI or MRS.

(o) A description of the detailed security measures for physical protection, including design features and the plans required by Subpart H. For an application from DOE for an ISFSI or MRS, DOE will provide a description of the physical security plan for protection against radiological sabotage as required by Subpart H. An application submitted by DOE for an ISFSI or MRS must include a certification that it will provide at the ISFSI or MRS such safeguards as it requires at comparable surface DOE facilities to promote the common defense and security.

(p) A description of the program covering preoperational testing and initial operations.

(q) A description of the decommissioning plan required under § 72.30.

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§ 72.26 Contents of application: Technical specifications.

Each application under this part shall include proposed technical specifications in accordance with the requirements of § 72.44 and a summary statement of the bases and justifications for these technical specifications.

§ 72.28 Contents of application: Applicant's technical qualifications.

Each application under this part must include:

(a) The technical qualifications, including training and experience, of the applicant to engage in the proposed activities;

(b) A description of the personnel training program required under Subpart I;

(c) A description of the applicant's operating organization, delegations of responsibility and authority and the minimum skills and experience qualifications relevant to the various levels of responsibility and authority; and

(d) A commitment by the applicant to have and maintain an adequate complement of trained and certified installation personnel prior to the receipt of spent fuel or high-level radioactive waste for storage.

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§ 72.30 Decommissioning planning, including financing and recordkeeping.

(a) Each application under this part must include a proposed decommissioning plan that contains sufficient information on proposed practices and procedures for the decontamination of the site and facilities and for disposal of residual radioactive materials after all spent fuel or high-level radioactive waste has been removed, in order to provide reasonable assurance that the decontamination and decommissioning of the ISFSI or MRS at the end of its useful life will provide adequate protection to the health and safety of the public. This plan must identify and discuss those design features of the ISFSI or MRS that facilitate its decontamination and decommissioning at the end of its useful life.

(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.

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(c) Financial assurance for decommissioning must be provided by one or more of the following methods:

(1) *Prepayment*. Prepayment is the deposit prior to the start of operation into an account segregated from licensee assets and outside the licensee's administrative control of cash or liquid assets such that the amount of funds would be sufficient to pay decommissioning costs. Prepayment may be in the form of a trust, escrow account, government fund, certificate of deposit, or deposit of government securities.

(2) *A surety method, insurance, or other guarantee method*. These methods guarantee that decommissioning costs will be paid should the licensee default. A surety method may be in the form of a surety bond, letter of credit, or line of credit. A parent company guarantee of funds for decommissioning costs based on a financial test may be used if the guarantee and test are as contained in Appendix A of 10 CFR Part 30. A parent company guarantee may not be used in combination with other financial methods to satisfy the requirements of this section. Any surety method or insurance used to provide financial assurance for decommissioning must contain the following conditions:

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(i) The surety method or insurance must be open-ended or, if written for a specified term, such as five years, must be renewed automatically unless 90 days or more prior to the renewal date, the issuer notifies the Commission, the beneficiary, and the licensee of its intention not to renew. The surety method or insurance must also provide that the full face amount be paid to the beneficiary automatically prior to the expiration without proof of forfeiture if the licensee fails to provide a replacement acceptable to the Commission withing 30 days after receipt of notification or cancellation.

(ii) The surety method or insurance must be payable to a trust established for decomissioning costs. The trustee and trust must be acceptable to the Commission. An acceptable trustee includes an appropriate State or Federal government agency or an entity which has the authority to act as a trustee and whose trust operations are regulated and examined by a Federal or State agency.

(iii) The surety or insurance must remain in effect until the Commission has terminated the license.

(3) An external sinking fund in which deposits are made at least annually, coupled with a surety method or insurance, the value of which may decrease by the amount being accumulated in the sinking fund. An external sinking fund is a fund establishing and maintained by setting aside funds periodically in an account segregated from licensee assets and outside the licensee's administrative control in which the total amount of funds would be sufficient to pay decommissioning costs at the time termination of operation is expected. An external sinking fund may be in the form of a trust, escrow account, government fund, certificate of deposit, or deposit of government securities. The surety or insurance provision must be as stated in paragraph (c)(2) of this section.

(4) In the case of Federal, State, or local government licensees, a statement of intent containing a cost estimate for decommissioning, and indicating that funds for decommissioning will be obtained when necessary.

(5) In the case of electric utility licensees, the methods of § 50.75(e) (1) and (3) of this chapter.

(d) Each licensee shall keep records of information important to the safe and effective decommissioning of the facility in an identified location until the license is terminated by the Commission. If records of relevant information are kept for other purposes, reference to these records and their locations may be used. Information the Commission considers important to decommissioning consists of—

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(1) Records of spills or other unusual occurrences involving the spread of contamination in and around the facility, equipment, or site. These records may be limited to instances when contamination remains after any cleanup procedures or when there is reasonable likelihood that contaminants may have spread to inaccessible areas as in the case of possible seepage into porous materials such as concrete. These records must include any known information on identification of involved nuclides, quantities, forms, and concentrations.

(2) As-built drawings and modifications of structures and equipment in restricted areas where radioactive materials are used and/or stored, and of locations of possible inaccessible contamination such as buried pipes which may be subject to contamination. If required drawings are referenced, each relevant document need not be indexed individually. If drawings are not available, the licensee shall substitute appropriate records of available information concerning these areas and locations.

(3) Records of the cost estimate performed for the decommissioning funding plan or of the amount certified for decommissioning, and records of the funding method used for assuring funds if either a funding plan or certification is used.

§ 72.32 Emergency plan.

(a) [Reserved]

(b) [Reserved]

(c) For an ISFSI that is located on the site of a nuclear power reactor licensed for operation by the Commission, the emergency plan required by 10 CFR 50.47 shall be deemed to satisfy the requirements of this section.

§ 72.34 Environmental report.

Each application for an ISFSI or MRS license under this part must be accompanied by an Environmental Report which meets the requirements of Subpart A of Part 51 of this chapter.

Subpart C—Issuance and Conditions of License

§ 72.40 Issuance of license.

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(a) Except as provided in paragraph (c) of this section, the Commission will issue a license under this part upon a determination that the application for a license meets the standards and requirements of the Act and the regulations of the Commission, and upon finding that:

(1) The applicant's proposed ISFSI or MRS design complies with Subpart F;

(2) The proposed site complies with the criteria in Subpart E;

(3) If on the site of a nuclear power plant or other licensed activity or facility, the proposed ISFSI would not pose an undue risk to the safe operation of such nuclear power plant or other licensed activity or facility;

(4) The applicant is qualified by reason of training and experience to conduct the operation covered by the regulations in this part;

(5) The applicant's proposed operating procedures to protect health and to minimize danger to life or property are adequate;

(6) Except for DOE, the applicant for an ISFSI or MRS is financially qualified to engage in the proposed activities in accordance with the regulations in this part;

(7) The applicant's quality assurance plan complies with Subpart G;

(8) The applicant's physical protection provisions comply with Subpart H. DOE has complied with the safeguards and physical security provisions identified in § 72.24(o);

(9) The applicant's personnel training program complies with Subpart I;

(10) Except for DOE, the applicant's decommissioning plan and its financing pursuant to § 72.30 provide reasonable assurance that the decontamination and decommissioning of the ISFSI or MRS at the end of its useful life will provide adequate protection to the health and safety of the public;

(11) The applicant's emergency plan complies with § 72.32;

(12) The applicable provisions of Part 170 of this chapter have been satisfied;

(13) There is reasonable assurance that: (i) The activities authorized by the license can be conducted without endangering the health and safety of the public and (ii) these activities will be conducted in compliance with the applicable regulations of this chapter; and

(14) The issuance of the license will not be inimical to the common defense and security.

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(b) Grounds for denial of a license to store spent fuel in the proposed ISFSI or to store spent fuel and high-level radioactive waste in the proposed MRS may be the commencement of construction prior to (1) a finding by the Director, Office of Nuclear Material Safety and Safeguards or designee or (2) a finding after a public hearing by the presiding officer, Atomic Safety and Licensing Board, Atomic Safety and Licensing Appeal Board, or the Commission acting as a collegial body, as appropriate, that the action called for is the issuance of the proposed license with any appropriate conditions to protect environmental values. This finding is to be made on the basis of information filed and evaluations made pursuant to Subpart A of Part 51 of this chapter or in the case of an MRS on the basis of evaluations made pursuant to sections 141(c) and (d) or 148(a) and (c) of NWPA (96 Stat. 2242, 2243, 42 U.S.C. 10161(c), (d); 101 Stat. 1330–235, 1330–236, 42 U.S.C. 10168(a), (c)), as appropriate, and after weighing the environmental, economic, technical and other benefits against environmental costs and considering available alternatives.

(c) For facilities that have been covered under previous licensing actions including the issuance of a construction permit under Part 50 of this chapter, a reevaluation of the site is not required except where new information is discovered which could alter the original site evaluation findings. In this case, the site evaluation factors involved will be reevaluated.

§ 72.42 Duration of license; renewal.

(a) Each license issued under this part must be for a fixed period of time to be specified in the license. The license term for an ISFSI must not exceed 20 years from the date of issuance. The license term for an MRS must not exceed 40 years from the date of issuance. Licenses for either type of installation may be renewed by the Commission at the expiration of the license term upon application by the licensee and pursuant to the requirements of this rule.

(b) Applications for renewal of a license should be filed in accordance with the applicable provisions of Subpart B at least two years prior to the expiration of the existing license. Information contained in previous applications, statements, or reports filed with the Commission under the license may be incorporated by reference: Provided, that such references are clear and specific.

(c) In any case in which a licensee, not less than two years prior to expiration of its existing license, has filed an application in proper form for renewal of a license, the existing license shall not expire until a final decision concerning the application for renewal has been made by the Commission.

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§ 72.44 License conditions.

(a) Each license issued under this part shall include license conditions. The license conditions may be derived from the analyses and evaluations included in the Safety Analysis Report and amendments thereto submitted pursuant to § 72.24. License conditions pertain to design, construction and operation. The Commission may also include additional license conditions as it finds appropriate.

(b) Each license issued under this part shall be subject to the following conditions, even if they are not explicitly stated therein:

(1) Neither the license nor any right thereunder shall be transferred, assigned, or disposed of in any manner, either voluntarily or involuntarily, directly or indirectly, through transfer of control of the license to any person, unless the Commission shall, after securing full information, find that the transfer is in accordance with the provisions of the Atomic Energy Act of 1954, as amended, and give its consent in writing.

(2) The license shall be subject to revocation, suspension, modification, or amendment in accordance with the procedures provided by the Atomic Energy Act of 1954, as amended, and Commission regulations.

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(3) Upon request of the Commission, the licensee shall, at any time before expiration of the license, submit written statements, signed under oath or affirmation if appropriate, to enable the Commission to determine whether or not the license should be modified, suspended, or revoked.

(4) Prior to the receipt of spent fuel for storage at an ISFSI or the receipt of spent fuel and high-level radioactive waste for storage at an MRS, the licensee shall have in effect an NRC-approved program covering the training and certification of personnel that meets the requirements of Subpart I.

(5) The license shall permit the operation of the equipment and controls that are important to safety of the ISFSI or the MRS only by personnel whom the licensee has certified as being adequately trained to perform such operations, or by uncertified personnel who are under the direct visual supervision of a certified individual.

(6)(i) Each licensee shall notify the appropriate NRC Regional Administrator, in writing, immediately following the filing of a voluntary or involuntary petition for bankruptcy under any Chapter of Title II (Bankruptcy) of the United States Code by or against:

(A) The licensee;

(B) An entity (as that term is defined in 11 U.S.C. 101(14)) controlling the licensee or listing the license or licensee as property of the estate; or

(C) An affiliate (as that term is defined in 11 U.S.C. 101(2)) of the licensee.

(ii) This notification must indicate:

(A) The bankruptcy court in which the petition for bankruptcy was filed, and

(B) The date of the filing of the petition.

(c) Each license issued under this part must include technical specifications. Technical specifications must include requirements in the following categories:

(1) *Functional and operating limits and monitoring instruments and limiting control settings*

(i) Functional and operating limits for an ISFSI or MRS are limits on fuel or waste handling and storage conditions that are found to be necessary to protect the integrity of the stored fuel or waste container, to protect employees against occupational exposures and to guard against the uncontrolled release of radioactive materials; and

(ii) Monitoring instruments and limiting control settings for an ISFSI or MRS are those related to fuel or waste handling and storage conditions having significant safety functions.

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(2) *Limiting conditions.* Limiting conditions are the lowest functional capability or performance levels of equipment required for safe operation.

(3) *Surveillance requirements.* Surveillance requirements include:

(i) Inspection and monitoring of spent fuel or high-level radioactive waste in storage;

(ii) Inspection, test and calibration activities to ensure that the necessary integrity of required systems and components is maintained;

(iii) confirmation that operation of the ISFSI or MRS is within the required functional and operating limits; and

(iv) confirmation that the limiting conditions required for safe storage are met.

(4) *Design features.* Design features include items that would have a significant effect on safety if altered or modified, such as materials of construction and geometric arrangements.

(5) *Administrative controls.* Administrative controls include the organization and management procedures, recordkeeping, review and audit, and reporting necessary to assure that the operations involved in the storage of spent fuel in an ISFSI and the storage of spent fuel and high-level radioactive waste in an MRS are performed in a safe manner.

(d) Each license authorizing the receipt, handling, and storage of spent fuel or high-level radioactive waste under this part must include technical specifications that, in addition to stating the limits on the release of radioactive materials for compliance with limits of Part 20 of this chapter and the "as low as is reasonably achievable" objectives for effluents, require that:

(1) Operating procedures for control of effluents be established and followed, and equipment in the radioactive waste treatment systems be maintained and used, to meet the requirements of § 72.104;

(2) An environmental monitoring program be established to ensure compliance with the technical specifications for effluents; and

(3) An annual report be submitted to the appropriate regional office specified in Appendix A of Part 73 of this chapter, with a copy to the Director, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555, within 60 days after January 1 of each year, specifying the quantity of each of the principal radionuclides released to the environment in liquid and in gaseous effluents during the previous 12 months of operation and such other information as may be required by the Commission to estimate maximum potential radiation dose commitment to the public resulting from effluent releases. On the basis of this report and any additional information the Commission may obtain from the licensee or others, the Commission may from time to time require the licensee to take such action as the Commission deems appropriate.

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(e) The licensee shall make no change that would decrease the effectiveness of the physical security plan prepared pursuant to § 72.180 without the prior approval of the Commission. A licensee desiring to make such a change shall submit an application for an amendment to the license pursuant to § 72.56. A licensee may make changes to the physical security plan without prior Commission approval, provided that such changes do not decrease the effectiveness of the plan. The licensee shall furnish to the Commission a report containing a description of each change within two months after the change is made, and shall maintain records of changes to the plan made without prior Commission approval for a period of 3 years from the date of the change.

(f) A licensee shall follow and maintain in effect an emergency plan that is approved by the Commission. The licensee may make changes to the approved plan without Commission approval only if such changes do not decrease the effectiveness of the plan. Within six months after any change is made, the licensee shall submit a report containing a description of any changes made in the plan to the appropriate NRC Regional Office specified in Appendix A to Part 73 of this chapter with a copy to the Director, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Proposed changes that decrease the effectiveness of the approved emergency plan must not be implemented unless the licensee has received prior approval of such changes from the Commission.

(g) A license issued to DOE under this part for an MRS authorized by section 142(b) of NWPA (101 Stat. 1330-232, 42 U.S.C. 10162(b)) must include the following conditions:

(1) Construction of the MRS may not begin until the Commission has authorized the construction of a repository under section 114(d) of NWPA (96 Stat. 2215, as amended by 101 Stat. 1330-230, 42 U.S.C. 10134(d)) and Part 60 of this chapter.

(2) Construction of the MRS or acceptance of spent nuclear fuel or high-level radioactive waste at the MRS is

prohibited during such time as the repository license is revoked by the Commission or construction of the repository ceases;

(3) The quantity of spent nuclear fuel or high-level radioactive waste at the site of the MRS at any one time may not exceed 10,000 metric tons of heavy metal until a repository authorized under NWPA and Part 60 of this chapter first accepts spent nuclear fuel or solidified high-level radioactive waste; and

(4) The quantity of spent nuclear fuel or high-level radioactive waste at the site of the MRS at any one time may not exceed 15,000 metric tons of heavy metal.

§ 72.46 Public hearings.

(a) In connection with each application for a license under this part, the Commission shall issue or cause to be issued a notice of proposed action and opportunity for hearing in accordance with § 2.105 or § 2.1107 of this chapter, as appropriate, or, if the Commission finds that a hearing is required in the public interest, a notice of hearing in accordance with § 2.104 of this chapter.

(b)(1) In connection with each application for an amendment to a license under this part, the Commission shall, except as provided in paragraph (b)(2) of this section, issue or cause to be issued a notice of proposed action and opportunity for hearing in accordance with § 2.105 or § 2.1107 of this chapter, as appropriate, or, if the Commission finds that a hearing is required in the public interest, a notice of hearing in accordance with § 2.104 of this chapter. 53 FR 31651

(2) The Director, Office of Nuclear Material Safety and Safeguards, or the Director's designee may dispense with a notice of proposed action and opportunity for hearing or a notice of hearing and take immediate action on an amendment to a license issued under this part upon a determination that the amendment does not present a genuine issue as to whether the health and safety of the public will be significantly affected. After taking the action, the Director or the Director's designee shall promptly publish a notice in the Federal Register of the action taken and of the right of interested persons to request a hearing on whether the action should be rescinded or modified. If the action taken amends an MRS license, the Director or the Director's designee shall also inform the appropriate State and local officials.

(c) The notice of proposed action and opportunity for hearing or the notice of hearing may be included in the notice of docketing required to be published by § 72.16 of this part.

(d) If no request for a hearing or petition for leave to intervene is filed within the time prescribed in the notice of proposed action and opportunity for hearing, the Director, Office of Nuclear Material Safety and Safeguards or the Director's designee may take the proposed action, and thereafter shall promptly inform the appropriate State and local officials and publish a notice in the Federal Register of the action taken. In accordance with § 2.764(c) of this chapter, the Director, Office of Nuclear Material Safety and Safeguards shall not issue an initial license for the construction and operation of an ISFSI or an MRS until expressly authorized to do so by the Commission.

§ 72.48 Changes, tests, and experiments.

(a)(1) The holder of a license issued under this part may:

(i) Make changes in the ISFSI or MRS described in the Safety Analysis Report,

(ii) Make changes in the procedures described in the Safety Analysis Report, or

(iii) Conduct tests or experiments not described in the Safety Analysis Report, without prior Commission approval, unless the proposed change, test or experiment involves a change in the license conditions incorporated in the license, an unreviewed safety question, a significant increase in occupational exposure or a significant unreviewed environmental impact. 53 FR 31651

(2) A proposed change, test, or experiment shall be deemed to involve an unreviewed safety question—

(i) If the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased;

(ii) If a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report may be created; or

(iii) If the margin of safety as defined in the basis for any technical specification is reduced.

(b)(1) The licensee shall maintain records of changes in the ISFSI or MRS and of changes in procedures made pursuant to this section if these changes constitute changes in the ISFSI or MRS or procedures described in the Safety Analysis Report. The licensee shall also maintain records of tests and experiments carried out pursuant to paragraph (a) of this section. These records must include a written safety evaluation that provides the bases for the determination that the change, test, or experiment does not involve an unreviewed safety question. The records of changes in the ISFSI or MRS and of changes in procedures and records of tests must be maintained until the Commission terminates the license.

(2) Annually, or at such shorter interval as may be specified in the license, the licensee shall furnish to the appropriate regional office, specified in Appendix A of Part 73 of this chapter, with a copy to the Director, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555, a report containing a brief description of changes, tests, and experiments made under paragraph (a) of the section, including a summary of the safety evaluation of each. Any report submitted by a licensee pursuant to this paragraph will be made a part of the public record pertaining to this license.

(c) The holder of a license issued under this part who desires—

(1) To make changes in the ISFSI or MRS or the procedures as described in the Safety Analysis Report, or to conduct tests or experiments not described in the Safety Analysis Report, that involve an unreviewed safety question, a significant increase in occupational exposure, or significant unreviewed environmental impact, or 53 FR 31651

(2) To change the license conditions shall submit an application for amendment of the license, pursuant to § 72.56.

§ 72.50 Transfer of license.

(a) No license or any part included in a license issued under this part for an ISFSI or MRS shall be transferred, assigned, or in any manner disposed of, either voluntarily or involuntarily, directly or indirectly, through transfer of control of the license to any person, unless the Commission gives its consent in writing.

(b)(1) An application for transfer of a license must include as much of the information described in §§ 72.22 and 72.28 with respect to the identity and the technical and financial qualifications of the proposed transferee as would be required by those sections if the application were for an initial license. The application must also include a statement of the purposes for which the transfer of the license is requested and the nature of the transaction necessitating or making desirable the transfer of the license.

(2) The Commission may require any person who submits an application for the transfer of a license pursuant to the provisions of this section to file a

written consent from the existing licensee, or a certified copy of an order or judgment of a court of competent jurisdiction, attesting to the person's right—subject to the licensing requirements of the Act and these regulations—to possession of the radioactive materials and the storage installation involved.

(c) After appropriate notice to interested persons, including the existing licensee, and observance of such procedures as may be required by the Act or regulations or orders of the Commission, the Commission will approve an application for the transfer of a license, if the Commission determines that:

(1) The proposed transferee is qualified to be the holder of the license; and

(2) Transfer of the license is consistent with applicable provisions of the law, and the regulations and orders issued by the Commission.

§ 72.52 Creditor regulations.

(a) This section does not apply to an ISFSI or MRS constructed and operated by DOE.

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(b) Pursuant to section 184 of the Act, the Commission consents, without individual application, to the creation of any mortgage, pledge, or other lien on special nuclear material contained in spent fuel not owned by the United States that is the subject of a license or on any interest in special nuclear material in spent fuel; Provided:

(1) That the rights of any creditor so secured may be exercised only in compliance with and subject to the same requirements and restrictions as would apply to the licensee pursuant to the provisions of the license, the Atomic Energy Act of 1954, as amended, and regulations issued by the Commission pursuant to said Act; and

(2) That no creditor so secured may take possession of the spent fuel pursuant to the provisions of this section prior to either the issuance of a license from the Commission authorizing possession or the transfer of the license.

(c) Any creditor so secured may apply for transfer of the license covering spent fuel by filing an application for transfer of the license pursuant to § 72.50(b). The Commission will act upon the application pursuant to § 72.50(c).

(d) Nothing contained in this regulation shall be deemed to affect the means of acquiring, or the priority of, any tax lien or other lien provided by law.

(e) As used in this section, "creditor" includes, without implied limitation, the trustee under any mortgage, pledge, or lien on spent fuel in storage made to secure any creditor; any trustee or receiver of spent fuel appointed by a court of competent jurisdiction in any action brought for the benefit of any creditor secured by such mortgage, pledge, or lien; any purchaser of the spent fuel at the sale thereof upon foreclosure of the mortgage, pledge, or lien or upon exercise of any power of sale contained therein; or any assignee of any such purchaser.

§ 72.54 Application for termination of license.

(a) Any licensee may apply to the Commission for authority to surrender a license voluntarily and to decommission the ISFSI or MRS. This application must be made within two years following permanent cessation of operations, and in no case later than one year prior to expiration of the license. Each application for termination of license must be accompanied, or preceded, by a proposed final decommissioning plan.

(b) The proposed final decommissioning plan must include—

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(1) The choice of the alternative for decommissioning with a description of activities involved. An alternative is acceptable if it provides for completion of decommissioning without significant delay. Consideration will be given to an alternative which provides for delayed completion of decommissioning only when necessary to protect the public health and safety. Factors to be considered in evaluating an alternative which provides for delayed completion of decommissioning include unavailability of waste disposal capacity and other site specific factors affecting the licensee's capability to carry out decommissioning safely, including presence of other nuclear facilities at the site.

(2) A description of controls and limits on procedures and equipment to protect occupational and public health and safety;

(3) A description of the planned final radiation survey; and

(4) An updated detailed cost estimate for the chosen alternative for decommissioning, comparison of that estimate with present funds set aside for decommissioning, and plan for assuring the availability of adequate funds for completion of decommissioning including means for adjusting cost estimates and associated funding levels over any storage or surveillance period.

(5) A description of technical specifications and quality assurance provisions in place during decommissioning.

(c) For final decommissioning plans in which the major dismantlement activities are delayed by first placing the ISFSI or MRS in storage, planning for these delayed activities may be less detailed. Updated detailed plans must be submitted and approved prior to the start of such activities.

(d) If the final decommissioning plan demonstrates that the decommissioning will be performed in accordance with the regulations in this chapter and will not be inimical to the common defense and security or to the health and safety of the public, and after notice to interested persons, the Commission will approve the plan subject to such conditions and limitations as it deems appropriate and necessary and issue an order authorizing the decommissioning.

(e) The Commission will terminate the license if it determines that—

(1) The decommissioning has been performed in accordance with the approved final decommissioning plan and the order authorizing decommissioning; and

(2) The terminal radiation survey and associated documentation demonstrates that the ISFSI or MRS and site are suitable for release for unrestricted use.

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§ 72.56 Application for amendment of license.

Whenever a holder of a license desires to amend the license, an application for an amendment shall be filed with the Commission fully describing the changes desired and the reasons for such changes, and following as far as applicable the form prescribed for original applications.

§ 72.58 Issuance of amendment.

In determining whether an amendment to a license will be issued to the applicant, the Commission will be guided by the considerations that govern the issuance of initial licenses.

§ 72.60 Modification, revocation, and suspension of license.

(a) The terms and conditions of all licenses are subject to amendment, revision, or modification by reason of amendments to the Atomic Energy Act of 1954, as amended, or by reason of rules, regulations, or orders issued in accordance with the Act or any amendments thereto.

(b) Any license may be modified, revoked, or suspended in whole or in part for any of the following:

(1) Any material false statement in the application or in any statement of fact required under section 182 of the Act;

(2) Conditions revealed by the application or statement of fact or any report, record, inspection or other means which would warrant the Commission to

refuse to grant a license on an original application.

(2) Failure to operate an ISFSI or MRS in accordance with the terms of the license;

(3) Violation of, or failure to observe, any of the terms and conditions of the Act, or of any applicable regulation, license, or order of the Commission.

(c) Upon revocation of a license, the Commission may immediately cause the retaking of possession of all special nuclear material contained in spent fuel held by the licensee. In cases found by the Commission to be of extreme importance to the national defense and security or to the health and safety of the public, the Commission prior to following any of the procedures provided under sections 551–558 of Title 5 of the United States Code, may cause the taking of possession of any special nuclear material contained in spent fuel held by the licensee.

§ 72.62 Backfitting.

(a) As used in this section, "backfitting" means the addition, elimination, or modification, after the license has been issued, of:

(1) Structures, systems, or components of an ISFSI or MRS, or

(2) Procedures or organization required to operate an ISFSI or MRS.

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(b) The Commission will require backfitting of an ISFSI or MRS if it finds that such action is necessary to assure adequate protection to occupational or public health and safety, or to bring the ISFSI or MRS into compliance with a license or the rules or orders of the Commission, or into conformance with written commitments by a licensee.

(c) The Commission may require the backfitting of an ISFSI or MRS if it finds:

(1) That there is a substantial increase in the overall protection of the occupational or public health and safety to be derived from the backfit, and

(2) That the direct and indirect costs of implementation for that ISFSI or MRS are justified in view of this increased protection.

(d) The Commission may at any time require a holder of a license to submit such information concerning the backfitting or the proposed backfitting of an ISFSI or MRS as it deems appropriate.

Subpart D—Records, Reports, Inspections, and Enforcement

§ 72.70 Safety analysis report updating.

(a) The design, description of planned operations, and other information submitted in the Safety Analysis Report shall be updated by the licensee and submitted to the Commission at least once every six months after issuance of the license during final design and construction, until preoperational testing is completed, with final Safety Analysis Report completion and submittal to the Commission at least 90 days prior to the planned receipt of spent fuel or high-level radioactive waste. The final submittal must include a final analysis and evaluation of the design and performance of structures, systems, and components that are important to safety taking into account any pertinent information developed since the submittal of the license application.

(b) After the first receipt of spent fuel or high-level radioactive waste for storage, the Safety Analysis Report must be updated annually and submitted to the Commission by the licensee. This submittal must include the following:

(1) New or revised information relating to applicable site evaluation factors, including the results of environmental monitoring programs.

(2) A description and analysis of changes in the structures, systems, and components of the ISFSI or MRS, with emphasis upon:

(i) Performance requirements,

(ii) The bases, with technical justification therefor upon which such requirements have been established, and

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(iii) Evaluations showing that safety functions will be accomplished.

(3) An analysis of the significance of any changes to codes, standards, regulations, or regulatory guides which the licensee has committed to meeting the requirements of which are applicable to the design, construction, or operation of the ISFSI or MRS.

§ 72.72 Material balance, inventory, and records requirements for stored materials.

(a) Each licensee shall keep records showing the receipt, inventory (including location), disposal, acquisition, and transfer of all spent fuel and high-level radioactive waste in storage. The records must include as a minimum the name of shipper of the material to the ISFSI or MRS, the estimated quantity of radioactive material per item (including special nuclear material in spent fuel), item identification and seal number, storage location, onsite movements of each fuel assembly or storage canister, and ultimate disposal. These records for spent fuel at an ISFSI or for spent fuel and high-level radioactive waste at an MRS must be retained for as long as the material is stored and for a period of five years after the material is disposed of or transferred out of the ISFSI or MRS.

(b) Each licensee shall conduct a physical inventory of all spent fuel and high-level radioactive waste in storage at intervals not to exceed 12 months unless otherwise directed by the Commission. The licensee shall retain a copy of the current inventory as a record until the Commission terminates the license.

(c) Each licensee shall establish, maintain, and follow written material control and accounting procedures that are sufficient to enable the licensee to account for material in storage. The licensee shall retain a copy of the current material control and accounting procedures until the Commission terminates the license.

(d) Records of spent fuel and high-level radioactive waste in storage must be kept in duplicate. The duplicate set of records must be kept at a separate location sufficiently remote from the original records that a single event would not destroy both sets of records. Records of spent fuel transferred out of an ISFSI or of spent fuel or high-level radioactive waste transferred out of an MRS must be preserved for a period of five years after the date of transfer.

§ 72.74 Reports of accidental criticality or loss of special nuclear material.

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(a) Each licensee shall notify the NRC Operations Center [1] within one hour of discovery of accidental criticality or any loss of special nuclear material.

(b) This notification must be made to the NRC Operations Center via the Emergency Notification System if the licensee is party to that system. If the Emergency Notification System is inoperative or unavailable, the licensee shall make the required notification via commercial telephonic service or any other dedicated telephonic system or any other method that will ensure that a report is received by the NRC Operations Center within one hour. The exemption of § 73.21(g)(3) of this chapter applies to all telephonic reports required by this section.

(c) Reports required under § 73.71 of this chapter need not be duplicated under the requirements of this section.

§ 72.76 Material status reports.

(a) Except as provided in paragraph (b) of this section, each licensee shall complete and submit to the Commission (on DOE/NRC Form-742, Material Balance Report) material status reports in accordance with the printed instructions for completing the form. These reports must provide information concerning the special nuclear material

[1] Commercial telephone number of the NRC Operations Center is (301) 951-0550.

contained in the spent fuel possessed, received, transferred, disposed of, or lost by the licensee. Material status reports must be made as of March 31 and September 30 of each year and filed within 30 days after the end of the period covered by the report. The Commission may, when good cause is shown, permit a licensee to submit material status reports at other times.

(b) Any licensee who is required to submit routine material status reports pursuant to § 75.35 of this chapter (pertaining to implementation of the US/IAEA Safeguards Agreement) shall prepare and submit such reports only as provided in that section instead of as provided in paragraph (a) of this section.

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§ 72.78 Nuclear material transfer reports.

(a) Except as provided in paragraph (b) of this section, whenever the licensee transfers or receives spent fuel, the licensee shall complete and distribute a Nuclear Material Transaction Report on DOE/NRC Form-741 in accordance with printed instructions for completing the form. Each ISFSI licensee who receives spent fuel from a foreign source shall complete both the supplier's and receiver's portion of DOE/NRC Form-741, verify the identity of the spent fuel, and indicate the results on the receiver's portion of the form.

(b) Any licensee who is required to submit inventory change reports on DOE/NRC Form-741 pursuant to § 75.34 of this chapter (pertaining to implementation of the US/IAEA Safeguards Agreement) shall prepare and submit such reports only as provided in that section instead of as provided in paragraph (a) of this section.

§ 72.80 Other records and reports.

(a) Each licensee shall maintain any records and make any reports that may be required by the conditions of the license or by the rules, regulations, and orders of the Commission in effectuating the purposes of the Act.

(b) Each licensee shall furnish a copy of its annual financial report, including the certified financial statements, to the Commission.

(c) Records that are required by the regulations in this part or by the license conditions must be maintained for the period specified by the appropriate regulation or license condition. If a retention period is not otherwise specified, the above records must be maintained until the Commission terminates the license.

(d) Any record that must be maintained pursuant to this part may be either the original or a reproduced copy by any state of the art method provided that any reproduced copy is duly authenticated by authorized personnel and is capable of producing a clear and legible copy after storage for the period specified by Commission regulations.

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§ 72.82 Inspections and tests.

(a) Each licensee under this part shall permit inspection by duly authorized representatives of the Commission of its records, premises, and activities and of spent fuel or high-level radioactive waste in its possession related to the specific license as may be necessary to effectuate the purposes of the Act, including section 105 of the Act.

(b) Each licensee under this part shall make available to the Commission for inspection, upon reasonable notice, records kept by the licensee pertaining to its receipt, possession, packaging, or transfer of spent fuel or high-level radioactive waste.

(c)(1) Each licensee under this part shall upon request by the Director, Office of Nuclear Material Safety and Safeguards or the appropriate NRC Regional Administrator provide rent-free office space for the exclusive use of the Commission inspection personnel. Heat, air conditioning, light, electrical outlets and janitorial services shall be furnished by each licensee. The office shall be convenient to and have full access to the installation and shall provide the inspector both visual and acoustic privacy.

(2) For a site with a single storage installation the space provided shall be adequate to accommodate a full-time inspector, a part-time secretary, and transient NRC personnel and will be generally commensurate with other office facilities at the site. A space of 250 sq. ft., either within the site's office complex or in an office trailer, or other onsite space, is suggested as a guide. For sites containing multiple facilities, additional space may be requested to accommodate additional full-time inspectors. The office space that is provided shall be subject to the approval of the Director, Office of Nuclear Material Safety and Safeguards or the appropriate NRC Regional Administrator. All furniture, supplies and Commission equipment will be furnished by the Commission.

(3) Each licensee under this part shall afford any NRC resident inspector assigned to that site, or other NRC inspectors identified by the Regional Administrator as likely to inspect the installation, immediate unfettered access, equivalent to access provided regular plant employees, following proper identification and compliance with applicable access control measures for security, radiological protection, and personal safety.

(d) Each licensee shall perform, or permit the Commission to perform, such tests as the Commission deems appropriate or necessary for the administration of the regulations in this part.

(e) A report of the preoperational test acceptance criteria and test results must be submitted to the appropriate Regional Office specified in Appendix A of Part 73 of this chapter with a copy to the Director, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555, at least 30 days prior to the receipt of spent fuel or high-level radioactive waste.

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§ 72.84 Violations.

An injunction or other court order may be obtained prohibiting any violation of any provision of the Atomic Energy Act of 1954, as amended, or title II of the Energy Reorganization Act of 1974, as amended, or any regulation or order issued thereunder. A court order may be obtained for the payment of a civil penalty imposed pursuant to section 234 of the Atomic Energy Act for violation of sections 53, 57, 62, 63, 81, or 82 of the Atomic Energy Act, or section 206 of the Energy Reorganization Act of 1974, or any rule, regulation, or order issued thereunder, or any term, condition, or limitation of any license issued thereunder, or for any violation for which a license may be revoked under section 186 of the Atomic Energy Act. Any person who willfully violates any provision of the Atomic Energy Act, or any regulation or order issued thereunder, may be guilty of a crime and, upon conviction, may be punished by fine or imprisonment or both, as provided by law.

Subpart E—Siting Evaluation Factors

§ 72.90 General considerations.

(a) Site characteristics that may directly affect the safety or environmental impact of the ISFSI or MRS must be investigated and assessed.

(b) Proposed sites for the ISFSI or MRS must be examined with respect to the frequency and the severity of external natural and maninduced events that could affect the safe operation of the ISFSI or MRS.

(c) Design basis external events must be determined for each combination of proposed site and proposed ISFSI or MRS design.

(d) Proposed sites with design basis external events for which adequate protection cannot be provided through ISFSI or MRS design shall be deemed unsuitable for the location of the ISFSI or MRS.

(e) Pursuant to Subpart A of Part 51 of this chapter for each proposed site for an ISFSI and pursuant to sections 141 or 118 of NWPA, as appropriate (96 Stat. 2241, 101 Stat. 1330-235, 42 U.S.C. 10161, 10168) for each proposed site for an MRS, the potential for radiological and other environmental impacts on the region must be evaluated with due consideration of the characteristics of the population, including its distribution, and of the regional environs, including its historical and esthetic values.

(f) The facility must be sited so as to avoid to the extent possible the long-term and short-term adverse impacts associated with the occupancy and modification of floodplains.

§ 72.92 Design basis external natural events.

(a) Natural phenomena that may exist or that can occur in the region of a proposed site must be identified and assessed according to their potential effects on the safe operation of the ISFSI or MRS. The important natural phenomena that affect the ISFSI or MRS design must be identified.

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(b) Records of the occurrence and severity of those important natural phenomena must be collected for the region and evaluated for reliability, accuracy, and completeness. The applicant shall retain these records until the license is issued.

(c) Appropriate methods must be adopted for evaluating the design basis external natural events based on the characteristics of the region and the current state of knowledge about such events.

§ 72.94 Design basis external man-induced events.

(a) The region must be examined for both past and present man-made facilities and activities that might endanger the proposed ISFSI or MRS. The important potential man-induced events that affect the ISFSI or MRS design must be identified.

(b) Information concerning the potential occurrence and severity of such events must be collected and evaluated for reliability, accuracy, and completeness.

(c) Appropriate methods must be adopted for evaluating the design basis external man-induced events, based on the current state of knowledge about such events.

§ 72.96 Siting limitations.

(a) An ISFSI which is owned and operated by DOE must not be located at any site within which there is a candidate site for a HLW repository. This limitation shall apply until such time as DOE decides that such candidate site is no longer a candidate site under consideration for development as a HLW repository.

(b) An MRS must not be sited in any State in which there is located any site approved for site characterization for a HLW repository. This limitation shall apply until such time as DOE decides that the candidate site is no longer a candidate site under consideration for development as a repository. This limitation shall continue to apply to any site selected for construction as a repository.

(c) If an MRS is located, or is planned to be located, within 50 miles of the first HLW repository, any Commission decision approving the first HLW repository application must limit the quantity of spent fuel or high-level radioactive waste that may be stored. This limitation shall prohibit the storage of a quantity of spent fuel containing in excess of 70,000 metric tons of heavy metal, or a quantity of solidified high-level radioactive waste resulting from the reprocessing of such a quantity of spent fuel, in both the repository and the MRS until such time as a second repository is in operation.

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(d) An MRS authorized by section 142(b) of NWPA (101 Stat. 1330-232, 42 U.S.C. 10162(b)) may not be constructed in the State of Nevada. The quantity of spent nuclear fuel or high-level radioactive waste that may be stored at an MRS authorized by section 142(b) of NWPA shall be subject to the limitations in § 72.44(g) of this part instead of the limitations in paragraph (c) of this section.

§ 72.98 Identifying regions around an ISFSI or MRS site.

(a) The regional extent of external phenomena, man-made or natural, that are used as a basis for the design of the ISFSI or MRS must be identified.

(b) The potential regional impact due to the construction, operation or decommissioning of the ISFSI or MRS must be identified. The extent of regional impacts must be determined on the basis of potential measurable effects on the population or the environment from ISFSI or MRS activities.

(c) Those regions identified pursuant to paragraphs (a) and (b) of this section must be investigated as appropriate with respect to:

(1) The present and future character and the distribution of population,

(2) Consideration of present and projected future uses of land and water within the region, and

(3) Any special characteristics that may influence the potential consequences of a release of radioactive material during the operational lifetime of the ISFSI or MRS.

§ 72.100 Defining potential effects of the ISFSI or MRS on the region.

(a) The proposed site must be evaluated with respect to the effects on populations in the region resulting from the release of radioactive materials under normal and accident conditions during operation and decommissioning of the ISFSI or MRS; in this evaluation both usual and unusual regional and site characteristics shall be taken into account.

(b) Each site must be evaluated with respect to the effects on the regional environment resulting from construction, operation, and decommissioning for the ISFSI or MRS; in this evaluation both usual and unusual regional and site characteristics must be taken into account.

§ 72.102 Geological and seismological characteristics.

(a)(1) East of the Rocky Mountain Front (east of approximately 104° west longitude), except in areas of known seismic activity including but not limited to the regions around New Madrid, MO, Charleston, SC, and Attica, NY, sites will be acceptable if the results from onsite foundation and geological investigation, literature review, and regional geological reconnaissance show no unstable geological characteristics, soil stability problems, or potential for vibratory ground motion at the site in excess of an appropriate response spectrum anchored at 0.2 g.

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(2) For those sites that have been evaluated under paragraph (a)(1) of this section that are east of the Rocky Mountain Front, and that are not in areas of known seismic activity, a standardized design earthquake (DE) described by an appropriate response spectrum anchored at 0.25 g may be used. Alternatively, a site-specific DE may be determined by using the criteria and level of investigations required by Appendix A of Part 100 of this chapter.

(b) West of the Rocky Mountain Front (west of approximately 104° west longitude), and in other areas of known potential seismic activity, seismicity will be evaluated by the techniques of Appendix A of Part 100 of this chapter. Sites that lie within the range of strong near-field ground motion from historical earthquakes on large capable faults should be avoided.

(c) Sites other than bedrock sites must be evaluated for their liquefaction potential or other soil instability due to vibratory ground motion.

(d) Site-specific investigations and laboratory analyses must show that soil conditions are adequate for the proposed foundation loading.

(e) In an evaluation of alternative sites, those which require a minimum of engineered provisions to correct site deficiencies are preferred. Sites with unstable geologic characteristics should be avoided.

(f) The design earthquake (DE) for use in the design of structures must be determined as follows:

(1) For sites that have been evaluated under the criteria of Appendix A of 10 CFR Part 100, the DE must be equivalent to the safe shutdown earthquake (SSE) for a nuclear power plant.

(2) Regardless of the results of the investigations anywhere in the continental U.S., the DE must have a value for the horizontal ground motion of no less than 0.10 g with the appropriate response spectrum.

§ 72.104 Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS.

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(a) During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other organ as a result of exposure to:

(1) Planned discharges of radioactive materials, radon and its decay products excepted, to the general environment,

(2) Direct radiation from ISFSI or MRS operations, and

(3) Any other radiation from uranium fuel cycle operations within the region.

(b) Operational restrictions must be established to meet as low as is reasonably achievable objectives for radioactive materials in effluents and direct radiation levels associated with ISFSI or MRS operations.

(c) Operational limits must be established for radioactive materials in effluents and direct radiation levels associated with ISFSI or MRS operations to meet the limits given in paragraph (a) of this section.

§ 72.106 Controlled area of an ISFSI or MRS.

(a) For each ISFSI or MRS site, a controlled area must be established.

(b) Any individual located on or beyond the nearest boundary of the controlled area shall not receive a dose greater than 5 rem to the whole body or any organ from any design basis accident. The minimum distance from the spent fuel or high-level radioactive waste handling and storage facilities to the nearest boundary of the controlled area shall be at least 100 meters.

(c) The controlled area may be traversed by a highway, railroad or waterway, so long as appropriate and effective arrangements are made to control traffic and to protect public health and safety.

§ 72.108 Spent fuel or high-level radioactive waste transportation.

The proposed ISFSI or MRS must be evaluated with respect to the potential impact on the environment of the transportation of spent fuel or high-level radioactive waste within the region.

Subpart F—General Design Criteria

§ 72.120 General considerations.

(a) Pursuant to the provisions of § 72.24, an application to store spent fuel in an ISFSI or to store spent fuel or high-level radioactive waste in an MRS must include the design criteria for the proposed storage installation. These design criteria establish the design, fabrication, construction, testing, maintenance and performance requirements for structures, systems, and components important to safety as defined in § 72.3. The general design criteria identified in this subpart establish minimum requirements for the design criteria for an ISFSI or MRS. Any omissions in these general design criteria do not relieve the applicant from the requirement of providing the necessary safety features in the design of the ISFSI or MRS.

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(b) The MRS must be designed to store either spent fuel or solid high-level radioactive wastes. Liquid high-level radioactive wastes may not be received or stored in an MRS. If the MRS is a water pool type facility, the solidified waste form shall be a durable solid with demonstrable leach resistance.

§ 72.122 Overall requirements.

(a) *Quality Standards.* Structures, systems, and components important to safety must be designed, fabricated, erected, and tested to quality standards commensurate with the importance to safety of the function to be performed.

(b) *Protection against environmental conditions and natural phenomena.* (1) Structures, systems, and components important to safety must be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI or MRS and to withstand postulated accidents.

(2) Structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lighting, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform safety functions. The design bases for these structures, systems, and components must reflect:

(i) Appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take into account the limitations of the data and the period of time in which the data have accumulated, and

(ii) Appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena.

The ISFSI or MRS should also be designed to prevent massive collapse of building structures or the dropping of heavy objects as a result of building structural failure on the spent fuel or high-level radioactive waste or on to structures, systems, and components important to safety.

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(3) Capability must be provided for determining the intensity of natural phenomena that may occur for comparison with design bases of structures, systems, and components important to safety.

(4) If the ISFSI or MRS is located over an aquifer which is a major water resource, measures must be taken to preclude the transport of radioactive materials to the environment through this potential pathway.

(c) *Protection against fires and explosions.* Structures, systems, and components important to safety must be designed and located so that they can continue to perform their safety functions effectively under credible fire and explosion exposure conditions. Noncombustible and heat-resistant materials must be used wherever practical throughout the ISFSI or MRS, particularly in locations vital to the control of radioactive materials and to the maintenance of safety control functions. Explosion and fire detection, alarm, and suppression systems shall be designed and provided with sufficient capacity and capability to minimize the adverse effects of fires and explosions on structures, systems, and components important to safety. The design of the ISFSI or MRS must include provisions to protect against adverse effects that might result from either the operation or the failure of the fire suppression system.

(d) *Sharing of structures, systems, and components.* Structures, systems, and components important to safety must not be shared between an ISFSI or MRS

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and other facilities unless it is shown that such sharing will not impair the capability of either facility to perform its safety functions including the ability to return to a safe condition in the event of an accident.

(e) *Proximity of sites.* An ISFSI or MRS located near other nuclear facilities must be designed and operated to ensure that the cumulative effects of their combined operations will not constitute an unreasonable risk to the health and safety of the public.

(f) *Testing and maintenance of systems and components.* Systems and components that are important to safety must be designed to permit inspection, maintenance and testing.

(g) *Emergency capability.* Structures, systems, and components important to safety must be designed for emergencies. The design must provide for accessibility to the equipment of onsite and available offsite emergency facilities and services such as hospitals, fire and police departments, ambulance service, and other emergency agencies.

(h) *Confinement barriers and systems.* (1) The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. This may be accomplished by canning of consolidated fuel rods or unconsolidated assemblies or other means as appropriate.

(2) For underwater storage of spent fuel or high-level radioactive waste in which the pool water serves as a shield and a confinement medium for radioactive materials, systems for maintaining water purity and the pool water level must be designed so that any abnormal operations or failure in those systems from any cause will not cause the water level to fall below safe limits. The design must preclude installations of drains, permanently connected systems, and other features that could, by abnormal operations or failure, cause a significant loss of water. Pool water level equipment must be provided to alarm in a continuously manned location if the water level in the storage pools falls below a predetermined level.

(3) Ventilation systems and off-gas systems must be provided where necessary to ensure the confinement of airborne radioactive particulate materials during normal or off-normal conditions.

(4) Storage confinement systems must have the capability for continuous monitoring in a manner such that the

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licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions.

(5) The high-level radioactive waste must be packaged in a manner that allows handling and retrievability without the release of radioactive materials to the environment or radiation exposures in excess of Part 20 limits. The package must be designed to confine the high-level radioactive waste for the duration of the license.

(i) *Instrumentation and control systems.* Instrumentation and control systems must be provided to monitor systems that are important to safety over anticipated ranges for normal operation and off-normal operation. Those instruments and control systems that must remain operational under accident conditions must be identified in the Safety Analysis Report.

(j) *Control room or control area.* A control room or control area, if appropriate for the ISFSI or MRS design, must be designed to permit occupancy and actions to be taken to monitor the ISFSI or MRS safely under normal conditions, and to provide safe control of the ISFSI or MRS under off-normal or accident conditions.

(k) *Utility or other services.* (1) Each utility service system must be designed to meet emergency conditions. The design of utility services and distribution systems that are important to safety must include redundant systems to the extent necessary to maintain, with adequate capacity, the ability to perform safety functions assuming a single failure.

(2) Emergency utility services must be designed to permit testing of the functional operability and capacity, including the full operational sequence, of each system for transfer between normal and emergency supply sources; and to permit the operation of associated safety systems.

(3) Provisions must be made so that, in the event of a loss of the primary electric power source or circuit, reliable and timely emergency power will be provided to instruments, utility service systems, the central security alarm station, and operating systems, in amounts sufficient to allow safe storage conditions to be maintained and to permit continued functioning of all systems essential to safe storage.

(4) An ISFSI or MRS which is located on the site of another facility may share common utilities and services with such a facility and be physically connected with the other facility; however, the sharing of utilities and services or the physical connection must not significantly:

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(i) Increase the probability or consequences of an accident or malfunction of components, structures, or systems that are important to safety; or

(ii) Reduce the margin of safety as defined in the basis for any technical specifications of either facility.

(l) *Retrievability.* Storage systems must be designed to allow ready retrieval of spent fuel or high-level radioactive waste for further processing or disposal.

§ 72.124 Criteria for nuclear criticality safety.

(a) *Design for criticality safety.* Spent fuel handling, packaging, transfer, and storage systems must be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. The design of handling, packaging, transfer, and storage systems must include margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the data and methods used in calculations and demonstrate safety for the handling, packaging, transfer and storage conditions and in the nature of the immediate environment under accident conditions.

(b) *Methods of criticality control.* When practicable the design of an ISFSI or MRS must be based on favorable geometry, permanently fixed neutron absorbing materials (poisons), or both. Where solid neutron absorbing materials are used, the design shall provide for positive means to verify their continued efficacy.

(c) *Criticality Monitoring.* A criticality monitoring system shall be maintained in each area where special nuclear material is handled, used, or stored which will energize clearly audible alarm signals if accidental criticality occurs. Underwater monitoring is not required when special nuclear material is handled or stored beneath water shielding. Monitoring of dry storage areas where special nuclear material is packaged in its stored configuration under a license issued under this subpart is not required.

§ 72.126 Criteria for radiological protection.

(a) *Exposure control.* Radiation protection systems must be provided for all areas and operations where onsite personnel may be exposed to radiation or airborne radioactive materials. Structures, systems, and components for which operation, maintenance, and required inspections may involve occupational exposure must be designed, fabricated, located, shielded, controlled, and tested so as to control external and internal radiation exposures to personnel. The design must include means to:

(1) Prevent the accumulation of radioactive material in those systems requiring access;

(2) Decontaminate those systems to which access is required;

(3) Control access to areas of potential contamination or high radiation within the ISFSI or MRS;

(4) Measure and control contamination of areas requiring access;

(5) Minimize the time required to perform work in the vicinity of radioactive components; for example, by providing sufficient space for ease of operation and designing equipment for ease of repair and replacement; and

(6) Shield personnel from radiation exposure.

(b) *Radiological alarm systems.* Radiological alarm systems must be provided in accessible work areas as appropriate to warn operating personnel of radiation and airborne radioactive material concentrations above a given setpoint and of concentrations of radioactive material in effluents above control limits. Radiation alarm systems must be designed with provisions for calibration and testing their operability.

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(c) *Effluent and direct radiation monitoring.* (1) As appropriate for the handling and storage system, effluent systems must be provided. Means for measuring the amount of radionuclides in effluents during normal operations and under accident conditions must be provided for these systems. A means of measuring the flow of the diluting medium, either air or water, must also be provided.

(2) Areas containing radioactive materials must be provided with systems for measuring the direct radiation levels in and around these areas.

(d) *Effluent control.* The ISFSI or MRS must be designed to provide means to limit to levels as low as is reasonably achievable the release of radioactive materials in effluents during normal operations; and control the release of radioactive materials under accident conditions. Analyses must be made to show that releases to the general environment during normal operations and anticipated occurrences will be within the exposure limit given in § 72.104. Analyses of design basis accidents must be made to show that releases to the general environment will be within the exposure limits given in (e) Systems designed to monitor the release of radioactive materials must have means for calibration and testing their operability.

§ 72.128 Criteria for spent fuel, high-level radioactive waste, and other radioactive waste storage and handling.

(a) *Spent fuel and high-level radioactive waste storage and handling systems.* Spent fuel storage, high-level radioactive waste storage, and other systems that might contain or handle radioactive materials associated with spent fuel or high-level radioactive waste, must be designed to ensure adequate safety under normal and accident conditions. These systems must be designed with—

(1) A capability to test and monitor components important to safety,

(2) Suitable shielding for radioactive protection under normal and accident conditions,

(3) Confinement structures and systems,

(4) A heat-removal capability having testability and reliability consistent with its importance to safety, and

(5) means to minimize the quantity of radioactive wastes generated.

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(b) *Waste treatment.* Radioactive waste treatment facilities must be provided. Provisions must be made for the packing of site-generated low-level wastes in a form suitable for storage onsite awaiting transfer to disposal sites.

§ 72.130 Criteria for decommissioning.

The ISFSI or MRS must be designed for decommissioning. Provisions must be made to facilitate decontamination of structures and equipment, minimize the quantity of radioactive wastes and contaminated equipment, and facilitate the removal of radioactive wastes and contaminated materials at the time the ISFSI or MRS is permanently decommissioned.

Subpart G—Quality Assurance

§ 72.140 Quality assurance requirements.

(a) *Purpose.* This subpart describes quality assurance requirements applying to design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, modification of structures, systems, and components, and decommissioning that are important to safety. As used in this subpart, "quality assurance" comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. Quality assurance includes quality control, which comprises those quality assurance actions related to control of the physical characteristics and quality of the material or component to predetermined requirements.

(b) *Establishment of program.* Each licensee [4] shall establish, maintain, and execute a quality assurance program satisfying each of the applicable criteria of this subpart, and satisfying any specific provisions which are applicable to the licensee's activities. The licensee shall execute the applicable criteria in a graded approach to an extent that is commensurate with the importance to safety. The quality assurance program must cover the activities identified in § 72.24(n) throughout the life of the licensed activity, from the site selection through decommissioning, prior to termination of the license.

(c) *Approval of program.* Prior to receipt of spent fuel at the ISFSI or spent fuel and high-level radioactive waste at the MRS, each licensee shall obtain Commission approval of its quality assurance program. Each licensee shall file a description of its quality assurance program, including a discussion of which requirements of this subpart are applicable and how they will be satisfied, with the Director, Office of Nuclear Material and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

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(d) *Previously approved programs.* A Commission-approved quality assurance program which satisfies the applicable criteria of Appendix B to Part 50 of this chapter and which is established, maintained, and executed with regard to an ISFSI will be accepted as satisfying the requirements of paragraph (b) of this section. Prior to first use, the licensee shall 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 approved Appendix B program to ISFSI activities. The licensee shall identify the program by date of submittal to the Commission, docket number, and date of Commission approval.

§ 72.142 Quality assurance organization.

The licensee shall be responsible for the establishment and execution of the quality assurance program. The licensee may delegate to others, such as contractors, agents, or consultants, the work of establishing and executing the quality assurance program, but shall

[4] While the term "licensee" is used in these criteria, the requirements are applicable to whatever design, construction, fabrication, assembly, and testing is accomplished with respect to structures, systems, and components prior to the time a license is issued.

[illegible] performing the functions associated with attaining quality objectives and the quality assurance functions. The quality assurance functions are

(a) Assuring that an appropriate quality assurance program is established and effectively executed and

(b) Verifying by procedures such as checking, auditing, and inspection, that activities affecting the functions that are important to safety have been correctly performed. The persons and organizations performing quality assurance functions must have sufficient authority and organizational freedom to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions.

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The persons and organizations performing quality assurance functions shall report to a management level that ensures that the required authority and organizational freedom, including sufficient independence from cost and schedule considerations when these considerations are opposed to safety considerations, are provided. Because of the many variables involved, such as the number of personnel, the type of activity being performed, and the location or locations where activities are performed, the organizational structure for executing the quality assurance program may take various forms provided that the persons and organizations assigned the quality assurance functions have the required authority and organizational freedom. Irrespective of the organizational structure, the individual(s) assigned the responsibility for assuring effective execution of any portion of the quality assurance program at any location where activities subject to this section are being performed must have direct access to the levels of management necessary to perform this function.

§ 72.144 Quality assurance program.

(a) The licensee shall establish, at the earliest practicable time consistent with the schedule for accomplishing the activities, a quality assurance program which complies with the requirements of this subpart. The licensee shall document the quality assurance program by written procedures or instructions and shall carry out the program in accordance with these procedures throughout the period during which the ISFSI or MRS is licensed. The licensee shall identify the structures, systems, and components to be covered by the quality assurance program, the major organizations participating in the program, and the designated functions of these organizations.

(b) The licensee, through its quality assurance program, shall provide control over activities affecting the quality of the identified structures, systems, and components to an extent commensurate with the importance to safety, and as necessary to ensure conformance to the approved design of each ISFSI or MRS. The licensee shall ensure that activities affecting quality are accomplished under suitably controlled conditions. Controlled conditions include the use of appropriate equipment; suitable environmental conditions for accomplishing the activity, such as adequate cleanliness; and assurance that all prerequisites for the given activity have been satisfied. The licensee shall take into account the need for special controls, processes, test equipment, tools and skills to attain the required quality and the need for verification of quality by inspection and test.

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(c) The licensee shall base the requirements and procedures of its quality assurance program on the following considerations concerning the complexity and proposed use of the structures, systems, or components:

(1) The impact of malfunction or failure of the item on safety;

(2) The design and fabrication complexity or uniqueness of the item;

(3) The need for special controls and surveillance over processes and equipment;

(4) The degree to which functional compliance can be demonstrated by inspection or test; and

(5) The quality history and degree of standardization of the item.

(d) The licensee shall provide for indoctrination and training of personnel performing activities affecting quality as necessary to ensure that suitable proficiency is achieved and maintained. The licensee shall review the status and adequacy of the quality assurance program at established intervals. Management of other organizations participating in the quality assurance program shall regularly review the status and adequacy of that part of the quality assurance program which they are executing.

§ 72.146 Design control.

(a) The licensee shall establish measures to ensure that applicable regulatory requirements and the design basis, as specified in the license application for those structures, systems, and components to which this section applies, are correctly translated into specifications, drawings, procedures, and instructions. These measures must include provisions to ensure that appropriate quality standards are specified and included in design documents and that deviations from standards are controlled. Measures must be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the functions of the structures, systems, and components which are important to safety.

(b) The licensee shall establish measures for the identification and control of design interfaces and for coordination among participating design organizations. These measures must include the establishment of written procedures among participating design organizations for the review, approval, release, distribution, and revision of documents involving design interfaces. The design control measures must provide for verifying or checking the adequacy of design, by methods such as design reviews, alternate or simplified calculational methods, or by a suitable testing program. For the verifying or checking process, the licensee shall designate individuals or groups other than those who were responsible for the original design, but who may be from the same organization. Where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, the licensee shall include suitable qualification testing of a prototype or sample unit under the most adverse design conditions. The licensee shall apply design control measures to items such as the following: criticality physics, radiation, shielding, stress, thermal, hydraulic, and accident analyses; compatibility of materials; accessibility for inservice inspection, maintenance, and repair; features to facilitate decontamination; and delineation of acceptance criteria for inspections and tests.

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(c) The licensee shall subject design changes, including field changes, to design control measures commensurate with those applied to the original design. Changes in the conditions specified in the license require NRC approval.

§ 72.148 Procurement document control.

The licensee shall establish measures to assure that applicable regulatory requirements, design bases, and other requirements which are necessary to assure adequate quality are included or referenced in the documents for procurement of material, equipment, and services, whether purchased by the licensee or by its contractors or subcontractors. To the extent necessary, the licensee shall require contractors or subcontractors to provide a quality assurance program consistent with the applicable provisions of this subpart.

§ 72.150 Instructions, procedures, and drawings.

The licensee shall prescribe activities affecting quality by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall require that these instructions, procedures, and drawings be followed. The instructions, procedures, and drawings must include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

§ 72.152 Document control.

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The licensee shall establish measures to control the issuance of documents such as instructions, procedures, and drawings, including changes, which prescribe all activities affecting quality. These measures must assure that documents, including changes, are reviewed for adequacy, approved for release by authorized personnel, and distributed and used at the location where the prescribed activity is performed. These measures must ensure that changes to documents are reviewed and approved.

§ 72.154 Control of purchased material, equipment, and services.

(a) The licensee shall establish measures to ensure that purchased material, equipment and services, whether purchased directly or through contractors and subcontractors, conform to the procurement documents. These measures must include provisions, as appropriate, for source evaluation and selection, objective evidence of quality furnished by the contractor or subcontractor, inspection at the contractor or subcontractor source, and examination of products upon delivery.

(b) The licensee shall have available documentary evidence that material and equipment conform to the procurement specifications prior to installation or use of the material and equipment. The licensee shall retain or have available this documentary evidence for the life of ISFSI or MRS. The licensee shall ensure that the evidence is sufficient to identify the specific requirements met by the purchased material and equipment.

(c) The licensee or applicant shall assess the effectiveness of the control of quality by contractors and subcontractors at intervals consistent with the importance, complexity, and quantity of the product or services.

§ 72.156 Identification and control of materials, parts, and components.

The licensee shall establish measures for the identification and control of materials, parts, and components. These measures must ensure that identification of the item is maintained by heat number, part number, serial number, or other appropriate means, either on the item or on records traceable to the item as required, throughout fabrication, installation, and use of the item. These identification and control measures must be designed to prevent the use of incorrect or defective materials, parts, and components.

§ 72.158 Control of special processes.

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The licensee shall establish measures to ensure that special processes, including welding, heat treating, and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

§ 72.160 Licensee inspection.

The licensee shall establish and execute a program for inspection of activities affecting quality by or for the organization performing the activity to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. The inspection must be performed by individuals other than those who performed the activity being inspected. Examinations, measurements, or tests of material or products processed must be performed for each work operation where necessary to assure quality. If direct inspection of processed material or products cannot be carried out, indirect control by monitoring processing methods, equipment, and personnel must be provided. Both inspection and process monitoring must be provided when quality control is inadequate without both. If mandatory inspection hold points, which require witnessing or inspecting by the licensee's designated representative and beyond which work should not proceed without the consent of its designated representative, are required, the specific hold points must be indicated in appropriate documents.

§ 72.162 Test control.

The licensee shall establish a test program to ensure that all testing required to demonstrate that the structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures that incorporate the requirements of this part and the requirements and acceptance limits contained in the ISFSI or MRS license. The test procedures must include provisions for assuring that all prerequisites for the given test are met, that adequate test instrumentation is available and used, and that the test is performed under suitable environmental conditions. The licensee shall document and evaluate the test results to ensure that test requirements have been satisfied.

§ 72.164 Control of measuring and test equipment.

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The licensee shall establish measures to ensure that tools, gauges, instruments, and other measuring and testing devices used in activities affecting quality are properly controlled, calibrated, and adjusted at specified periods to maintain accuracy within necessary limits.

§ 72.166 Handling, storage, and shipping control.

The licensee shall establish measures to control, in accordance with work and inspection instructions, the handling, storage, shipping, cleaning, and preservation of materials and equipment to prevent damage or deterioration. When necessary for particular products, special protective environments, such as inert gas atmosphere, and specific moisture content and temperature levels must be specified and provided.

§ 72.168 Inspection, test, and operating status.

(a) The licensee shall establish measures to indicate, by the use of markings such as stamps, tags, labels, routing cards, or other suitable means, the status of inspections and tests performed upon individual items of the ISFSI or MRS. These measures must provide for the identification of items which have satisfactorily passed required inspections and tests where necessary to preclude inadvertent bypassing of the inspections and tests.

(b) The licensee shall establish measures to identify the operating status of structures, systems, and components of the ISFSI or MRS, such as tagging valves and switches, to prevent inadvertent operation.

§ 72.170 Nonconforming materials, parts, or components.

The licensee shall establish measures to control materials, parts, or components that do not conform to the licensee's requirements in order to prevent their inadvertent use or installation. These measures must include, as appropriate, procedures for identification, documentation, segregation, disposition, and notification to affected organizations. Nonconforming items must be reviewed and accepted, rejected, repaired, or reworked in accordance with documented procedures.

§ 72.172 Corrective action.

The licensee shall establish measures to ensure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances, are promptly identified and corrected. In the case of a significant condition adverse to quality, the measures must ensure that the cause of the condition is determined and corrective action is taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken must be documented and reported to appropriate levels of management.

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§ 72.174 Quality assurance records.

The licensee shall maintain sufficient records to furnish evidence of activities affecting quality. The records must include the following: design records, records of use and the results of reviews, inspections, tests, audits, monitoring of work performance, and materials analyses. The records must include closely related data such as qualifications of personnel, procedures, and equipment. Inspection and test records must, at a minimum, identify the inspector or data recorder, the type of observation, the results, the acceptability, and the action taken in connection with any noted deficiencies. Records must be identifiable and retrievable. Records pertaining to the design, fabrication, erection, testing, maintenance, and use of structures, systems, and components important to safety shall be maintained by or under the control of the licensee until the Commission terminates the license.

§ 72.176 Audits.

The licensee shall carry out a comprehensive system of planned and periodic audits to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program. The audits must be performed in accordance with written procedures or checklists by appropriately trained personnel not having direct responsibilities in the areas being audited. Audited results must be documented and reviewed by management having responsibility in the area audited. Follow-up action, including re-audit of deficient areas, must be taken where indicated.

Subpart H—Physical Protection

§ 72.180 Physical security plan.

The licensee shall establish a detailed plan for security measures for physical protection. The licensee shall retain a copy of the current plan as a record until the Commission terminates the license for which the procedures were developed and, if any portion of the plan is superseded, retain the superseded material for three years after each change. This plan must consist of two parts. Part I must demonstrate how the applicant plans to comply with the applicable requirements of Part 73 of this chapter and during transportation to and from the proposed ISFSI or MRS and must include the design for physical protection and the licensee's safeguards contingency plan and guard training plan. Part II must list tests, inspections, audits, and other means to be used to demonstrate compliance with such requirements.

§ 72.182 Design for physical protection.

The design for physical protection must show the site layout and the design features provided to protect the ISFSI or MRS from sabotage. It must include:

(a) The design criteria for the physical protection of the proposed ISFSI or MRS;

(b) The design bases and the relation of the design bases to the design criteria submitted pursuant to paragraph (a) of this section; and

(c) Information relative to materials of construction, equipment, general arrangement, and proposed quality assurance program sufficient to provide reasonable assurance that the final security system will conform to the design bases for the principal design criteria submitted pursuant to paragraph (a) of this section.

§ 72.184 Safeguards contingency plan.

(a) The requirements of the licensee's safeguards contingency plan for dealing with threats and radiological sabotage must be as defined in § 73.40(b) of this chapter. This plan must include Background, Generic Planning Base, Licensee Planning Base, and Responsibility Matrix, the first four categories of information relating to nuclear facilities licensed under Part 50 of this chapter. (The fifth category of information, Procedures, does not have to be submitted for approval.)

(b) The licensee shall prepare and maintain safeguards contingency plan procedures in accordance with Appendix C to 10 CFR Part 73 for effecting the actions and decisions contained in the Responsibility Matrix of the licensee's safeguards contingency plan. The licensee shall retain a copy of the current procedures as a record until the Commission terminates the license for which the procedures were developed and, if any portion of the procedures is superseded, retain the superseded material for three years after each change.

§ 72.186 Change to physical security and safeguards contingency plans.

(a) The licensee shall make no change that would decrease the safeguards effectiveness of the physical security plan, guard training plan or the first four categories of information (Background, Generic Planning Base, Licensee Planning Base, and Responsibility Matrix) contained in the licensee safeguards contingency plan without prior approval of the Commission. A licensee desiring to make a change must submit an application for a license amendment pursuant to § 72.56.

(b) The licensee may, without prior Commission approval, make changes to the physical security plan, guard training plan, or the safeguards contingency plan, if the changes do not decrease the safeguards effectiveness of these plans. The licensee shall maintain records of changes to any such plan made without prior approval for a period of three years from the date of the change and shall furnish to the Regional Administrator of the appropriate NRC Regional Office specified in Appendix A of Part 73 of this chapter, with a copy to the Director, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555, a report containing a description of each change within two months after the change is made.

Subpart I—Training and Certification of Personnel

§ 72.190 Operator requirements.

Operation of equipment and controls that have been identified as important to safety in the Safety Analysis Report and in the license must be limited to trained and certified personnel or be under the direct visual supervision of an individual with training and certification in the operation. Supervisory personnel who personally direct the operation of equipment and controls that are important to safety must also be certified in such operations.

§ 72.192 Operator training and certification program.

The applicant for a license under this part shall establish a program for training, proficiency testing, and certification of ISFSI or MRS personnel. This program must be submitted to the Commission for approval with the license application.

§ 72.194 Physical requirements.

The physical condition and the general health of personnel certified for the operation of equipment and controls that are important to safety must not be such as might cause operational errors that could endanger other in-plant personnel or the public health and safety. Any condition that might cause impaired judgment or motor coordination must be considered in the selection of personnel for activities that are important to safety. These conditions need not categorically disqualify a person, if appropriate provisions are made to accommodate such defect.

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Subpart J—Provision of MRS Information to State Governments and Indian Tribes

§ 72.200 Provision of MRS information.

(a) The Director, Office of Nuclear Material Safety and Safeguards, or the Director's designee shall provide to the Governor and legislature of any State in which an MRS authorized under the Nuclear Waste Policy Act of 1982, as amended, is or may be located, to the Governors of any contiguous States, to each affected unit of local government and to the governing body of any affected Indian tribe, timely and complete information regarding determinations or plans made by the Commission with respect to siting, development, design, licensing, construction, operation, regulation or decommissioning of such monitored retrievable storage facility.

(b) Notwithstanding paragraph (a) of this section, the Director or the Director's designee is not required to distribute any document to any entity if, with respect to such document, that entity or its counsel is included on a service list prepared pursuant to Part 2 of this chapter.

(c) Copies of all communications by the Director or the Director's designee under this section shall be placed in the Commission's Public Document Room and shall be furnished to DOE.

§ 72.202 Participation in license reviews.

State and local governments and affected Indian tribes may participate in license reviews as provided in Subpart C of Part 2 of this chapter.

§ 72.204 Notice to States.

If the Governor and legislature of a State have jointly designated on their behalf a single person or entity to receive notice and information from the Commission under this part, the Commission will provide such notice and information to the jointly designated person or entity instead of the Governor and the legislature separately.

§ 72.206 Representation.

Any person who acts under this subpart as a representative for a State (or for the Governor or legislature thereof) or for an affected Indian tribe shall include in the request or other submission, or at the request of the Commission, a statement of the basis of his or her authority to act in such representative capacity.

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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 possess or 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 that spent fuel which the general licensee is authorized to possess at the site under the specific license for the site.

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

(3) The general license for the storage of spent fuel in each cask fabricated under a Certificate of Compliance terminates 20 years after the date that the particular cask is first used by the general licensee to store spent fuel, unless the cask's Certificate of Compliance is renewed, in which case the general license terminates 20 years after the cask's Certificate of Compliance renewal date. In the event that a cask vendor does not apply for a cask model reapproval under § 72.240, any cask user or user's representative may apply for a cask design reapproval. If a Certificate of Compliance expires, casks of that design must be removed from service after a storage period not to exceed 20 years.

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(b) The general licensee shall:

(1)(i) Notify the Nuclear Regulatory Commission using instructions in § 72.4 at least 90 days prior to first storage of spent fuel under this general license. The notice may be in the form of a letter, but must contain the licensee's name, address, reactor license and docket numbers, and the name and means of contacting a person responsible for providing additional information concerning spent fuel under this general license. 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 of this chapter.

(ii) Register use of each cask with the Nuclear Regulatory Commission no later than 30 days after using that cask to store spent fuel. This registration may be accomplished by submitting a letter using instructions in § 72.4 containing the following information: the licensee's name and address, the licensee's reactor license and docket numbers, the name and title of a person responsible for providing additional information concerning spent fuel storage under this general license, the cask certificate and model numbers, and the cask identification number. 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 of this chapter.

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(iii) Fee. Fees for inspections related to spent fuel storage under this general license are those shown in § 170.31 of this chapter.

(2) Perform written evaluations, prior to use, that establish that (i) conditions set forth in the Certificate of Compliance have been met; (ii) cask storage pads and areas have been designed to adequately support the static load of the stored casks; and (iii) the requirements of § 72.104 have been 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 Safety Analysis Report (SAR) referenced in the Certificate of Compliance and the related NRC Safety Evaluation Report, prior to use of the general license, to determine whether or not the reactor site parameters, including analyses of earthquake intensity and tornado missiles, are enveloped by the cask design bases considered in these reports. The results of this review must be documented in the evaluation made in paragraph (b)(2) of this section.

(4) Prior to use of the general license, determine whether activities related to

storage of spent fuel under this general license involve any unreviewed facility safety question or change in the facility technical specifications, as provided under § 50.59. Results of this determination must be documented in the evaluation made in paragraph (b)(2) of this section.

(5) Protect the spent fuel against the design basis threat of radiological sabotage in accordance with the same provisions and requirements as are set forth in the licensee's physical security plan pursuant to § 73.55 of this chapter with the following additional conditions and exceptions.

(i) The physical security organization and program for the facility must be modified as necessary to assure that activities conducted under this general license do not decrease the effectivens 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.

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(iii) For purposes of this general license, searches required by § 73.55(d)(1) of this chapter before admission to a new protected area may be performed by physical pat-down searches of persons in lieu of firearms and explosives detection equipment.

(iv) The observational capability required by § 73.55(h)(6) of this chapter as applied to a new protected area may be provided by a guard or watchman on patrol in lieu of closed circuit television.

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

(6) Review the reactor emergency plan, quality assurance program, training program, and radiation protection program to determine if their effectiveness is decreased and, if so, prepare the necessary changes and seek and obtain the necessary approvals.

(7) Maintain a copy of the Certificate of Compliance and documents referenced in the certificate for each cask model 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) Accurately maintain the record provided by the cask supplier for each cask that shows, in addition to the information provided by the cask vendor, the following:

(A) The name and address of the cask vendor or lessor;

(B) The listing of spent fuel stored in the cask; and

(C) 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 NRC-approved quality assurance program.

(iii) In the event that a cask is sold, leased, loaned, or otherwise transferred to another registered user, 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 only in accordance with written procedures.

(10) Make records and casks available to the Commission for inspection.

§ 72.214 List of approved spent fuel storage casks.

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The following casks are approved for storage of spent fuel under the conditions specified in their Certificates of Compliance.

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)
Docket Number: 72-1000
Certification Expiration Date: August 17, 2010
Model Number: CASTOR V/21

Certificate Number: 1001
SAR Submitted by: Westinghouse Electric Corporation
SAR Title: Topical Safety Analysis Report for the Westinghouse MC-10 Cask for an Independent Spent Fuel Storage Installation (Dry Storage)
Docket Number: 72-1001
Certification Expiration Date: August 17, 2010
Model Number: MC-10

Certificate Number: 1002
SAR Submitted by: Nuclear Assurance Corporation
SAR Title: Topical Safety Analysis Report for the NAC Storage/Transport Cask for Use at an Independent Spent Fuel Storage Installation
Docket Number: 72-1002
Certification Expiration Date: August 17, 2010
Model Number: NAC S/T

Certificate Number: 1003
SAR Submitted by: Nuclear Assurance Corporation
SAR Title: Topical Safety Analysis Report for the NAC Storage/Transport Cask Containing Consolidated Fuel for Use at an Independent Spent Fuel Storage Installation
Docket Number: 72-1003
Certification Expiration Date: August 17, 2010
Model Number: NAC-C28 S/T

§ 72.216 Reports.

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

(1) Defect discovered in any spent fuel storage cask structure, system, or component which is important to safety, or

(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 using instructions in § 72.4 within 30 days of the report submitted in paragraph (a) of this section. 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.

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§ 72.218 Termination of licenses.

(a) The notification regarding the program for the management of 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) An application for termination of the reactor operating license submitted under § 50.82 of this chapter must contain a description of how the spent fuel stored under this general license will be removed from the reactor site.

(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.

This general license is subject to the provisions of § 72.84 for violation of the regulations under this part.

Subpart L—Approval of Spent Fuel Storage Casks

§ 72.230 Procedures for spent fuel storage cask submittals.

(a) An application for approval of a spent fuel storage cask design 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 for the cask under part 71 of this chapter, 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.

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(d) *Fees.* Fees for reviews and evaluations related to issuance of a spent fuel storage cask Certificate of Compliance and inspections related to storage cask fabrication 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 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 requirements 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) Fabrication of casks under the Certificate of Compliance 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 was started;

(v) Date fabrication was 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 current copy of a composite record of all casks manufactured under a Certificate of Compliance, showing the information in paragraph (d)(2) of this section must be initiated and maintained by the cask vendor for each model cask. If the cask vendor permanently ceases production of casks under a Certificate of

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Compliance, this composite 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 available to the Commission for inspection.

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

§ 72.236 Specific requirements for spent fuel storage cask approval.

(a) Specification must be provided for the spent fuel to be stored in the cask, such as, but not limited to, type of spent fuel (i.e., BWR, PWR, both), maximum allowable enrichment of the fuel prior to any irradiation, burn-up (i.e., megawatt-days/MTU), minimum acceptable cooling time of the spent fuel prior to storage in the cask, maximum heat designed to be dissipated, maximum spent fuel loading limit, condition of the spent fuel (i.e., intact assembly or consolidated fuel rods), the inerting atmosphere requirements.

(b) Design bases and design criteria must be provided for structures, systems, and components 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 sufficient to meet the requirements in §§ 72.104 and 72.106.

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

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

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

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

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

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

(k) The cask must be conspicuously and durably marked with:

(1) A model number;

(2) A unique identification number; and

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(3) An empty weight.

(l) The cask and its systems important to safety must be evaluated, by appropriate tests or by other means acceptable to the Commission, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

(m) To the extent practicable in the design of storage casks, consideration should be given to compatibility with removal of the stored spent fuel from a reactor site, transportation, and 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 the requirements in § 72.236 (a) through (i) are met.

§ 72.240 Conditions for spent fuel storage cask reapproval.

(a) The holder of a cask Certificate of Compliance, a user of a cask approved by NRC, or the representative of a cask user must apply for a cask model reapproval.

(b) The application for reapproval of a cask model must be submitted not less than 30 days prior to the expiration date of the Certificate of Compliance. When the applicant has submitted a timely application for reapproval, the existing Certificate of Compliance will not expire until the application for reapproval has

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been finally determined by the Commission. The application must be accompanied by a safety analysis report (SAR). The new SAR may reference the SAR originally submitted for the cask model approval.

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

PART 72

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

PROPOSED RULE MAKING

50 FR 23960
Published 6/7/85
Comment period expires 10/7/85.

Financial Responsibility Requirements Applicable to NRC Licensees for Cleanup of Accidental and Unexpected Releases of Radioactive Materials

See Part 30 Proposed Rulemaking

50 FR 41904
Published 10/16/85
Comment period extended 11/7/85

Financial Responsibility Requirements Applicable to NRC Licensees for Cleanup of Accidental and Unexpected Releases of Radioactive Materials

See Part 30 Proposed Rule Making

➤ **54 FR 19379**
Published 5/5/89
Comment period expires 6/19/89

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

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.

ADDRESS: 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.

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 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 NWPA 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 NWPA. 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 18 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

10 CFR 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.

10 CFR Part 72

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

10 CFR 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, 98 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)(3), (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 licensees 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, 50.53(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 (b)(2)(vii) 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 revised 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

[1] Types 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 29th day of April 1988.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

54 FR 30049
Published 7/18/89
Comment period expires 9/18/89.

Preserving the Free Flow of Information to the Commission

See Part 30 Proposed Rule Making

54 FR 33570
Published 8/15/89
Comment period expires 9/29/89.

Minor Amendments to the Physical Protection Requirements

See Part 73 Proposed Rule Making

> **55 FR 12374**
Published 4/3/90.
Comment period expires 6/18/90.

Willful Misconduct by Unlicensed Persons

See Part 30 Proposed Rule Making

55 FR 13542
Published 4/11/90

Willful Misconduct by Unlicensed Persons (correction)

See Part 30 Proposed Rule Making

UNITED STATES NUCLEAR REGULATORY COMMISSION

RULES and REGULATIONS

TITLE 10, CHAPTER 1, CODE OF FEDERAL REGULATIONS — ENERGY

PART 72

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

STATEMENTS OF CONSIDERATION

52 FR 1292
Published 1/12/87
Effective 2/11/87

Bankruptcy Filing: Notification Requirements

Part 30 Statements of Consideration

52 FR 21651
Published 6/9/87
Effective 10/8/87

Changes to Safeguards Reporting Requirements

See Part 70 Statements of Consideration

52 FR 31601
Published 8/21/87
Effective 9/18/87

Statement of Organization and General Information

See Part 1 Statements of Consideration

52 FR 49362
Published 12/31/87
Effective 2/1/88

Completeness and Accuracy of Information

See Part 2 Statements of Consideration

53 FR 4109
Published 2/12/88
Effective 2/12/88

Redesignation of NRC Offices—NMSS, OI and GPA

See Part 30 Statements of Consideration

53 FR 24018
Published 6/27/88
Effective 7/27/88

General Requirements for Decommissioning Nuclear Facilities

See Part 30 Statements of Consideration

53 FR 31651
Published 8/19/88
Effective 9/19/88

10 CFR Parts 2, 19, 20, 21, 51, 70, 72, 73, 75 and 150

Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Nuclear Waste Policy Act of 1982, as amended (NWPA) requires that monitored retrievable storage facilities (MRS) for spent nuclear fuel and high-level radioactive waste (HLW) be subject to licensing by the Nuclear Regulatory Commission (NRC). The NRC is adding language to its regulations in 10 CFR Part 72 to

provide for licensing the storage of spent nuclear fuel and HLW in an MRS. The Commission intends to have the appropriate regulation to fulfill the requirements of the NWPA in place in a timely manner. The rule would also clarify certain issues that have arisen since Part 72 was made effective on November 28, 1980 and incorporate other changes resulting from public comments received.

EFFECTIVE DATE: September 19, 1988.

ADDRESSES: Copies of NUREG-0575, NUREG-1092, and NUREG-1140 may be purchased from the Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013-7082. Copies are also available from the National Technical Information Service, 5285 Port Royal Road, Springfield, VA 22161. A copy of each NUREG is also available for public inspection and/or copying at the NRC Public Document Room, 1717 H Street NW., Washington, DC.

FOR FURTHER INFORMATION CONTACT: Keith G. Steyer or C.W. Nilsen, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 492-3824 or 492-3834, respectively.

SUPPLEMENTARY INFORMATION: On May 27, 1986, following Commission approval, the proposed revision to 10 CFR Part 72 relating to MRS licensing was published in the Federal Register (51 FR 19106) for comment. The comment period expired on August 25, 1986.

The NRC received 195 comment letters from utilities, engineering companies, State offices, environmental groups, private citizens, and a member of the U.S. House of Representatives. The comment letters from private citizens numbered about 145. (Some of these were signed by several individuals or were submitted on behalf of private business firms.) From the comment letters received, the staff identified 27 separate topics to which specific responses were directed. Comments were also received which addressed the original rule, not the proposed amendment. In response to the comments, several changes have been made to the proposed rule. The majority of these changes are mainly clarifying in nature.

In order to provide sufficient space to accommodate possible future amendments to Part 72, the sections of the final rule have been renumbered. To aid the reader in following the discussion of comments in the preamble of the final rule, each reference to a specific section of the final rule is followed by a bracketed reference to the parallel section of the proposed rule.

A compilation of the issues raised as a result of public comment and the accompanying Commission response follow:

1. Backfitting

Comment: Several commenters indicated that the proposed rule should incorporate the sense of the reactor backfitting rule set out in 10 CFR 50.109.

Response: Although these storage facilities are not like reactors but are, for the most part, static by nature with very little need for design changes, the staff has revised the backfitting requirements of 10 CFR 72.62 (§ 72.42). The change is being made to conform § 72.62 (§ 72.42) more closely to § 50.109 as modified by the court decision in *Union of Concerned Scientists, et al., v. U.S. Nuclear Regulatory Commission, et al.,* Nos. 85-1757 and 86-1219, 824 F.2d 108 (U.S.C.A.D.C. August 4, 1987).

2. Opportunity for Hearing Prior to the First Receipt of Spent Fuel or High-Level Radioactive Waste (HLW)

Comment: A new proposed § 72.46(c) (§ 72.34(c)) was added to 10 CFR Part 72 specifically providing that the Commission may, upon its own initiative, issue a notice of opportunity for hearing prior to the first receipt of spent fuel or high-level radio-active waste at an MRS if it finds this to be in the public interest. In the supplementary information in the May 27, 1986 Proposed Rule, the Commission indicated its own considerations on this topic and expressed particular interest in receiving public comment on (1) the need to make a finding before MRS operation that construction conforms to the license application, (2) provisions for second stage hearing rights to address specific new issues which could not have been litigated at the first stage and/or new information which has been revealed since issuance of the license, and (3) the format of the hearing, if held. Of the comment letters that addressed these points, some expressed no preference, some favored the provisions, some thought the provisions were unnecessary.

The principal reasons given by proponents of these provisions are that the public will have more confidence that the MRS will be operated safely and that there should be a clear opportunity to examine new issues which could be raised. Other comments of proponents were that the Department of Energy has had poor public performance in the past, that the degree of hazard is similar to nuclear power reactors which require a two-stage process, and that the opportunity for a second hearing could be an appropriate time to examine technical/financial information. Additional comments suggested that the rule require a second mandatory hearing and that funding be provided for nonprofit groups to participate in a second hearing.

On the topic of a finding it was suggested that (1) criteria be set forth for any finding the Commission may make, and (2) the NRC inspections should certify quality assurance and completeness of construction in an inspection report prior to initiation of operation. One comment suggested that start-up of the MRS should be linked to the repository authorization as an issue at a second hearing.

The principal reasons given by those opposed to the new provisions for a second hearing were that (1) it would cause unnecessary delay, (2) the Commission's regulations in 10 CFR Part 2 were sufficient to examine any new issues, (3) the NRC's normal systematic inspections are adequate to assure that construction was proper, (4) the nature of the MRS is such that all issues could be covered by the opportunity for public review prior to issuing a license and starting construction, and (5) the backfitting provision (§ 72.62 (§ 72.42)) provides additional assurance that significant issues may be raised by staff after the license is issued. Other reasons offered in objection to the new provisions were that (6) there was no basic difference between an MRS and an Independent Spent Fuel Storage Installation (ISFSI), (7) the small amount of solidified high-level waste which could be received could not justify any change in procedure from an ISFSI, and (8) the Safety Analysis Report (SAR) update procedure will assure that any new issue will be known and understood by NRC staff.

Response: The Commission specifically added the new provision and requested comments in order to obtain as complete an understanding as possible of whether or not any benefits would accrue to the public from such a procedure. This was done with full knowledge that the Atomic Energy Act of 1954, as amended, only requires one hearing and that under the procedures in 10 CFR Part 2 the opportunity always exists for any member of the public to bring any new issues to the Commission's attention.

In the comments received from the public there was no indication that there were likely to be any new safety issues brought forward which could not have been fully addressed on the occasion of the hearing held prior to issuance of the license. The licensing process of Part 72 supports one-stage licensing as it requires that all information needed for the licensing action be available and complete before a license is issued, i.e., final design, quality assurance/control procedures, operator training procedures, operating technical specifications, etc. Unlike a reactor license where a construction permit is issued prior to final design, an MRS application for license contains a final and complete design and therefore one-stage licensing is achievable. As to conformance of construction with the application and license, the Commission believes that, unlike reactors, [illegible] Part 72 type facilities will be simple and straightforward. Accordingly, in the Commission's judgment, there will be no need, as part of the safety review prior to license issuance, to require an applicant to "prove" conformance of the as built facility with the application. NRC would audit construction progress and, in the event some problems were found, enforcement action could be taken to correct them and, if necessary, halt the receipt of spent fuel until they were corrected. In this regard, § 72.82(c)(3) (§ 72.50(c)(3)) provides for establishing an NRC resident inspection program if warranted.

3. Interaction with States

Comment: Comments were received concerning providing of information to State and local governments and their interaction in the licensing process with DOE and the Commission.

Response: Under § 72.200 (§ 72.310) of the proposed rule, the Governor and legislature of any State in which a monitored retrievable storage installation may be located and the governing body of any affected Indian tribe will be provided timely and complete information regarding determinations or plans made by the Commission with respect to siting, development, design, licensing, construction, operation, regulation or decommissioning of such monitored retrievable storage facility. In response to the comment, the Commission will change § 72.200 (§ 72.310) "Provision of MRS Information" to require that the above information will also be provided to each affected unit of local government and to the Governors of any contiguous States. The definition of "affected unit of local government" which has been added to § 72.3 tracks the definition used in the Nuclear Waste Policy Amendments Act of 1987. (Sec. 5002, Pub. L. 100–203, 101 Stat. 1330–227 (42 U.S.C. 10101 (31)).) Participation by persons, including States, in license reviews is as provided for in 10 CFR Part 2, Subpart G.

4. High Burn-Up Fuel

Comment: In response to a 1980 petition for rulemaking, the Commission agreed (51 FR 23233, June 26, 1986) to prepare an environmental assessment on high burn-up fuel. The Commission's response concerning impacts of high burn-up fuel should be provided.

Response: The Commission issued an environmental assessment addressing the subject of high burn-up fuel in February 1988 "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors" (NUREG/CR-5009). The assessment concluded "Environmentally, this burnup increase would have no significant impact over normal burnup."

5. Emergency Planning

Comment: As discussed in supplementary information to the proposed revisions to 10 CFR Part 72 the rule was rewritten to set forth explicit requirements appropriate to an ISFSI or an MRS, rather than refer to Appendix E to CFR Part 50, which is specific to nuclear power reactors. Responders commented on this change. Several thought that there should be a wider dissemination of the emergency plan which an applicant would have to prepare pursuant to the rewritten § 72.32 (§ 72.19), as well as a comment period longer than the specified 60 days. Another responder thought that 60 days was adequate. Other comments were that (1) sabotage of casks and terrorism, sabotage and military attack scenarios should be considered in an emergency plan, (2) a fully developed and tested offsite emergency plan should be developed, (3) the new version of § 72.32 (§ 72.19) implies a need for offsite protective actions which is incorrect, (4) the supplementary information which will accompany the issuance of the final rule should discuss worldwide experience and previous reviews and studies as support for the new emergency planning provisions, and (5) the emergency plan should continue to be the same as that for nuclear power reactors.

Response: The basic concept of emergency planning in § 72.32 (§ 72.19) has not been changed. None of the respondents provided any additional information to the staff or questioned the staff analyses such as to change the basis for the staff's approach to emergency planning for an ISFSI or an MRS. Moreover, in view of the relatively passive nature of facilities for the receipt, handling and storage of spent fuel and high-level radioactive waste, as compared to operating power reactors, emergency plans for ISFSI and MRS need not be equivalent to emergency plans for reactors.

Since the proposed revision of Part 72 was published for comment on May 27, 1986, the NRC has published proposed amendments to 10 CFR Parts 30, 40, and 70[1] which would require certain NRC fuel cycle and other radioactive materials licensees that engage in activities that may have the potential for a significant accidental release of NRC-licensed materials to establish and maintain approved emergency plans for responding to such accidents. Although applicable to persons licensed under different parts of the Commission's regulations, the proposed requirements for emergency plans in Parts 30, 40, and 70 contain substantially identical provisions because they are designed to protect the public against similar radiological hazards. The proposed revision of Part 72 as published for comment also requires applicants for an ISFSI or MRS license to submit an emergency plan (see § 72.32 (§ 72.19).) Although the texts of proposed § 72.19 (redesignated § 72.32) and the parallel provisions of the proposed Emergency Preparedness rule are not identical, these provisions have the same purpose and use the same approach. In both cases, the proposed regulations require onsite emergency planning with provisions for offsite emergency response in terms of coordination and communication with offsite authorities and the public. It is therefore appropriate that in both cases these requirements should be expressed in the same way.

Until the Commission promulgates the Emergency Preparedness rule in final form, it is not possible to ascertain exactly the language that should be used. In view of these circumstances and since there is every expectation that this period of uncertainty will be of relatively short duration, we believe the prudent course of action is to reserve § 72.32 (§ 72.19), Emergency plan, in the final rule with the understanding that the text of this section will be promulgated in final form as a conforming amendment when the Commission adopts and promulgates the final Emergency Preparedness rule or shortly thereafter. We should point out that the temporary absence from Part 72 of requirements respecting emergency plans does not present any difficulties from a regulatory standpoint. To date, only three licenses have been issued under Part 72. Two licensees also hold Part 50 licenses and are required to comply with the provisions respecting emergency plans set out in the Part. The Part 72 license held by the third licensee contains conditions relating to emergency planning with which that licensee must comply.

Sabotage, terrorism, and military attacks are not treated as emergency preparedness issues. The Commission's established practice with respect to dangers of enemy action is that the protection of the United States against hostile enemy acts is a responsibility of the nation's defense establishment and the various agencies having internal security functions. Acts other than military are covered under a planning system included in Subpart H of Part 72.

[1] Proposed rule on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees 52 FR 12921, April 20, 1987.

which contains requirements respecting physical security and safeguards contingency plans that are specifically designed to preclude the occurrence of such acts. The primary purpose of an emergency response plan is to prescribe measures to be taken to mitigate the effects of accidental releases of radioactivity, irrespective of their cause. Thus, in the unlikely event that there should be an accidental release of radioactivity by reason of an act of terrorism or an act of sabotage, protective actions would be taken as prescribed in the emergency response plan, just as they would be taken in the case of accidental release arriving from other causes.

6. Department of Energy as Licensee for the MRS

Comment. Respondents commented on several aspects of the licensing of the Department of Energy for the MRS. One commenter requested that in every instance in which there would be a difference in requirement between the Department and other licensees, that that difference should be specifically defined in Part 72. Other commenters pointed out that the funding for the MRS was from the Nuclear Waste Fund as stipulated in the NWPA and, therefore, the Department should be required, through Part 72, to show how these funds will be adequate for operation and decommissioning. A further commenter questioned the Department's authority pursuant both to Part 72 and its own orders to delegate quality assurance responsibilities to its contractor(s). One commenter suggested that Part 72 should permit revocation or suspension of the Department's license for the MRS since the NRC could not impose civil penalties for license violations.

Response: As discussed in the supplementary information to the proposed revisions to Part 72, the Department of Energy is exempted from certain financial reports, creditor information and financial plans for decommissioning. As pointed out in the comment above, funding for the MRS will be from the Nuclear Waste Fund, separately accountable from public funds. Consistent with the principle of full cost recovery in section 302 of the NWPA (96 Stat. 2257, 42 U.S.C. 10222) this fund will provide all financial resources for the MRS, i.e., licensing, construction, operation and decommissioning. Since DOE is a federal agency and the status of the NWPA waste fund is reported to and reviewed by the Congress yearly, the Commission believes that Congress will assure that adequate funds are available and appropriated for DOE to carry out its statutory responsibility. Under these circumstances additional NRC oversight is unnecessary and inappropriate.

As to possible conflicts in the licensing and regulatory process between orders and procedures of the Department of Energy and NRC requirements, two government agencies, the commenter provided no specifics and the Commission is not aware of any such conflict. The Department will be provided the same latitude as any other licensee pursuant to § 72.142 (§ 72.101) wherein it is stated that "the licensee may delegate to others, such as contractors, agents, or consultants, the work of establishing and executing the quality assurance program, but shall retain responsibility for the program."

The Energy Reorganization Act of 1974, as amended, and the Nuclear Waste Policy Act of 1982, as amended, provide that upon authorization by Congress an MRS shall be subject to licensing by the Commission. Accordingly, no exemptions from the provisions of § 72.60 (§ 72.41), "Modification, revocation, and suspension of licenses" and § 72.84 (§ 72.57), "Violation" are shown for the Department. In the exercise of this broad statutory authority and consistent with its customary practice in regulating other Federal licensees, the Commission may impose penalties on the Department if there is sufficient justification. The Commission knows of no other differences between the Department and other licensees for which a change in Part 72 is warranted. (The commenters recommended no specific changes in this area.)

7. Minimum Decay Period (Age) for Receipt of Spent Fuel

Comment: It was noted that there is a seeming discrepancy between the minimum decay period (age) of spent fuel as specified in § 72.2 (one year) and a reference to the environmental analysis in NUREG–1140, "A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees" (five-year decay assumed).

Response: The minimum one-year decay period in § 72.2 is based on assuming the decay of radioisotopes having half-lives on the order of a few days or less. In actuality, the decay periods are likely to be much longer than one year. Accordingly, the NUREG-1140 analyses were based on the more realistic, but still conservative, assumption that five or more years of decay would have taken place for the spent fuel for which an accident in a dry cask was assumed. This is not a discrepancy since different purposes are being served in each instance. In choosing a nominal decay period of 10 years and a five-year minimum decay period in the design parameters for the MRS the Department of Energy (DOE) is merely exercising its own prerogative to use a longer decay criterion for purposes of fuel receipt. Selection of a five-year minimum decay period also reflects DOE's understanding that the spent fuel to be received at the MRS will already have decayed for periods of time likely to be even much greater than five years at individual power reactor sites. The original analysis for Part 72 was based on one-year decay.

8. Physical Security Plan

Comment: A few commenters were concerned about the proposed change in the requirements of the physical security plan for the Department of Energy in that the Department must provide a certification that it will provide at the MRS "such safeguards as it requires at comparable surface DOE facilities to promote the common defense and security." The concerns were that this was an added requirement imposed only on the Department and that there was no definition of what a "comparable" DOE facility would consist of.

Response: For all licensees physical security plans are designed for two purposes: (1) To protect against sabotage and (2) to promote the common defense and security. The change in the requirements of the physical security plan is intended to be consistent with 10 CFR Part 60, "Disposal of High-Level Radioactive Wastes in Geologic Repositories," wherein it is recognized that the Department already carries these responsibilities for all of its facilities.

The Department in carrying out its responsibility to promote the common defense and security of all its facilities can best identify the surface DOE facilities to which the MRS is most comparable for purposes of physical security without the unnecessary burden of an NRC definition of "Comparable." Comparability in this context is a function of the kinds and quantities of nuclear materials held at the facilities and the potential consequences of theft or sabotage. However, the NRC staff believes that the Receiving Basin for Off-Site Fuel at the Savannah River Plant may be an appropriately comparable facility.

9. Continous Cask Monitoring Provision

Comment: Several commenters pointed out that the wording of the provision in § 72.122(h)(4) (§ 72.92(h)(4)) for monitoring of storage confinement systems was inconsistent with section 141(b)(1)(B) of the NWPA (96 Stat. 2242, 42 U.S.C. 10161(b)(1)(B)) wherein it is required that an MRS facility shall be designed to permit continous monitoring. Another commenter suggested that the State should participate in the monitoring.

Response: The difference in wording between section 141(b)(1)(B) of the NWPA (96 Stat. 2242, 42 U.S.C. 10161(b)(1)(B)) and § 72.122(h)(4) (§ 72.92(h)(4)) was inadvertent. The staff has corrected the wording of § 72.122(h)(4) (§ 72.92(h)(4)) in the final rule to agree with the NWPA. As to State participation in monitoring, this is a matter to be resolved with the Department or as indicated in Response Number 3.

10. Inspection and/or Monitoring

Comment: In § 72.44(c)(3) (§ 72.33(c)(3)) the words "inspection and monitoring" have been changed to "inspection or monitoring."

Response: The proposed change serves no useful purpose. The degree and method of inspection and monitoring will be dependent upon design and operational limits for specific cases. The words "inspection and monitoring" will be reinstated.

11. Foreign Fuel

Comment: One commenter expressed objection to the processing and storage of foreign spent fuel or HLW at the MRS and stated that it should be specifically prohibited.

Response: The reference to foreign fuel in § 72.78 (§ 72.54) of the proposed rule was limited to material transfer report requirements and was not intended either to restrict or to permit such processing or storage. Section 302(a) of the NWPA (96 Stat. 2257, 42 U.S.C. 10222(a)) does specify only "high-level radioactive waste, or spent nuclear fuel of domestic origin" and therefore the reference to foreign fuel at an MRS will be removed.

12. Tornado Missile

Comment: Commenters have disagreed with the deletion of the exemption regarding protection against tornado missile impact, that is, as expressed in the existing rule, "* * * An ISFSI need not be protected from tornado missiles * * *". Another commenter who favors the deletion concerning protection from tornado missiles would also have the restriction limiting its scope to "* * * structures, systems, and components important to safety" deleted.

Response: The explanation of the exemption for tornado missiles, set out in the preamble of the existing rule (45 FR 74693, November 12, 1980) states that radionuclide releases from spent fuel which has undergone at least a year of radioactive decay would not be significant in the event of tornado missile impact, citing an accident evaluation from NUREG-0575 "Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuels" with gaseous radionuclide releases from water pool storage. With the continuing development of dry storage technologies, which include metal casks, concrete silos, dry wells, and air-cooled vaults, the Commission decided the designs should take into account tornado missile protection, unless it is shown that tornado missiles will not have any effect on structures, systems and components important to safety. While offsite gaseous release impacts from fuel rod rupture due to a tornado missile incident would remain insignificant, it is important to assure that design criteria for dry storage designs continue to address maintaining confinement of particulate material. All safety reviews for storage licensed under Part 72, both water pool and dry storage, have evaluated designs with respect to tornado missile impact. Since safety considerations drive the concern with respect to the tornado missile phenomenon, it is not necessary to expand that concern beyond "structures, systems, and components important to safety."

13. Use of Part 50 Criteria

Comment: To expedite the licensing process for facilities proposed on sites which currently possess a 10 CFR Part 50 license, it was proposed that the applicable siting evaluation factors and general design criteria which have been reviewed and approved by the NRC for the Part 50 license be directly adopted for the Part 72 facility without additional review, hearings or approvals. Adequate reviews and approvals have been completed, and any change to those previously approved should be treated as a backfit.

Response: The storage of an increased amount of spent fuel on a reactor site, over that covered under an existing Part 50 license, requires staff action through safety and environmental reviews. In taking this action to authorize additional storage capacity for spent fuel, the staff will apply criteria from Part 50 or Part 72, depending on the type of licensing action being sought. Licensing action for an ISFSI would use criteria contained in Part 72 and Part 50 would be used for amending an existing reactor license. Storage of spent fuel on a reactor site outside of an existing reactor basin is already regulated under the criteria of Part 72 and these criteria have been used in reviewing applications for additional fuel storage at reactor sites.

14. Cladding

Comment: Opposition is expressed to any lowering of fuel cladding protection, as provided for in the existing § 72.122(h)(1) (§ 72.92(h)(1)).

Response: The revision of this provision (i.e., § 72.122(h)(1) (§ 72.92(h)(1))) addressed confinement of fuel material, which is the purpose of protecting the fuel cladding. The revised provision specifically provides for additional alternative means of accomplishing this objective. This serves to enhance confinement protection capability rather than diminish it.

15. Rod Consolidation

Comment: Comments were received concerning the Department of Energy's plan to consolidate rods from spent fuel assemblies into sealed packages. One commenter suggested inserting the word "chemically" after the word "separated" in the definition of spent nuclear fuel. Another comment suggested that a separate environmental impact statement be prepared on rod consolidation. It was suggested that the NRC give rod consolidation special consideration and that it is not clear at present what requirements the NRC will use for rod consolidation.

Response: Rod consolidation is the most elaborate operation contemplated for the MRS. The Department of Energy in its proposal and elsewhere has indicated its intention to fully develop the rod consolidation process for installation and operation. The rod consolidation system must meet all applicable portions of the general design criteria. There is no precedent for the preparation of an environmental impact statement in connection with a single system of a facility for which a complete environmental impact statement will be prepared. The aspect of rod consolidation will be covered in that statement, as well as in the safety review and evaluation by the staff in connection with the application for an MRS. The NRC does expect to be kept informed by the Department of its developmental activities prior to receipt of an application.

The insertion of the word "chemically" as suggested has been accepted by the staff for the final rule.

16. Accident Analysis For Two Barriers

Comment: A comment was received regarding engineered barriers such as canisters. "* * * the design basis accident scenario (i.e., release of gap activity from all fuel contained in a dry cask) should be revised to account for cases in which canister or other engineered barriers are incorporated."

Response: Most cask designs do not incorporate canistering of spent fuel assemblies. Therefore, for purposes of this rulemaking, choice of a lesser accident scenario assuming canistering is not appropriate for a bounding analysis. In a safety review involving a specific design, which incorporates an additional engineered barrier, the design basis accident scenario should, of course, consider this addition in the review analysis.

17. Records

Comment: Comments were received concerning archiving of records: by whom and how long?

Response: The proposed rule is consistent with current NRC policy concerning retention periods for records. The specific details of their physical storage is action taken at time of licensing.

18. Operator Safety

Comment: Comments were received concerning design for ALARA.

Response: The licensee is responsible for meeting the requirements of 10 CFR Part 20 "Standards for Protection Against Radiation," and all its provisions for maintaining ALARA. In addition § 72.24 (§ 72.15) Contents of Application: Technical Information requires applicants for a license to supply information for maintaining ALARA for occupational exposure.

19. MRS Collocation with Waste Repository

Comment: Commenter suggested expanding limitation for collocation with repository to include other facilities.

Response: The collocation restrictions in § 72.96 (§ 72.75) are specifically included in order to comply with sections 141(g) and 145(g) of the NWPA (96 Stat. 2243, 42 U.S.C. 10161(g); 101 Stat. 1330–235, 42 U.S.C. 10165(g)). (See also section 135(a)(2), 96 Stat. 2232, 42 U.S.C. 10155(a)(2).)

20. MRS Collocation with Other Nuclear Facilities

Comment: Commenter was concerned about other nuclear facilities that are not licensed.

Response: The licensing process considers all activities and facilities, licensed or unlicensed, that could increase the probability or consequences of safety significant events at licensed facilities.

21. Definition of High-Level Radioactive Waste

Comment: Some commenters noted that the definition of "high-level radioactive waste" used in Part 72 was not the same as the definition used in 10 CFR Part 60 and expressed the view that the two definitions should be consistent.

Response: Since it was first promulgated in November 1980 for the purpose of establishing licensing requirements for the storage of spent fuel in an independent spent fuel storage installation, Part 72, unlike Part 60, has always contained a separate definition of spent fuel. In revising Part 72 to provide for licensing the storage of spent fuel and high-level radioactive waste in an MRS, the Commission has revised the definition of spent fuel to conform more closely to the definition set out in section 2(23) of the Nuclear Waste Policy Act of 1982, as amended (96 Stat. 2204, 42 U.S.C. 10101(23)). The Commission has also amended § 72.3 by adding a definition of "high-level radioactive waste" which conforms to the language used in section 2(12) of that Act (42 U.S.C. 10101(12)). The definitions of spent fuel and high-level radioactive waste used in Part 72, though not identical to the definition of high-level radioactive waste used in 10 CFR Part 60 which encompasses "irradiated reactor fuel," are not inconsistent with that definition. It should be noted, however, that as explained in the Commission's advance notice of proposed rulemaking relating to the definition of high-level radioactive waste (52 FR 5992, February 27, 1987), the definition of high-level radioactive waste used in Part 60 serves a jurisdictional function, specifically identification of the class of Department of Energy facilities that, under section 202 of the Energy Reorganization Act of 1974 (42 U.S.C. 5842) are subject to the licensing and related regulatory authority of the Commission.

22. High Level Liquid Waste

Comment: Several commenters were concerned about the storage of liquid High-Level Waste (HLW).

Response: The MRS will be designed and licensed for the storage of irradiated fuel and solidified waste from the processing of fuel. The MRS will not receive liquid HLW and the form of the solid waste stored will be that which is compatible with the requirements for permanent disposal in a repository.

Any liquid wastes generated at the MRS will be handled in accordance with existing regulations.

23. Quality Assurance—Quality Control

Comment: Comments were associated with the apparent difference between the quality assurance criteria proposed and the previous quality assurance criteria.

Response: The proposed rule quality assurance subpart was written to incorporate the previously referenced 10 CFR Part 50, Appendix B quality assurance criteria specifically into Part 72. There was no intent to change the criteria. Minor conforming changes have been made in the final rule.

24. Criticality

Comment: A comment was received concerning the removal of the requirement for verifying continued efficacy of solid neutron poisons.

Response: Several changes have been made to the criticality section of the final rule to make it correspond to other Parts of the Commission's regulations and standard criticality review practices. Verification of solid neutron poisons has been retained. Double contingency criteria and requirements for criticality monitors have been added. It is not the intent of the revision concerning criticality monitors to require monitors in the open areas where loaded casks are positioned for storage as that system is static. Monitors are required where the systems are dynamic.

25. MRS Storage Capacity

Comment: Commenters questioned the MRS storage capacity as stated in the proposed rule in §§ 72.1 and 72.96 (§§ 72.1 and 72.75).

Response: In the proposed rule, MRS storage capacity values are based on the NWPA, as approved by Congress. (See section 135(a)(1)(A), 96 Stat. 2232, 42 U.S.C. 10155(a)(1)(A) and section 114(d), 96 Stat. 2215 as amended by 101 Stat. 1330–230, 42 U.S.C. 10134(d) and section 141(g), 96 Stat. 2243, 42 U.S.C. 10161(g)). In addition, the Nuclear Waste Policy Amendments Act of 1987 provides that the MRS authorized by section 142(b) of NWPA (101 Stat. 1330–232, 42 U.S.C. 10162(b)) shall be subject to the storage capacity limits specified in sections 148(d) (3) and (4) (101 Stat. 1330–236, 42 U.S.C. 10168(d) (3) and (4)). These requirements have been incorporated in new § 72.44(g) which has been added to the final rule.

26. The Term—"Temporary Storage"

Comment. Comments objected to the removal of the term "Temporary Storage" from § 72.3 Definitions and the removal of the word "temporary" from § 72.2 Scope.

Response: In making these changes, the Commission does not intend to change the scope of Part 72 which relates to the licensing of ISFSI and MRS for the purpose of storage only. Part 72 does not nor is it intended to cover permanent disposal. Accordingly, use of the word "temporary" in the rule is non-definitive and unnecessary.

27. MRS Rule Making

Comment: Many commenters (approximately 150), through the use of form letters or paraphrasing, did not want the MRS in Tennessee, did not support any form of rulemaking until Congress had authorized the MRS through funding appropriation, and made reference to "license it twice."

Response: The Nuclear Waste Policy Amendments Act of 1987 authorizes the Department of Energy to site, construct and operate one MRS and prescribes procedures for the selection of an appropriate site. The Act expressly annuls and revokes the Department's proposal "to locate a monitored retrievable storage facility at a site on the Clinch River in the Roane County portion of Oak Ridge, Tennessee, with alternative sites on the Oak Ridge Reservation of the Department of Energy and on the former site of a proposed nuclear powerplant in Hartsville, Tennessee * * *" (Section 142(a), 101 Stat. 1330–232, 42 U.S.C. 10162(a)). The Commission's regulations are promulgated to permit the Commission to carry out its mandate of providing for the health and safety of the public. Except for the siting limitations in § 72.96 (§ 72.75) of the final rule, which, among other things, prohibits an MRS authorized by section 142(b) of NWPA (101 Stat. 1330–232, 42 U.S.C. 10162(b)) from being constructed in Nevada, the Commission's regulations are silent on the location of an MRS. The "license it twice" concept is addressed in Response Number 2.

28. Increase of Licensing Period for the MRS

Comment: Comments questioned the Commission's basis, as described in the statement of considerations for the proposed changes to Part 72, for providing a longer license term for an MRS (40 years) than for an ISFSI (20 years). Comments also included (1) the term should start with the receipt of spent fuel, and (2) ISFSI should also have a 40-year license term. Further explanation of the basis for the license term was also requested. All of the commenters seemed to concentrate on a license for the spent fuel rather than a license covering a facility for storage.

Response: An MRS as described in the NWPA is intended for storage, but not necessarily for the same fuel since fuel will continually be moved in and out over the life of the facility in concert with operation of a repository. A longer license term is therefore appropriate for an MRS considering the purpose and mode of operation of the facility.

In contrast to the MRS, the spent fuel stored in an ISFSI at reactor sites or elsewhere will be collected until the Department of Energy waste disposal system is ready for its receipt. The current schedule indicates that this transfer from reactor sites to an MRS could begin to occur within about 10 years. The Commission has in place a license renewal process for ISFSI storage which provides an opportunity for extension of the 20-year license term, with staff reevaluation of safety and environmental aspects of the operation. In any event the systematic inspection program of the Commission wherein the licensee's adherence to all license conditions and technical specifications is continually being examined applies to both MRS and ISFSI storage over the entire period of a license. The Commission will provide a 40-year license term for an MRS in the final rule.

On December 22, 1987, the Nuclear Waste Policy Amendments Act of 1987 (Subtitle A of Title V of the Omnibus Budget Reconciliation Act for Fiscal Year 1988; Pub. L. 100–203, 101 Stat. 1330–227) was approved by the President and became public law. The 1987 amendments authorized the Secretary of the Department of Energy to site, construct and operate one monitored retrievable storage facility subject to certain statutory conditions (sec. 142(b), 101 Stat. 1330–232, 42 U.S.C. 10162(b)). As a result of these changes in the statute, it has been necessary to make certain conforming changes in the text of the final rule. Most of the changes are minor in nature. For example, references have been added to the authority section and conforming changes have been made in the following sections of the rule: §§ 72.22(d)(5), 72.40(b), 72.90(e) and 72.96(d) (§§ 72.14(d)(5), 72.31(h), 72.70(e) and 72.75(d)). A new paragraph (g) has been added to § 72.44 (§ 72.33), License conditions, to incorporate into the Commission's regulations the specific statutory conditions (*see* sec. 148(d) of the NWPA, 101 Stat. 1330–238, 42 U.S.C. 10168(d)) which must be included in a Commission license for the monitored retrievable storage installation authorized pursuant to section 142(b) of the NWPA (101 Stat. 1330–232, 42 U.S.C. 10162(b)). For an explanation of these conditions, *see* 133 Cong. Rec. H11973–75 and S18683–84 (daily ed. December 21, 1987).

Having considered all of the above, the Commission has determined that a final rule be promulgated. The text of the final rule has some changes as noted from the proposed rule.

Finding of No Significant Environmental Impact

The Commission has determined not to prepare an environmental impact statement for the proposed amendments to 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."

NUREG–0575, "Final Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel," August 1979, was issued in support of the final rule promulgating 10 CFR Part 72, "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation ISFSI)," which became effective November 28, 1980. On January 7, 1983, the Nuclear Waste Policy Act of 1982 was signed into law. On December 22, 1987, the Act was amended by the Nuclear Waste Policy Amendments Act of 1987 (Pub. L. 100–203, Title V, Subtitle A, 101 Stat. 1330–227). Section 142(b) of the amended Act (101 Stat. 1330–232, 42 U.S.C. 10162(b)) authorized the Secretary of the Department of Energy to site, construct and operate one MRS. NWPA also established procedures which a State or an Indian tribe may use to negotiate an agreement with the Federal Government under which the State or Indian tribe would agree to host an MRS within the State or reservation. Following enactment of legislation to implement the negotiated agreement, the Secretary of the Department of Energy could proceed to evaluate appropriate sites. As in the case of the MRS authorized by section 142(b) of NWPA (101 Stat. 1330–232, 42 U.S.C. 10162(b)), DOE must also obtain an NRC license for an MRS authorized by Congress pursuant to a negotiated agreement. The NRC staff has concluded that although existing 10 CFR Part 72 is generally applicable to the design, construction, operation, and decommissioning of MRS, additions are necessary to explicitly cover the licensing of spent nuclear fuel and high-level radioactive waste storage in an MRS. In August 1984, the NRC published an environmental assessment for this proposed revision of Part 72. NUREG-1092. "Environmental Assessment for 10 CFR Part 72. Licensing Requirements for the Independent Storage of Spent Fuel and High-Level Radioactive Waste." NUREG-1092 discusses the major issues of the rule and the potential impact on the environment. The findings of the environmental assessment are "(1) past experience with water pool storage of spent fuel establishes the technology for long-term storage of spent fuel without affecting the health and safety of the public, (2) the proposed rulemaking to include the criteria of 10 CFR Part 72 for storing spent nuclear fuel and high-level radioactive waste does not significantly affect the environment, (3) solid high-level waste is comparable to spent fuel in its heat generation and in its radioactive material content on a per metric ton basis, and (4) knowledge of material degradation mechanisms under dry storage conditions and the ability to institute repairs in a reasonable manner without endangering the health [and safety] of the public shows dry storage technology options do not significantly impact the environment." The assessment concludes that, among other things, there are no significant environmental impacts as a result of promulgation of these revisions of 10 CFR Part 72.

Based on the above assessment the Commission concludes that the rulemaking action will not have a significant incremental environmental impact on the quality of the human environment.

Paperwork Reduction Act Statement

This final rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 *et seq.*). These requirements were approved by the Office of Management and Budget approval number 3150-0132.

Regulatory Analysis

The NRC has prepared a regulatory analysis on this final rule. The analysis examines the benefits and alternatives considered by the NRC. The analysis is available for inspection in the NRC Public Document Room, 1717 H Street NW., Washington, DC. Single copies of the analysis may be obtained from C.W. Nilsen, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555 (301-492-3834).

Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act of 1980 (5 U.S.C. 605(b)), the Commission certifies that this rule will not have a significant economic impact on a substantial number of small entities. This final rule affects only the licensing and operation of independent spent fuel storage installations and of monitored retrievable storage installations. The owners of these installations, nuclear power plant utilities or DOE, do not fall within the scope of the definition of "small entities" set forth in section 601(3) of the Regulatory Flexibility Act or within the definition of "small business" in section 3 of the Small Business Act, 15 U.S.C. 632, or within the Small Business Size Standards set out in regulations issued by the Small Business Administration at 13 CFR Part 121.

List of Subjects

10 CFR Part 2

Administrative practice and procedure, Antitrust, Byproduct material, Classified information, Environmental protection, Nuclear materials, Nuclear power plants and reactors, Penalty, Sex discrimination, Source material, Special nuclear material, Waste treatment and disposal.

10 CFR Part 19

Environmental protection, Nuclear materials, Nuclear power plants and reactors, Occupational safety and health, Penalty, Radiation protection, Reporting and recordkeeping requirements, Sex discrimination.

10 CFR Part 20

Byproduct material, Licensed material, Nuclear materials, Nuclear power plants and reactors, Occupational safety and health, Packaging and containers, Penalty, Radiation protection, Reporting and recordkeeping requirements, Special nuclear material, Source material, Waste treatment and disposal.

10 CFR Part 21

Nuclear power plants and reactors, Penalty, Radiation protection, Reporting and recordkeeping requirements.

10 CFR Part 51

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

10 CFR Part 70

Hazardous materials—transportation, Material control and accounting, Nuclear materials, Packaging and containers, Penalty, Radiation protection, Reporting and recordkeeping requirements, Scientific equipment, Security measures, Special nuclear material.

10 CFR Part 72

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

10 CFR Part 73

Hazardous materials—transportation, Incorporation by reference, Nuclear materials, Nuclear power plants and reactors, Penalty, Reporting and recordkeeping requirements, Security measures.

10 CFR Part 75

Intergovernmental relations, Nuclear materials, Nuclear power plants and reactors, Penalty, Reporting and recordkeeping requirements, Security measures.

10 CFR Part 150

Hazardous materials—transportation, Intergovernmental relations, Nuclear materials, Penalty, Reporting and recordkeeping requirements, Security measures, Source material, Special nuclear material.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, 5 U.S.C. 552 and 553, and the Nuclear Waste Policy Act of 1982, as amended, the NRC is adopting the following revision to 10 CFR Part 72 and related conforming amendments to 10 CFR Parts 2, 19, 20, 21, 51, 70, 73, 75, and 150

53 FR 43419
Published 10/27/88
Effective 10/27/88

Relocation of NRC's Public Document Room; Other Minor Nomenclature Changes

See Part 1 Statements of Consideration

> **55 FR 10397**
Published 3/21/90.
Effective 4/20/90

Preserving the Free Flow of Information to the Commission

See Part 30 Statements of Consideration

55 FR 13883
Published 4/12/89.

10 CFR Part 72

RIN 3150-AD21

Preserving the Free Flow of Information to the Commission

Correction

In rule document 90-6424 beginning on page 10397 in the issue of Wednesday, March 21, 1990, make the following correction:

§ 72.10 Employee protection.

On page 10405, in the first column, the section heading following amendatory instruction "14" should read as set forth above.

55 FR 29181
Published 7/18/90
Effective 8/17/90

10 CFR Parts 50, 72, and 170

RIN 3150-AC76

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

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is amending its regulations to provide for the storage of spent nuclear fuel under a general license on the site of any nuclear power reactor provided the reactor licensee notifies the NRC, only NRC-certified casks are used for storage, and the spent fuel is stored under conditions specified in the cask's certificate of compliance. This final rule also provides procedures and criteria for obtaining NRC approval of spent fuel storage cask designs.

EFFECTIVE DATE: August 17, 1990.

FOR FURTHER INFORMATION CONTACT: John L. Telford, Office of Nuclear Regulatory Research (Telephone: (301) 492-3796) or John P. Roberts (Telephone: (301) 492-0608), Office of Nuclear Material Safety and Safeguards, Nuclear Regulatory Commission, Washington, DC 20555.

SUPPLEMENTARY INFORMATION:

Background

The Commission published the proposed rule on this : ubject in the Federal Register on May 5, 1989 (54 FR 19379). The rule proposed to amend 10 CFR part 72 to provide for storage of spent fuel on the sites of nuclear power reactors without the need for additional site-specific Commission approvals, as directed by the Nuclear Waste Policy Act of 1982 (NWPA). Section 218(a) of the NWPA directed the Department of Energy to establish a spent fuel storage development program with the objective of establishing one or more technologies that the NRC might approve for use at civilian nuclear power reactor sites without, to the maximum extent practicable, the need for additional site-specific approvals by the Commission. Section 133 of the NWPA directs the Commission to establish, by rule, procedures for licensing any technology approved under Section 218(a). The approved technology is storage of spent fuel in dry casks. The final rule is not significantly different from the proposed rule. In order to utilize an NRC certified cask under a general license, power reactor licensees must (1) perform written evaluations showing that there is no unreviewed safety question or change in reactor technical specifications related to the spent fuel storage, and that spent fuel will be stored in compliance with the cask's Certificate of Compliance, (2) provide adequate safeguards; (3) notify NRC prior to first storage of spent fuel and whenever a new cask is added to storage; and (4) maintain the records specified in the rule.

Public Responses

The comment period expired on June 19, 1989, but all of the comments received were considered in this final rulemaking. The NRC received 273 comment letters from individuals, environmental groups, utilities, utility representatives, engineering groups, States, and a Federal agency. Among the comment letters were 237 from individuals, including several signed by more than one person. Many commenters discussed topics that were not the subject of this rulemaking, e.g., that the generation of radioactive wastes should be stopped and that environmentally safe alternative sources of power should be developed.

The Western Governors' Association recently passed a resolution expressing their position on the storage of spent commercial power reactor fuel. In this resoultion the governors endorsed at-reactor dry storage of spent fuel as an interim solution until a permanent repository is available. This resolution was forwarded to NRC Chairman Kenneth M. Carr in a memorandum dated December 5, 1989.

Included in the comments received was a "petition" addressed to the Commission, which was signed by 188 people, who are opposed to the proposed rule and who specifically oppose:

1. Storage at the Pilgrim nuclear power plant of spent fuel generated at other reactors,
2. Storage of spent fuel in casks outside the reactor building,
3. Storage of spent fuel without the need for specific approval of the storage site, and
4. Storage of spent fuel without requiring any specific safeguards to prevent its theft

Many of the letters contained comments that were similar in nature. These comments are grouped, as appropriate, and addressed as single issues. The NRC has identified and responded to 50 separate issues that include the significant points raised. Among the comments that discussed technology, the majority expressed a preference for spent fuel storage in dry casks over wet storage.

On August 19, 1988, the Commission promulgated a final rule revising 10 CFR part 72 (53 FR 31651), which became effective on September 19, 1988. Among the changes made in that final rule was a renumbering of the sections. These revised section numbers are the ones referenced in this rulemaking. Because many people interested in this rulemaking may not have a copy of the newly revised part 72, sections referenced in this Supplementary Information section are followed by a bracketed number that refers to the corresponding section number in the old rule (43 FR 74693, made effective on November 12, 1980).

Analyses of Public Comments

1. *Comments.* Elimination of public input from licensing of spent fuel storage at reactors under the general license was discussed in 237 letters of comment and 52 of the commenters were opposed to the rule for this reason. Many of these comments were opposed to the NRC allowing dry cask storage without going through the formal procedure currently required for a facility license amendment that requires public notification and opportunity for a hearing. One commenter stated that the proposed rule does not guarantee hearing rights mandated by the Atomic Energy Act, and, therefore, the proposed rule must be amended to provide for site-specific hearing rights before it can be lawfully adopted. Another commenter stated that, by proposing to issue a general license before determining whether license modifications are required in order to allow the actual storage of spent fuel onsite, the NRC apparently intends to circumvent the requirement for public hearings on individual applications for permission to use dry cask storage. This comment continued that this approach would violate the statutory scheme for licensing nuclear power plants, in which the NRC must approve all proposed license conditions before the license is issued. This comment further stated that the NRC cannot lawfully issue a general license for actual onsite storage of the waste without also obtaining and reviewing the site-specific information that would allow it to find that the proposed modification to each plant's design and operation are in conformance with the Atomic Energy Act (the Act) and the regulations.

Response. This rule does not violate any hearing rights granted by the Act. Under 10 CFR parts 2, 50, and 72, interested persons have a right to request a formal hearing or proceeding for the granting of a license for a power reactor or the granting of a specific license to possess power reactor spent fuel in an independent spent fuel storage installation (ISFSI) or a monitored retrievable storage installation (MRS). However, hearing processes do not apply when issues are resolved generically by rulemaking. Under this rule, casks will be approved by rulemaking and any safety issues that are connected with the casks are properly addressed in that rulemaking rather than in a hearing procedure.

There is a possibility that the use of a certified cask at a particular site may entail the need for site-specific licensing action. For example, an evaluation under 10 CFR 50.59 for a new cask loading procedure could require a part 50 license amendment in a particular case. In this event the usual formal hearing requirements would apply. However, generic cask approval (issuance of a certificate of compliance) would, in accordance with section 133 of the Nuclear Waste Policy Act of 1982 (NWPA), eliminate the need for site-specific approvals to the maximum extent practicable.

Under the rule, actual use of an NRC certified cask will require reviews by individual facility licensees to show, among other things, that conditions of the certificate of compliance for the cask will be met. These reviews and necessary follow-up actions by the licensee are conditions for use of the cask. For example, licensees must review their reactor security plan to ensure that its effectiveness is not decreased by the use of the casks. But these requirements for license reviews do not constitute requirements for Commission approval prior to cask use: that is no Commission finding with respect to these reviews are needed prior to use of the casks. Therefore, no hearing rights will accrue to these reviews unless, of course, the reviews point to the need for an amendment of the facility license. The Commission is satisfied that public health and safety, the common defense and security, and protection of the environment is reasonably assured without the requirement for Commission approval of these license reviews because conservative requirements apply, such as a safety analysis of cask designs, including design bases, design criteria, and margins of safety; an evaluation of siting factors, including earthquake intensity and tornado missiles; an application of quality assurance, including control of cask design and cask fabrication; and physical protection. These conservative requirements and stringent controls assure safe cask storage for any reactor site.

2. *Comments.* The NRC apparently intends to exercise no systematic or mandatory review of applications to store fuel in dry casks, despite the numerous changes involved in the reactor's design and procedures. This commenter further stated that the rule should provide for mandatory submission and review by the NRC of technical documents required in § 72.212 and that these documents should be placed in the public document rooms for inspection by the public.

Response. A condition of the general license is that a reactor licensee must determine whether activities related to storage of spent fuel at the reactor site involve any unreviewed safety question or require any change in technical specifications. This written determination becomes part of the reactor licensee's records. Under 10 CFR 50.59, an unreviewed safety question is involved if (1) the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR may be increased; or (2) if a possibility for an accident or malfunction of a different type than any evaluated previously in the SAR may be created; or (3) if the margin of safety as defined in the basis for any technical specification is reduced. If the evaluation made under 10 CFR 50.59 reveals any unreviewed safety question or if use of a cask design requires any change in technical specifications or a facility license amendment is needed for any reason, then casks of that design cannot be used to store spent fuel under the general license. The reactor licensee must apply for and obtain specific NRC approval of those changes to the facility license necessary to use the desired cask design, use a different cask design, or apply for a specific license under 10 CFR part 72. If the reactor licensee chooses to make changes to accommodate the desired cask design, e.g., revise technical specifications, an application for a license amendment would have to be submitted under 10 CFR 50.90.

3. *Comments.* It appears that a hearing would be mandated under the Act, as spent fuel storage under the general license would involve a license amendment. The commenter argued that nuclear power reactor licenses contain a clause stating that the facility has been constructed and will operate in accordance with the application and that the application will operate in accordance with the application and that the application includes the FSAR (10 CFR 50.34(b)). If the FSAR does not describe cask storage of spent fuel, then a facility using cask storage would not be operating in accordance with the application and the license, necessitating a license amendment.

Response. According to 10 CFR 50.34(b) each application for a license to operate a power reactor must include an FSAR. The FSAR must include information that describes the facility, presents the design bases and limits on its operation, and presents a safety analysis of the structures, systems, and components of the reactor. A power reactor is licensed to operate under the regulations in 10 CFR part 50. If spent fuel is stored in an ISFSI on a reactor site, this storage will be licensed under the regulations in 10 CFR part 72. The ISFSI may share utilities and services with the reactor for activities related to the storage of spent fuel, e.g., facilities for loading spent fuel storage casks. A power reactor FSAR will contain a description of cask loading and unloading, because reactor fuel (both fresh and spent) must be handled for operation of the reactor. If no amendment of the operating license is necessary (e.g., there is no problem in fuel handling concerning heavy loads and there is no unreviewed safety question), then spent fuel may be stored under the general license. The authority for storage of spent fuel in the certified cask would be derived from the general license, not from the part 50 license.

4. *Comments.* The NRC should reconsider the indiscriminate storage on a reactor site of spent nuclear fuel that was generated at other reactor sites. One commenter stated that there should be a restriction to permit only transfer of spent fuel from plant to plant within a utility-owned group of plants. Another commenter stated that storage of spent fuel from two or more reactors inevitably makes the host site a de facto regional repository, without the same benefit of review and discussion given the regional site. Another commenter suggested that the amount of spent fuel stored on a site should be limited to that amount produced by the site's reactor operations. The major concern of these commenters appeared to be that spent fuel from a number of reactors would be deliberately accumulated and stored at one reactor site under this general license.

Response. This rulemaking is not concerned with transfer or shipment of spent fuel from one reactor site to another. As explained in the discussion of the proposed rule (54 FR 19379), transfer of spent fuel from one reactor site to another must be authorized by the receiving reactor's operating license. Such authorization usually will require a license amendment action conducted under the regulations in 10 CFR part 50. The transportation of the spent fuel is subject to the regulations in 10 CFR part 71. This rulemaking is not germane to either spent fuel transfer or transportation procedures. The NRC anticipates that, beginning in the early 1990s, there will be a significant need for additional spent fuel storage capacity at many nuclear power reactors. This was a major reason for initiating this rulemaking at this time. Dry storage of spent fuel in casks under a general license would alleviate the necessity of transferring spent fuel from one reactor site to another.

5. *Comment.* The Commission should reconsider a petition for rulemaking submitted by the State of Wisconsin. The petition requested that the NRC expand the scope of its regulations pertaining to spent fuel transport "to ensure that both the need for and the safety and environmental consequences of proposed shipments have been considered in a public forum prior to approval of the shipment and route."

Response. As explained in the response to comment number 4, this rulemaking does not apply to transportation of spent fuel. Transportation of spent fuel is the subject of 10 CFR part 71, under which the issues raised by this petition were considered. There is no reason to reconsider this petition in terms of the issues under consideration in this rulemaking.

6. *Comment.* How would the rulemaking process for cask approvals be implemented?

Response. The initial step would be taken by a cask vendor submitting an application for NRC approval of a cask design. The NRC would review the cask safety analysis report (SAR) and other relevant documents. If the cask design is approved, the NRC would initiate a rulemaking to amend 10 CFR 72.214 to add certification of the cask design. The NRC would also revise the NUREG containing the Certificates of Compliance for all approved storage casks to add the new cask's Certificate of Compliance.

7. *Comment.* The proposed 10 CFR 72.236(c) would establish a criterion that casks must be designed and fabricated so that subcriticality is maintained. This seems to suggest that the actual fabrication takes place before cask approval. Otherwise how could NRC find that the cask has been fabricated to maintain subcriticality?

Response. Findings by the NRC concerning safety of cask design are based on analyses presented in the cask SAR. In the case of criticality analyses, the SAR must include a description of the calculational methods and input values used to determine nuclear criticality, including margins of safety and benchmarks, justification and validation of calculational methods, fuel loading, enrichment of the unirradiated fuel, burnup, cooling time of the spent fuel prior to cask storage, and neutron cross-sectional values used in the analysis. Further, in order to obtain approval of a cask design, the vendor must demonstrate that casks will be designed and fabricated under a quality assurance program approved by the NRC. As an example, if neutron poison material were part of the cask design to prevent inadvertent criticality, the quality assurance program would have to ensure that the material was actually installed as designed. The NRC will not inspect fabrication of each cask, but will ensure that each cask is fabricated under an NRC-approved quality assurance program. Thus, there is reasonable assurance that the cask will be designed and fabricated to maintain spent fuel in a subcritical configuration in storage.

8. *Comment.* Each utility should be required to present a plan for inspecting the casks in the storage area.

Response. Surveillance requirements for spent fuel storage casks in the storage area are required and are described in the cask's Certificate of Compliance. Also, periodic inspections for safety status and periodic radiation surveys are required by the certificate. Further, licensees will have to keep records showing the results of these inspections and surveys.

9. *Comments.* The 20-year limit on approval of cask designs seems unduly restrictive and was not supported by any discussion of safety or environmental issues in the preamble of the proposed rule. One comment stated that unless there are overriding institutional issues or a defect in a cask model, which would preclude providing adequate protection of the environment or public health and safety, there would be no need to revoke or modify a Certificate of Compliance. Three commenters suggested that the criteria for cask design reapproval should be limited to safety and environmental issues related to the storage period, because there may have been proprietary information involved in the initial approval that might not be available for reapproval. Another commenter stated that the licensing period for spent fuel storage casks should be extended to be at least equal to the operating license of the reactor. Another commenter stated that because a 100-year period is being considered by the Commission in its waste confidence review, an extension should be considered for a cask certification period.

Response. The procedure for reapproval of cask designs was not intended to repeat all of the analyses required for the original approval. However, the Commission believes that the staff should review spent fuel storage cask designs periodically to consider any new information, either generic to spent fuel storage or specific to cask designs, that may have arisen since issuance of the cask's Certificate of Compliance. A 20-year reapproval period for cask designs was chosen because it corresponds to the 20-year license renewal period currently under part 72.

10. *Comment.* It is conceivable that, after 20 years of storage, the regulations could force the transfer of spent fuel at the reactor to a new cask or a different cask design only because it better conforms to DOE's preference. If considerations such as safety risks and occupational exposure from spent fuel transfer are not a significant factor, this potential uncertainty should be removed from the rule.

Response. The Department of Energy (DOE) will be the ultimate receiver of spent fuel. If a cask design were not compatible with DOE's criteria for receipt of spent fuel, then measures would need to be taken so that spent fuel could be transferred offsite. What these measures might be would depend on the cask design and DOE's criteria.

11. *Comment.* The practice of permitting each vendor to not seek reapproval of the cask design after a 20-year period seems "fragile and irresponsible."

Response. This comment is interpreted to mean that the Commission should require each cask vendor to submit an application for reapproval of their cask design. The Commission's authority over corporate entities is limited to licensing matters and it cannot control the economic status of spent fuel storage cask manufacturers. The NRC cannot require that a cask vendor submit an application for renewal of a storage cask design if the vendor is no longer in business. A cask vendor who remains in the business of manufacturing spent fuel storage casks is required to submit an application for renewal of a cask design. Otherwise the cask's Certificate of Compliance would expire and that cask design could not be used to store spent fuel. Licensees cannot use any cask that does not have a valid Certificate of Compliance. If a cask vendor goes out of the business of supplying spent fuel storage casks, it would not invalidate NRC approval of the spent fuel storage casks that were manufactured by this vendor and remain in use. That is the reason the Commission will permit general licensees or their representatives to apply for cask design reapproval. Accordingly, the Commission will keep appropriate historical records and conduct inspections, as required, related to spent fuel storage in casks. Cask vendors are requested to notify the Commission if they do not intend to submit an application for reapproval of a cask design. Also, vendors are required under 10 CFR 72.234 to submit their composite record to the NRC of casks manufactured and sold or leased to reactor licensees if they permanently cease manufacture of casks under a Certificate of Compliance. In any case, the cask design renewal procedure will be coordinated through historical records, inspections, and communications with cask vendors.

12. *Comments.* The requirements in proposed § 72.234(c) that cask fabrication cannot start prior to receipt of the Certificate of Compliance is unnecessarily restrictive. The commenter indicated that a vendor should have the option of being able to

start fabrication (taking the risk of building a cask that may not ever be licensed) prior to NRC issuing the Certificate of Compliance.

Response. Section 72.234(c) is not intended to prevent vendors from taking a risk. The Certificate of Compliance provides the specific criteria for cask design and fabrication. If a vendor has not received the certificate, then the vendor does not have the necessary approved specifications and may design and fabricate casks to meet incorrect criteria.

13. *Comments.* Requiring a submittal for reapproval of cask design 3 years before the expiration date of a Certificate of Compliance seems excessive. Another commenter suggested that a procedure similar to that used for renewal of materials-type licenses could be used, which is that when a licensee submits an application for license renewal in proper form not less than 30 days prior to the expiration date of the license that the existing license does not expire until the application for renewal has been finally determined by the Commission.

Response. Current regulations in 10 CFR part 72 requires that applications for license renewal be submitted 2 years prior to the expiration date of the license. This was a major consideration for setting the date for submittal of a cask design reapproval application in the proposed rule. The NRC has reconsidered this requirement and believes that the period required for cask design reapproval can be reduced. The final rule has been revised to incorporate language similar to that for other materials-type license renewals, which would allow a Certificate of Compliance to continue in effect until the application for reapproval has been finally determined by the Commission.

14. *Comments.* No spent fuel dry storage should be allowed at sites that do not have fully operational State approved emergency preparedness plans. Another commenter stated that, for emergency response purposes and for proper inclusion in emergency planning, the utility must notify State and local governments simultaneously with the NRC when spent fuel storage is begun. Another commenter inquired whether or not States would be notified of spent fuel storage at the reactor site in order to minimize emergency response planning impacts.

Response. The new 10 CFR 72.32(c) [no section in the old rule is applicable] states that "For an ISFSI that is located on the site of a nuclear power reactor licensed for operation by the Commission, the emergency plan required by 10 CFR 50.47 shall be deemed to satisfy the requirements of this section." One condition of the general license is that the reactor licensee must review the reactor emergency plan and modify it as necessary to cover dry cask storage and related activities. If the emergency plan is in compliance with 10 CFR 50.47, then it is in compliance with the Commission's regulations with respect to dry cask storage. Thus, the utility does not need to separately notify State and local governments before beginning spent fuel storage.

15. *Comment.* What extra information, beyond that currently required in safety analysis reports, will be required in topical safety analysis reports for cask certification?

Response. Currently a Topical Safety Analysis Report (TSAR) is submitted to obtain spent fuel storage cask certification. NRC procedures allow applicants and licensees to reference appropriate Sections of a TSAR in licensing proceedings, which reduces investigative and evaluation costs for them. Under this final rule, applications and a Safety Analysis Report (SAR) (equivalent to a TSAR) will have to be submitted to cask design certification. There will not be any "extra" information required in an SAR as a result of this rulemaking. Guidance on the information to be submitted in an SAR for cask design certification is contained in Regulatory Guide 3.61, "Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask."

16. *Comment.* One comment stated that it is unclear from the proposed rule as to whether full-scale or scale model testing is required for cask certification.

Response. The safety of cask designs is analyzed in the SAR. The staff reviews cask design bases and criteria. The design and performance of the cask and the means of controlling and limiting occupational radiation exposures are analyzed. Appropriate functional and operating limits (technical specifications) are developed. However, in instances where cask design, construction, or operation can not be satisfactorily substantiated, the staff may require that some component or system testing be performed. During the first use of a certified design the licensee, in conjunction with the vendor, may be required to conduct preoperational testing on the first cask and submit a report to the NRC. This preoperational testing would assess the extent to which data supports the critical aspects of design, for example, the resultant cask temperature, pressure, and external radiation. Full-scale testing is not currently required for spent fuel dry storage cask design certification. However, testing of systems and components important to safety is required, and is specified in the Certificate of Compliance.

17. *Comment.* Can the NRC provide examples of acceptable means of demonstrating that a cask will reasonably maintain confinement of radioactive material under normal, off-normal, and accident conditions?

Response. Certification of a cask design is based on analyses described in each cask's SAR. These analyses must show how radioactive materials will be confined through evaluations of the cask's systems, structures, and components, and the designed markings of safety. These analyses are performed on an individual case basis considering each cask's design, materials of construction, cask sealing systems, fuel basket criticality considerations, and gamma and neutron shielding mechanisms. Thus, analyses are the acceptable means of demonstration.

18. *Comment.* The NRC should use this amendment to provide guidance or criteria on use of burnup credit in criticality analyses.

Response. Evaluations of burnup credit are dependent on parameters such as fuel design, exposure, and characteristics. These evaluations are best conducted on an individual case basis, because the variables that must be evaluated are closely related to the individual case history of the spent fuel. Thus, guidance on such evaluations would be more appropriately set forth in regulatory guides, rather than in regulations. To date allowance for burnup credit has not been accepted in reviews conducted under 10 CFR part 72, however, regulatory guides may be issued in the future.

19. *Comment.* What will a current reactor licensee have to do to obtain a general license?

Response. As specified in § 72.212(b), a power reactor licensee must (1) perform written evaluations establishing that spent fuel storage will be in compliance with a cask's Certificate of Compliance and that there is no unreviewed safety question or change in technical specifications involved in activities at the reactor related to the storage of spent fuel in casks, (2) provide adequate safeguards for the spent fuel in storage, (3) notify NRC prior to first-storage of spent fuel and whenever a new cask is used, and (4) keep records of spent fuel storage and related activities.

20. *Comment.* Could the general license be used to store spent fuel beyond the term of the reactor operating

license? Several utilities hold operating licenses at more than one site; thus, clarification is needed as to when an operating license is terminated and how licensees may use a general license.

Response. A licensee who holds reactor operating licenses at more than one site must notify NRC for each site involved. A licensee who holds operating licenses for more than one reactor located on a single site need notify NRC only once.

Spent fuel can be stored on a site only as long as there is a power reactor with a valid license or the possession of spent fuel is authorized under some other regulation or form of license. This could be an amended license issued under 10 CFR 50.82, under which any reactor licensee may apply for termination of the operating license and to decommission the facility. When the reactor is put into a condition in which it cannot operate, the operating license would be amended to permit the licensee to possess the byproduct, source, and special nuclear material remaining on the site. Storage of spent fuel in dry casks under the general license could continue under the amended license, which is often called a "possession-only" license.

Decommissioning means to remove a facility from service, reduce the residual radioactivity to a level that permits termination of the license, and release of the site for unrestricted use. Spent fuel stored under a general license must be removed before the site can be released for unrestricted use (i.e., decommissioned).

21. *Comment.* The proposed rule is unclear as to when the general license would terminate if a cask model has been reapproved by NRC following use of the cask for a period of up to 20 years. One commenter also suggested that § 72.212(a)(2) be changed to read: "The general license for the storage of spent fuel in each cask fabricated under a Certificate of Compliance shall terminate either 20 years after the date that the cask is first used by the licensee to store spent fuel, or, if the cask model is reapproved for storage of fuel for more than 20 years, at the conclusion of this newly-approved storage period, beginning on the date that the cask is first used by the licensee to store spent fuel."

Response. The intent of proposed § 72.212(a)(2) is that spent fuel may be stored under a valid Certificate of Compliance for a particular cask for a period of us to 20 years starting on the date the cask is first used for storage of spent fuel by the licensee. If a cask design is reapproved, the 20-year storage period begins anew, including casks of that design that remain in use. The 20-year storage period will also apply to new casks put into use after a Certificate of Compliance is reapproved. If a particular cask's Certificate of Compliance expires, the spent fuel stored in casks of this design must be removed after a period not exceeding 20 years following first use by the general licensee of a particular cask. Revisions have been made to 10 CFR 72.212(a)(2) to more accurately reflect this intent.

22. *Comment.* The $150 application fee shown in § 70.31 should be included in the total fee for the license and not required to be submitted at the time of the application.

Response. The Federal Register notice for the proposed rule was in error in that it indicated a revision to § 70.31; the revision is actually being made to § 170.31. The Commission agrees that the $150 filing fee is not required to be submitted at the time of the application. The necessary changes to eliminate the filing fee have been made in § 170.31. This is consistent with a similar change made with respect to filing fees in § 170.21 effective January 30, 1989. There is no application fee for the general license. However, the Commission has decided that it will assess fees for those inspections conducted under the general license (§ 72.212(b)(1)(iii)).

23. *Comment.* Cask vendors, some of which are small businesses, will be affected by the rule and should be considered in the Regulatory Flexibility Act Certification statement.

Response. Under this rulemaking the NRC will recover full costs, which are currently estimated to be between $250,000 and $300,000 for cask vendors. No other significant incremental impacts are anticipated, because the criteria for cask design approvals in this final rule are not significantly different from those currently required under part 72. The Regulatory Flexibility Act Certification Section of the final rule has been revised accordingly.

24. *Comment.* Some qualification is needed for the requirement in § 72.212(b)(2) that a licensee perform written evaluations showing compliance with the cask's certificate for the anticipated total number of casks to be used for storage. There is no certainty regarding when any spent fuel will be accepted by DOE, and this uncertainty should be clarified in the final rule.

Response. Each cask SAR includes an analysis of cask arrays, and licensees must consider these analyses in their selection of a cask model. Multiple storage arrays may be used if additional spent fuel storage capacity is needed. However, it was not intended that licensees be required to anticipate how much storage capacity would be needed before DOE begins accepting spent fuel for storage or disposal. Thus, revisions to § 72.212(b)(2) have been made to clarify the intent.

25. *Comment* Spent fuel should be required to be stored in the reactor fuel storage pool for a minimum of 5 years prior to dry cask storage. Such a provision would place considerably less thermal stress on the storage casks. Other commenters also questioned why this was not made a requirement.

Response. It is likely that the spent fuel will be stored in the reactor fuel pool for at least 5 years before storage in a cask. However, it is not necessary to make this a requirement, because casks can be designed to safely store spent fuel having a wide range of previous pool storage times.

26. *Comments.* The language in proposed 10 CFR 72.230 should be changed to reflect the condition that an application for certification of a storage cask must be made available to the public.

Response. The language of this section parallels the language in § 72.20 [§ 72.13] on which it is based, i.e., that "Applications and documents submitted to the Commission in connection with applications may be made available for public inspection in accordance with provisions of the regulations contained in parts 2 and 9 of this chapter." In general, applications will be made available except to the extent that they contain information exempt from disclosure such as proprietary or classified information.

27. *Comments.* The proposed rule should be modified to include alternative storage technologies. Two commenters indicated that the proposed rule approval of only one storage technology (i.e., spent fuel storage in dry casks) provides an unfair competitive advantage to suppliers of these systems.

Response. The reasons for Commission approval of spent fuel storage in dry casks are discussed in the Federal Register notice for the proposed rule. An important consideration is that free-standing casks, being very strong and massive structures, are independent of the effects of site-specific natural phenomena. For instance, in a worst case scenario considering the effects of earthquakes, a cask could topple. Forces from this fall would be well within a cask's design limits for safe confinement of radioactivity. Importantly, site-specific approvals would not be required by the Commission, provided conditions in subpart K are met. One system specifically mentioned in the comments is NUHOMS (registered trade mark by NUTECH Inc.), which consists of storing spent fuel in sealed canisters and storing the canisters in concrete modules. Another system mentioned is the Modular Vault Dry Store (FW Energy Applications, Inc.), which consists of storing the spent fuel in sealed containers and storing the containers in racks set in concrete or earth for shielding. A major reason that these spent fuel storage systems, which are being considered by the Commission for use under a general license, are not being approved at this time is that they have components that are dependent on site-specific parameters and; thus, require site-specific approvals. For instance the concrete storage modules used in the NUHOMS system and the racks and concrete shielding required by the Modular Vault Dry Store system, which are structures and systems important to safety, are usually constructed in-place and require site-specific evaluations of earthquake intensity and soil characteristics.

28. *Comment.* Paragraph 5 and 6 of "Discussion" in the proposed rule Federal Register notice did not include NUHOMS topical safety analysis reports (TSAR), although they have been approved by the staff.

Response. Two topical safety analysis reports for NUHOMS systems have been reviewed and approved by the NRC staff. Approval of a TSAR allows an applicant for a specific license under Part 72 to reference the document, instead of having to develop separate safety evaluations.

29. *Comments.* A licensee should be required to register use of casks prior to actual use of the cask, rather than within 30 days. Another commenter stated that the Commission has not demonstrated that the requirement to report initial storage of spent fuel in a cask within 30 days is the least burdensome necessary to achieve the Commission's objective. This commenter suggested that this information could be reported at the annual inventory.

Reponse. The purpose of the registration notice in § 72.212(b)(1)(ii) is to enable NRC's Office of Nuclear Material Safety and Safeguards to establish and maintain a record of the use of each cask. If safety issues arise during storage of spent fuel under the general license, they will be reported under § 72.216. The purpose of the records related to spent fuel inventory, required under § 72.72 [§ 72.51], is to enable NRC's Office of Nuclear Reactor Regulation to inspect for compliance with safeguards regulations. The information submitted under § 72.212(b)(1)(ii) is necessary to enable the NRC to take appropriate action in a timely manner on any issue that may arise.

30. *Comments.* The proposed rule requires that spent fuel storage cask designers give consideration to compatibility of cask designs with transportation and ultimate disposal by DOE. Some commenters favored this consideration and others questioned its advisability, unless specific criteria could be provided. Some commenters indicated that NRC should also address the lack of consistency between parts 71 and 72.

Response. Specific design criteria for spent fuel disposal may not be available until a repository design is approved. However, cask designers should remain aware that spent fuel ultimately will be received by DOE and that cask designs should adopt DOE criteria as they become available. This does not mean that cask designs previously certified by NRC will have to be recertified for this reason in order to continue to store spent fuel.

It is not necessary that storage casks be designed for transport of spent fuel (i.e., to meet requirements in part 71), because the spent fuel could be unloaded and transferred into transport casks approved under part 71, if necessary. However, in the interest of reducing radiation exposure, storage casks should be designed to be compatible with transportation and DOE design criteria to the extent practicable. Transportation compatibility will be attainable to the extent that cask designers can avoid return of spent fuel from dry storage to reactor basins for transfers to a transport cask before moving it off-site for disposal.

31. *Comment.* Section 72.236 should be revised to read "The criteria in § 72.236 (a) through (l) and (m)."

Response. Section 72.236(m) states that, to the extent practicable in the design of casks, consideration should be given to the compatibility of the dry storage cask system and components with transportation and other activities related to the removal of the stored spent fuel from the reactor site for ultimate disposition by DOE. DOE is developing repository storage designs that will be acceptable for use at their permanent spent fuel storage facility. However, specific criteria for designing spent fuel storage casks for compatibility may not be available until the design for a high-level waste repository is complete. Revision of § 72.236 is not considered to be appropriate at this time, although requirements in proposed § 72.236(m) have been retained separately.

32. *Comment.* The environmental assessment fails to conform to the requirements of the National Environmental Protection Act of 1969 (NEPA) and the guidelines of the Council on Environmental Quality (CEQ).

Response. The Commission's regulations for implementing section 102(2) of NEPA in a manner consistent with NRC's domestic licensing and related regulatory authority under the Atomic Energy Act are set forth in 10 CFR part 51. These regulations were revised in March of 1984 (49 FR 9352), taking into account the guidelines of CEQ. The environmental assessment for this rule was performed in conformity with the agency's environmental review procedures in 10 CFR part 51 and thereby conforms to NEPA requirements.

33. *Comment.* While the public notice provides a list of documents which contain current information, a supplemental environmental impact statement is required in order to inform the public as to the nature of the information and to allow an opportunity for public comment.

Response. Potential environmental impacts related to this rulemaking were analyzed in its environmental assessment, in previous rulemakings related to revision of part 72, and in the Commission's waste confidence proceedings that resulted in publication of the Waste Confidence Decision in the Federal Register on August 31, 1984 (49 FR 34658). In its waste confidence proceedings the Commission found that it has reasonable assurance that no significant environmental impacts will result from the storage of spent fuel for at least 30 years beyond the expiration of nuclear power reactor operating licenses. As a result of its Waste Confidence Decision, the Commission revised its regulations in 10 CFR 51.23 to eliminate discussion of the environmental impact of spent fuel storage in reactor storage pools or independent spent fuel storage installations for the period following the term of the license. In addition, the Commission recently published a review of its waste confidence decision (54 FR 39765; September 27, 1989). Accordingly, an environmental assessment, rather than an environmental impact statement, is considered suitable for this rulemaking. Also all of these documents were published in the Federal Register to allow an opportunity for public comment.

34. *Comment.* The NRC has misrepresented the requirements of the NWPA. The environmental assessment

and finding of no significant environmental impact states that the NWPA directs the Commission to approve one or more technologies for use of spent fuel storage. While the demonstration program is mandated, the adoption of one or more technologies is not.

Response. Section 218(a) of the NWPA does not direct the Commission to approve any spent fuel storage technology. However, the objective of the demonstration program is clearly meant to provide the basis for Commission approval of one or more technologies for use at civilian nuclear power reactor sites. Section 133 of the NWPA directs that the Commission shall, by rule, establish procedures for the licensing of any technology approved by the Commission under section 218(a). Thus, the NRC has properly represented the directives of the NWPA. The environmental assessment explains this relationship in the section entitled "The Need for the Proposed Action."

35. *Comments.* The NRC failed to discuss the consequences of a failure of its assumptions. The NRC states that the potential for corrosion of fuel cladding and reaction with the fuel is reduced "because an inert atmosphere is expected to be maintained" inside the casks. Further, the NRC "anticipates that most spent fuel stored in the casks will be 5 years old or more." What are the consequences if the scenarios the NRC "anticipates" does not happen?

Response. The potential consequences from off-normal and accident conditions involving spent fuel storage were discussed in the proposed rule. Licensees are required to store spent fuel under the general license, in accordance with the regulations in 10 CFR part 72 and the cask's Certificate of Compliance. Part 72 prohibits the storage of spent fuel that is less than 1 year old. The Certificate of Compliance requires that the spent fuel be stored in accordance with the technical specifications developed in the safety analysis report. These specifications set forth the age, number of fuel assemblies, maximum initial enrichment, maximum burnup, and maximum heat generation rate of the spent fuel. I[illegible] general terms, the longer the spent fuel is aged, the greater the capacity of the cask. Cask atmospheres will be required to be filled with an inert gas and provided with monitoring systems to detect leaks in the cask sealing system. If the redundant seals and the monitoring system fail, oxidation of the fuel cladding could occur if the inert gas leaked out, atmospheric air leaked in, and the internal cask temperature increased markedly. But, there would not be any significant increase in radioactivity, because any release of radioactive particles from the fuel rods would remain confined within the cask. If the redundant seals fail and the monitoring system does not fail, the monitoring system would detect the failure and the seals would be promptly repaired. If removal of the spent fuel were required, unloading procedures call for checking the cask's atmosphere before removing the lid and the radioactive material within the cask would be retained by the reactor fuel handling facility containment systems with no significant release to the environment.

Improper loading of spent fuel aged for less than 5 years is readily detectable by spent fuel assembly identification, independent verification, and monitoring procedures. If an improper fuel loading should occur, the results would be limited to a marginally higher storage temperature and possibly a slight increase in radiation from the cask. Any significant increase in temperature or radiation would be detected through procedures for cask monitoring, which have been added to the requirements in the Certificate of Compliance.

36. *Comments.* The criteria for locating storage cask sites, for ensuring adequate cooling for casks, for evaluating the adequacy of radiation shielding, or for other aspects of cask designs in the proposed rule have not been assessed for environmental impact.

Response. These technical criteria have been assessed and are currently used by the NRC for approval of cask designs under part 72. As previously mentioned, the environmental impacts related to storage of spent fuel under part 72 have been generically evaluated under two previous rulemakings and the Commission's waste confidence proceedings. Thus, these potential environmental impacts need not be reassessed.

37. *Comment.* The environmental impact of decommissioning contaminated casks after the 20-year storage period has not been assessed.

Response. The decommissioning of contaminated casks was discussed in the environmental assessment for this rule, which points out that decommissioning of dry cask spent fuel storage under a general license may be carried out as part of the power reactor site decommissioning plan. Decommissioning would consist of removing the spent fuel from the site and decontaminating cask surfaces. Alternately, this decontamination could take place at a DOE operated facility. In either case, the decontamination solutions would be combined with larger volumes of contaminated solutions resulting from decontamination of the reactor or DOE facility; thus, environmental impacts from decommissioning casks are expected to be a small fraction of the overall decommissioning impacts. Also the incremental costs associated with decommissioning casks are expected to represent a small fraction of the cost of decommissioning a nuclear power reactor. It is noted that, if the decommissioning of a reactor presents no significant safety hazard and if there is no significant change in types or amounts of effluents or increase in radiation exposure, then this decommissioning is covered by a categorical exclusion under 10 CFR 51.22.

38. *Comment.* The fire in the spent fuel storage pool subsequent to the major accident at Chernobyl has not been considered in the proposed rulemaking.

Response. In the early stages of the Chernobyl accident a hypothesis was developed that a fire occurred in the spent fuel pool. This hypothesis was not based on observation of any real fire at the Chernobyl installation, but rather inferred from fallout spectra observed in eastern Europe. Officials of the USSR have confirmed that indeed a fire did not occur in the spent fuel pool at Chernobyl. In fact, a fire in a spent fuel storage pool is not credible and, therefore, was not considered in the proposed rulemaking.

39. *Comment.* The NRC has studied responses of loaded casks to a range of sabotage scenarios. The four casks that are referenced in the background information are all metal casks, and there is limited reference to concrete systems. Because the referenced study is classified, we do not have any indication that this study specifically addressed concrete dry storage systems with respect to small arms, fire, and explosives.

Response. The referenced study did not specifically consider concrete storage systems. However, the general conclusions of the study could be extended to concrete storage systems because of the difficulty of using small arms, fire, or explosives to (1) create respirable particles and (2) cause those particles to be spread off site. These difficulties derive from both the inherent resistance to dispersal of the spent fuel and the massiveness of the storage casks required to provide both shielding from radiation and protection of the spent fuel from earthquakes and tornado missiles, which are requirements that all designs must meet.

40 *Comments* Safeguards requirements were either inadequate or too stringent. One commenter stated that the safeguards system for the existing site cannot be considered adequate for the additional burden of spent fuel cask storage. Unless a utility commits to a location for cask storage adjacent to the reactor building, the existing safeguards can be compromised and any cask storage area should be located greater than 100 meters from the nearest public access (roadway, park, beach, etc.). Another commenter suggested that terrorists need targets and that above-ground storage of spent fuel provides terrorists with a target. It further stated that a small bomb dropped from a light plane or helicopter could spread the contents of an above-ground cask over many states. Another commenter stated that there is no reason why the licensee should be exempt from §§ 73.55(h)(4)(iii)(A) and 73.55(h)(5), which requires that guards interpose themselves between vital areas and any adversary, and respond using deadly force if necessary. Another commenter stated that § 73.55 requirements are not needed for a spent fuel storage area that is a new protected area separate from the existing reactor protected area. This commenter further stated that the background material for this proposed rule indicates that requirements should be significantly reduced from § 73.55 requirements for storage areas within a new separate protected area and, specifically, that § 72.212 should specify the requirements instead of referencing exemptions from § 73.55.

Response. As described in the proposed rule (54 FR 19379), none of the information the staff has collected confirms the presence of an identifiable domestic threat to cask storage facilities. Despite the absence of an identifiable domestic threat, the NRC considered it prudent to study the response of loaded casks to a range of sabotage scenarios. After considering various technical approaches to radiological sabotage, and experiments and calculations, the NRC concluded that radiological sabotage, to be successful, would have to be carried out using large quantities of explosives, not a small bomb dropped from an airplane, and that the consequences to public health and safety would be low because most of the resultant contamination would be localized to the storage site. (See response to comment 39 above.) Thus, the condition to be protected against is protracted loss of control of the storage area. For that reason, protection requirements were proposed to provide for (1) early detection of malevolent moves against the storage site and (2) a means to quickly summon response forces to ensure protection against protracted loss of control of the storage area. Given these conditions, exemptions were provided for those § 73.55 provisions not essential to early detection of malevolent acts and for summoning local law enforcement agencies or other response forces. With the exception of one change in the rule that is being adopted (which is consistent with the intent of the proposed rule and is discussed in Comment 46), the NRC does not believe that these comments provide any new information or sufficient rationale for changing the proposed rule. Further, 10 CFR 72.106(b) requires that the minimum distance from the storage facility to the nearest boundary of the controlled area shall be at least 100 meters.

41 *Comment*. Could the cask body be the protected area boundary?

Response No, because that would not meet the requirements in § 73.55(c) for an isolation zone. An isolation zone must be maintained adjacent to the physical barrier and must be of sufficient size to permit observation of the activities of people on either side of the barrier in the event of its penetration. Thus, the cask body cannot be the physical barrier.

42. *Comment*. Please clarify the requirement for a periodic inventory of the special nuclear material contained in the spent fuel.

Response. It is the same as the current requirement for periodic inventory of special nuclear material that is required by § 72.72 (§ 72.51). Cask records must show the contents of the cask, including the special nuclear material. In lieu of periodically opening a cask, a licensee may use tamper indicating seals to show that the cask has not been opened. If any tamper indicating seals are broken, then the contents of the cask may have to be verified.

43 *Comment*. The requirements for vital areas are delineated in other paragraphs of § 73.55, and all vital area requirements throughout § 73.55 should be exempted in 10 CFR 72.212(b)(5)(ii), not just § 73.55(c).

Response. The NRC agrees with this comment. Proposed § 72.212(b)(5)(ii) states that storage of spent fuel under this general license need not be within a separate vital area. If spent fuel is not stored within a vital area (i.e., rather in a separate protected area), then regulations that pertain only to vital areas would not apply to a spent fuel storage area.

44. *Comment*. Paragraph (b)(5)(iii) of § 72.212 should distinguish between the security requirements for an existing protected area that is expanded and a new protected area. In the case of a new protected area, § 73.55(h)(6) should not be required. Instead, the requirement should be only an alarm assessment via CCTV, guard, or watchman.

Response. The NRC agrees with this comment. For an existing protected area, the current requirements will continue. Proposed §§ 72.212(b)(5) (iii) and (iv) have been revised to apply only to new protected areas. Proposed § 72.212(b)(5)(iv) has been revised to allow a guard or watchman on patrol in lieu of closed circuit television to provide the necessary observational capability.

45. *Comment*. For purposes of this rule, if the licensee is exempt from §§ 73.55(h)(4)(iii)(A) and (5) (i.e., neutralize threat), then § 73.55(h)(3) requirements (i.e., number of armed responders) should also be exempted.

Response. The general license presumes that the same essential physical security organization and program will be applied to spent fuel storage as are currently applied to protection of the reactor. Paragraph (b)(5)(i) of § 72.212 requires that the organization and program be modified as necessary to ensure that there is no decrease in effectiveness. Accordingly, additional personnel need be added only if it is necessary to ensure that there is no decrease in effectiveness. The rule does not require an independent application of § 73.55(h)(3), which specifies the minimum number of armed responders for a spent fuel storage area.

46. *Comment*. The requirement in § 73.55(d)(1) that searches for firearms and explosives be accomplished by equipment designed for such detection should be deleted when a new protection area is added that is not contiguous with the existing protection area. The only requirement in this case should be to perform a visual search for bulk explosives. This is supported by the discussion in the Federal Register notice.

Response. The NRC agrees that searches for firearms and explosives for the purposes of a general license under this rulemaking need not be conducted using equipment capable of detecting these devices. Accordingly, the final rule had been revised to allow the use of physical pat-down searches, in lieu of detection equipment, for firearms and explosives searches.

47. *Comments.* Is the use of the word "defect" in § 72.216(a) consistent with the definition of "defect" in 10 CFR part 21? What is the purpose of the reporting requirements in proposed § 50.72(b)(2)?

Response. Section 72.216(a) states that cask users must report defects discovered in storage cask systems, structures, and components important to safety and any instance in which there is a significant reduction in the effectiveness of a cask's confinement system. This information is necessary to inform the NRC of potential hazards to the public health and safety. Proposed § 72.216(a) is not being revised to replace the word defect, because the definition of "defect" in 10 CFR part 21 is compatible with the intent of this reporting requirement. However, proposed § 50.72(b)(2) is being revised to clarify such reporting, in order to avoid an apparent duplication of reporting requirements.

48. *Comment.* Proposed § 72.234(d)(3) requires a composite record for all casks to be maintained by the cask vendor "for the life of the cask." It further states that the vendor would not necessarily be in a position to know how long the general license will be extended; thus, this provision should be clarified.

Response. The intent of this section is that cask vendors should maintain a record of all casks that are fabricated and sold or leased to power reactor licensees. This record would be used by the NRC to confirm information supplied by cask users and to determine whether or not a cask vendor will submit an application for cask design reapproval. The commenter raised a valid point, thus, § 72.234(d)(3) has been revised to require only a composite record of casks fabricated.

49. *Comment.* The Commission has not demonstrated the practical utility of requiring cask fabrication initiation and completion dates to be included as part of the cask record in § 72.234(d)(2) (iv) and (v).

Response. The purpose for including the cask fabrication initiation and completion dates in a cask record is to ensure that any safety problem that might arise related to fabrication procedures of a particular cask model can be traced and corrected in all casks of that model. For instance, if a faulty batch of steel is fabricated into closure bolts, which could be discovered through quality assurance procedures, these fabrication dates would enable the staff to determine which specific casks were involved. Thus, corrective actions could be taken, if necessary, based on this information.

50. *Comments.* Although § 72.6(b) [§72.6] provides for issuance of a general license, § 72.6(c) might be interpreted to disallow storage of spent fuel in an ISFSI by a licensee under the general license, unless the holder of such a license also has a specific license for that purpose. One commenter suggested that existing § 72.6(c) be revised or clarified to specifically provide for storage of spent fuel under a general license without the requirement for a specific license, as long as the provisions of subpart K are met.

Response. Paragraph 72.6(c) has been revised to make an exception of spent fuel storage under a general license according to the provisions of subpart K. Subpart K sets forth conditions under which the holder of a power reactor operating license may store spent fuel under the general license being promulgated by this rulemaking. Conditions set forth in § 72.6 are now considered sufficient to allow storage of spent fuel under the general license. However, it is not intended that this rule serve as authorization for storage of spent fuel in amounts or for durations beyond those provided for in a power reactor license.

Having considered all comments received and other input, the Commission has determined that the following final rule should be promulgated.

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 (EIS) is not required. The finding is premised on two actions, which are (i) the licensing of an operating reactor for a particular site for which an EIS has been previously prepared and (ii) the independent certification of spent fuel storage casks for use at any reactor site. Thus, the rule does not add any significant environmental impacts and does not change any safety requirements. The environmental assessment and finding of no significant impact on which this determination is based are available for inspection at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC.

Paperwork Reduction Act Statement

This final rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.). These requirements were approved by the Office of Management and Budget with approval numbers 3150-0011 and 3150-0132.

Public reporting burden for this collection of information is estimated to average 134 hours per response for a power reactor licensee and 2,448 hours per response for a cask vendor licensee including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for reducing this burden, to the Information and Records Management Branch (MNBB-7714), U.S. Nuclear Regulatory Commission, Washington, DC 20555; and to the Paperwork Reduction Project (3150-0011 and 3150-0132), Office of Management and Budget, Washington, DC 20503.

Regulatory Analysis

The Commission prepared a preliminary regulatory analysis for the proposed rulemaking on this subject. The analysis examined the benefits and impacts considered by the Commission. The Commission requested public comments on the preliminary regulatory analysis, but no comments were received. No changes to the regulatory analysis are considered necessary, so as separate regulatory analysis has not been prepared for the final rule.

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 final rule affects licensees owning nuclear power reactors. Owners of nuclear power reactors 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.

Only one cask model is currently being used to store spent fuel under 10 CFR part 72, but an additional three cask models are being certified under § 72.214 of this final rule. Companies involved in the design, manufacture, and sale of casks are large private entities employing more than 500 persons and having sales in excess of $1 million. Some companies involved in the actual sale of these casks may not employ over 500 persons, but have sales in excess of

$1 million. These companies may fall within the scope of "small entities" as defined above, but there are not a substantial number of them. The Preliminary Regulatory Analysis, which was made available for public comment when the proposed rule was published, analyzed potential impacts on cask vendors. No comments were received on the analysis. In any case, cask vendors will decide whether or not to submit applications for cask design approval based on their analysis of the potential market.

Backfit Analysis

The NRC has determined that the backfit rule, 10 CFR 50.109, does not apply to this final rule, and, thus, a backfit analysis is not required, because these amendments do not contain any provisions which would impose backfits as defined in § 50 109(a)(1).

List of Subjects

10 CFR Part 50

Antitrust, Classified information, Criminal penalty, Fire protection, Incorporation by reference, Intergovernmental relations, Nuclear power plants and reactors, Radiation protection, Reactor siting criteria, and Reporting and recordkeeping requirements

10 CFR Part 72

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

10 CFR Part 170

Byproduct material, Non-payment penalties, Nuclear materials, Nuclear power plants and reactors, 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, as amended, and 5 U.S.C. 552 and 553, the NRC is adopting the following revisions to 10 CFR part 72 and conforming amendments to 10 CFR parts 50 and 170.

Table 1

STATUS OF DRY SPENT FUEL STORAGE LICENSING

Topical Reports Approved

TYPE	VENDOR	MODEL	CAPACITY (Assemblies)	DATE OF NRC STAFF APPROVAL
Metal Cask	General Nuclear Systems, Inc.	CASTOR V/21	21 PWR	9/85
Concrete Module	NUTECH, Inc.*	NUHOMS-7P	7 PWR	3/86
Metal Cask	Westinghouse	MC-10	24 PWR	9/87
Metal Cask	Nuclear Assurance Corporation	S/T	26 PWR	3/88
Concrete Vault	FW Energy Applications, Inc.	Modular Vault Dry Store	83 PWR or 150 BWR	3/88
Metal Cask	Nuclear Assurance Corporation	NAC-C28 S/T	28 canisters (for fuel rods from 56 PWR assemblies)	9/88
Metal Cask	Nuclear Assurance Corporation	NAC-I28 S/T**	28 PWR	2/90
Concrete Module	NUTECH, Inc.*	NUHOMS-24P	24 PWR	4/89
Metal Cask	Transnuclear, Inc.	TN-24	24 PWR	7/89
Concrete Cask	Pacific Sierra Nuclear Associates	VSC	24 PWR	3/91

* Firm's name changed to Pacific Nuclear Fuel Services, Inc.
** Identical to NAC-C28 S/T, but reviewed and approved by NRC staff for storage of intact fuel assemblies.

ENCLOSURE 3

Topical Reports Under Review

TYPE	VENDOR	MODEL	CAPACITY (Assemblies)
Metal Cask	General Nuclear Systems, Inc.	CASTOR X	28 PWR or 33 PWR
Metal Cask	Nuclear Assurance Corporation	NAC-STC (Dual Purpose)	26 PWR

Certificates of Compliance for Dry Spent fuel Storage Casks

TYPE	VENDOR	DOCKET AND CERT. NOS.	DATE OF ISSUANCE	MODEL	CAPACITY (Assemblies)
Metal Cask	General Nuclear Systems, Inc.	72-1000 1000	8/17/90	CASTOR V/21	21 PWR
Metal Cask	Westinghouse	72-1001 1001	8/17/90	MC-10	24 PWR
Metal Cask	Nuclear Assurance Corporation	72-1002 1002	8/17/90	NAC S/T	26 PWR
Metal Cask	Nuclear Assurance Corporation	72-1003 1003	8/17/90	NAC-C28 S/T	28 canisters (for fuel rods from 56 PWR assemblies

Licenses Issued

UTILITY	REACTOR SITE	DOCKET AND LICENSE NOS.	DATE OF ISSUANCE	MODEL
Virginia Electric Power Co.	Surry Power Station (Surry County, Virginia)	72-2; SNM-2501	7/86	Castor V/21 MC-10 NAC-I28 S/T Castor X
Carolina Power and Light Co.	H. B. Robinson Steam Electric Plant, Unit 2 (Darlington Co., South Carolina)	72-3; SNM-2502	8/86	NUHOMS-7P
Duke Power Co.	Oconee Nuclear Station, Oconee County, South Carolina	72-4; SNM-2503	1/90	NUHOMS-24P

License Applications Received

UTILITY	REACTOR SITE	DOCKET NO.	DATE OF RECEIPT	MODEL
Carolina Power and Light Co.	Brunswick Steam Electric Plant	72-6	5/89	NUHOMS-7P
Consumers Power	Palisades	72-7	3/90 withdrawn 8/90	VSC
Baltimore Gas and Electric Co.	Calvert Cliffs Nuclear Power Plant	72-8	12/89	NUHOMS-24P
Public Service Colorado	Ft. St. Vrain	72-9	6/90	MVDS
Northern States Power	Prairie Island	72-10	9/90	Casks (TN)

License Applications Expected

UTILITY	REACTOR SITE	DATE EXPECTED	MODEL
Carolina Power and Light Co.	H. B. Robinson	1990	NUHOMS-7P

APPLICATIONS FOR CERTIFICATE OF COMPLIANCE

Vendor	Cask Model	Capacity	Date Received	Docket No.
Pacific Nuclear Fuel Services, Inc.	Standardized (NUHOMS)		1/91	72-1004
	NUHOMS-24P	24 PWR		
	NUHOMS-52B	52 BWR		
Transnuclear	TN-24	24 PWR	1/91	72-1005
B&W Fuel Co.	CONSTAR	32 PWR	expected	72-1006
Pacific Sierra Fuel	Concrete VSC-24	24 PWR	4/91	72-1007

TOPICAL REPORTS FOR STORAGE CASKS SUBMITTED TO FUEL CYCLE SAFETY BRANCH

Project M-34, 37, 50 Docket 72-1000	General Nuclear Systems, Inc. ATTN: Robert T. Anderson, Director 220 Stoneridge Drive Columbia, SC 29210
Project M-39, 49 Docket 72-1004	Pacific Nuclear Fuel Services, Inc. Mr. William J. McConaghy 145 Martinvale Lane San Jose, CA 95119
Project M-40, 51 52, 54, & 55 Docket 72-1002 Docket 72-1003	Nuclear Assurance Corporation ATTN: James M. Viebrock Senior Vice President Engineering and Transportation Systems 6251 Crooked Creek Road Suite 200 Norcross, GA 30092
Project M-41 Docket 72-1001	Westinghouse Electric Corporation ATTN: William J. Johnson, Manager Nuclear Safety Department Box 355 Pittsburgh, PA 15230-0355
Project M-42 Docket 72-1005	Transnuclear, Inc. ATTN: Alan Hanson President Two Skyline Drive Hawthorne, NY 10532-2120

Project M-46

FW Energy Applications, Inc.
ATTN: H. C. Pickering, Jr.
President & Chief Executive Officer
8 Peach Tree Hill Road
Livingston, NJ 07039

Project M-53
Docket 72-1007

Pacific Sierra Nuclear Associates
ATTN: James V. Massey, Ph.D.
General Manager
5619 Scotts Valley Drive
Scotts Valley, CA 95066

Docket 72-1006

B&W Fuel Company
ATTN: Garry Garner
Project Manager
3315 Old Forest Road
P. O. Box 10935
Lynchburg, VA 24506-0935

PART 71

PACKAGING AND TRANSPORTATION OF RADIOACTIVE MATERIAL

Authority: Secs. 53, 57, 62, 63, 81, 161, 182, 183, 68 Stat. 930, 932, 933, 935, 948, 953, 954, as amended (42 U.S.C. 2073, 2077, 2092, 2093, 2111, 2201, 2232, 2233); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended, 1244, 1246 (42 U.S.C. 5841, 5842, 5846).

Section 71.97 also issued under sec. 301, Pub. L. 96-295, 94 Stat. 789-790.

For the purposes of sec. 223, 68 Stat. 958, as amended (42 U.S.C. 2273), §§ 71.3, 71.43, 71.45, 71.55, 71.63(a) and (b), 71.83, 71.85, 71.87, 71.89, and 71.97 are issued under sec. 161b, 68 Stat. 948, as amended (42 U.S.C. 2201(b)); and §§ 71.5(b), 71.6a, 71.91, 71.93, 71.95, and 71.101(a) are issued under sec. 161o, 68 Stat. 950, as amended (42 U.S.C. 2201(o)).

Subpart A—General Provisions

§ 71.0 Purpose and scope.

(a) This part establishes: (1) requirements for packaging, preparation for shipment, and transportation of licensed material; and (2) procedures and standards for NRC approval of packaging and shipping procedures for fissile material and for a quantity of other licensed material in excess of a Type A quantity.

(b) The packaging and transport of licensed material are also subject to other parts of this chapter (e.g., Parts 20, 21, 30, 39, 40, 70, and 73) and to the regulations of other agencies (e.g., the U.S. Department of Transportation (DOT) and the U.S. Postal Service (USPS) [1]) having jurisdiction over means of transport. The requirements of this part are in addition to, and not in substitution for, other requirements.

(c) The regulations in this part apply to any licensee authorized by specific license issued by the Commission to receive, possess, use, or transfer licensed material if the licensee delivers that material to a carrier for transport or transports the material outside the confines of the licensee's facility, plant,

[1] Postal Service Manual (Domestic Mail Manual), section 124.3, which is incorporated by reference at 39 CFR 111.2 (1974).

or other authorized place of use. No provision of this part authorizes possession of licensed material.

(d) Exemptions from the requirement for license in § 71.3 are specified in § 71.10. General licenses for which no NRC package approval is required are issued in §§ 71.14–71.24. The general license in § 71.12 requires that an NRC certificate of compliance or other package approval be issued for the package to be used under the general license. Application for package approval must be completed in accordance with Subpart D of this part, demonstrating that the design of the package to be used satisfies the package approval standards contained in Subpart E of this part as related to the tests of Subpart F of this part. The transport of licensed material or delivery of licensed material to a carrier for transport is subject to the operating controls and procedures requirements of Subpart G of this part, to the quality assurance requirements of Subpart H of this part, and to the general provisions of Subpart A of this part, including DOT regulations referenced in § 71.5.

48 FR 35600

§ 71.1 Communications and records.

(a) All communications concerning the regulations in this part should be addressed to the Director, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555, or may be delivered in person at the Commission Office at 2120 L Street, NW, Washington, DC, or its Offices at 11555 Rockville Pike, Rockville, Maryland.

53 FR 43419

(b) Each record required by this part must be legible throughout the retention period specified by each Commission regulation. The record may be the original or a reproduced copy or a microform provided that the copy or microform is authenticated by authorized personnel and that the microform is capable of producing a clear copy throughout the required retention period. The record may also be stored in electronic media with the capability for producing legible, accurate, and complete records during the required retention period. Records such as letters, drawings, specifications, must include all pertinent information such as stamps, initials, and signatures. The licensee shall maintain adequate safeguards against tampering with and loss of records.

53 FR 19240

§ 71.2 Interpretations.

Only written interpretations of the regulations in this part by the Commission's General Counsel are binding upon the Commission.

§ 71.3 Requirement for license.

A licensee subject to the regulations in this part may not (a) deliver any licensed material to a carrier for transport or (b) transport licensed material except as authorized in a general license or a specific license issued by the Commission, or as exempted in this part.

48 FR 35600

§ 71.4 Definitions.

The following terms are as defined here for the purpose of this part. Throughout this part, the standards are expressed in metric units; the approximate English equivalents presented in parentheses are for information only.

A_1 means the maximum activity of special form radioactive material permitted in a Type A package. *A_2* means the maximum activity of radioactive material, other than special form radioactive material, permitted in a Type A package. These values are either listed in Appendix A of this part, Table A-1, or may be derived in accordance with the procedure prescribed in Appendix A of this part.

Carrier means a person engaged in the transportation of passengers or property by land or water as a common, contract, or private carrier, or by civil aircraft.

Close reflection by water means immediate contact by water of sufficient thickness for maximum reflection of neutrons.

48 FR 35600

Containment system means the components of the packaging intended to retain the radioactive material during transport.

Conveyance means any vehicle, aircraft, vessel, freight container, or hold, compartment, or defined deck area of an inland waterway craft or seagoing vessel.

Exclusive use (also referred to in other regulations as "sole use" or "full load") means the sole use of a conveyance by a single consignor and for which all initial, intermediate, and final loading and unloading are carried out in accordance with the direction of the consignor or consignee.

Fissile classification means the categorization of fissile material packages into one of the following three classes according to the controls needed to provide nuclear criticality safety during transportation:

(1) Fissile Class I: A package which may be transported in unlimited numbers and in any arrangement, and which requires no nuclear criticality safety controls during transportation. A transport index is not assigned for purposes of nuclear criticality safety but may be required because of external radiation levels.

(2) Fissile Class II: A package which may be transported together with other packages in any arrangement but, for criticality control, in numbers which do not exceed an aggregate transport index of 50. These shipments require no other nuclear criticality safety control during transportation. Individual packages may have a transport index not less than 0.1 and not more than 10.

(3) Fissile Class III: A shipment of packages which is controlled in transportation by specific arrangements between the shipper and the carrier to provide nuclear criticality safety.

Fissile material and *fissile radionuclides*: "Fissile material" means any material consisting of or containing one or more fissile radionuclides. Fissile radionuclides are plutonium-238, plutonium-239, plutonium-241, uranium-233, and uranium-235. Neither natural nor depleted uranium is fissile material. Fissile materials are classified in this section according to the controls needed to provide nuclear criticality safety during transportation. Certain exclusions are provided in § 71.53.

48 FR 35600

Low specific activity material means any of the following:

(1) Uranium or thorium ores and physical or chemical concentrates of those ores;

(2) Unirradiated natural or depleted uranium or unirradiated natural thorium;

(3) Tritium oxide in aqueous solutions provided the concentration does not exceed 5.0 millicuries per milliliter;

(4) Material in which the radioactivity is essentially uniformly distributed and in which the estimated average concentration per gram of contents does not exceed:

(i) 0.0001 millicurie of radionuclides for which the A_2 quantity in Appendix A of this part is not more than 0.05 curie;

(ii) 0.005 millicurie of radionuclides for which the A_2 quantity in Appendix A of this part is more than 0.05 curie, but not more than 1 curie; or

(iii) 0.3 millicurie of radionuclides for which the A_2 quantity in Appendix A of this part is more than 1 curie.

(5) Objects of nonradioactive material externally contaminated with radioactive material, provided that the radioactive material is not readily dispersible and the surface contamination, when averaged over an area of 1 square meter, does not exceed 0.0001 millicurie (220,000 disintegrations per minute) per square centimeter of radionuclides for which the A_2 quantity in Appendix A of Part 71 is not more than 0.05 curie, or 0.001 millicurie (2,200,000 disintegrations per minute) per square centimeter for other radionuclides.

48 FR 38449

Maximum normal operating pressure means the maximum gauge pressure that would develop in the containment system in a period of one year under the heat test specified in § 71.71(c)(1), in the absence of venting, external cooling by an ancillary system, or operational controls during transport.

48 FR 35600

Natural thorium means thorium with the naturally occurring distribution of thorium isotopes (essentially 100 weight percent thorium-232).

Normal form radioactive material means radioactive material which has not been demonstrated to qualify as "special form radioactive material."

Optimum interspersed hydrogenous moderation means the presence of hydrogenous material between packages to such an extent that the maximum nuclear reactivity results.

Package means the packaging together with its radioactive contents as presented for transport.

(1) *Fissile material package* means a fissile material packaging together with its fissile contents.

(2) *Type B package* means a Type B package together with its radioactive contents. On approval, a Type B package design is designated by NRC as B(U) unless the package has a maximum normal operating pressure of more than 700 kilopascal (100 lb/in²) gauge or a pressure relief device which would allow the release of radioactive material to the environment under the tests specified in § 71.73 (hypothetical accident conditions), in which case it will receive a designation B(M). B(U) refers to the need for unilateral approval of international shipments. B(M) refers to the need for multilateral approval. There is no distinction made in how packages with these designations may be used in domestic transportation. To determine their distinction for international transportation, see DOT regulations in 49 CFR Part 173. A Type B package approved prior to September 6, 1983, was designated only as Type B. Limitations on its use are specified in § 71.13.

48 FR 35600

Packaging means the assembly of components necessary to ensure compliance with the packaging requirements of this part. It may consist of one or more receptacles, absorbent materials, spacing structures, thermal insulation, radiation shielding, and devices for cooling or absorbing mechanical shocks. The vehicle, tie-down system, and auxiliary equipment may be designated as part of the packaging.

Special form radioactive material means radioactive material which satisfies the following conditions:

(1) It is either a single solid piece or is contained in a sealed capsule that can be opened only by destroying the capsule;

(2) The piece or capsule has at least one dimension not less than 5 millimeters (0.197 inch); and

(3) It satisfies the test requirements of § 71.75.

A special form encapsulation designed in accordance with the requirements of § 71.4(o) of this part in effect on June 30, 1983, and constructed prior to July 1, 1985 may continue to be used. A special form encapsulation either designed or constructed after June 30, 1985 must meet requirements of this paragraph applicable at the time of its design or construction.

Specific activity of a radionuclide means the radioactivity of the radionuclide per unit mass of that nuclide. The specific activity of a material in which the radionuclide is essentially uniformly distributed is the radioactivity per unit mass of the material.

State means the several States of the Union, the District of Columbia, the Commonwealth of Puerto Rico, the Virgin Islands, Guam, American Samoa, the trust Territory of the Pacific Islands, and the Commonwealth of the Northern Mariana Islands.

Transport index means the dimensionless number (rounded up to the first decimal place) placed on the label of a package to designate the degree of control to be exercised by the carrier during transportation. The transport index is determined as follows:

(1) The number expressing the maximum radiation level in millirem per hour at 1 meter from the external surface of the package; or

(2) For Fissile Class II packages, the number expressing the maximum radiation level in millirem per hour at 1 meter from the external surface of the package, or the number obtained by dividing 50 by the allowable number of the packages which may be transported together as determined under § 71.59, whichever number is larger.

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Type A quantity means a quantity of radioactive material, the aggregate radioactivity of which does not exceed A_1 for special form radioactive material or A_2 for normal form radioactive material, where A_1 and A_2 are given in Appendix A of this part or may be determined by procedures described in Appendix A of this part.

Type B quantity means a quantity of radioactive material greater than a Type A quantity.

Uranium—natural depleted enriched

(1) *Natural uranium* means uranium with the naturally occurring distribution of uranium isotopes (approximately 0.711 weight percent uranium-235, and the remainder essentially uranium-238).

(2) *Depleted uranium* means uranium containing less uranium-235 than the naturally occurring distribution of uranium isotopes.

(3) *Enriched uranium* means uranium containing more uranium-235 than the naturally occurring distribution of uranium isotopes.

§ 71.5 Transportation of licensed material.

(a) Each licensee who transports licensed material outside of the confines of its plant or other place of use, or who delivers licensed material to a carrier for transport, shall comply with the applicable requirements of the regulations appropriate to the mode of transport of DOT in 49 CFR Parts 170 through 189.

(1) The licensee shall particularly note DOT regulations in the following areas:

(i) Packaging—49 CFR Part 173, Subparts A and B and §§ 173.401-173.478.

(ii) Marking and labeling—49 CFR Part 172, Subpart D and §§ 172.400-172.407; 172.436-172.440.

(iii) Placarding—49 CFR Part 172.500-172.519, 172.556 and Appendices B and C.

(iv) Monitoring—49 CFR Part 172, Subpart C.

(v) Accident reporting—49 CFR Part 171.15 and 171.16.

(vi) Shipping papers—49 CFR Part 172, Subpart C.

(2) The licensee shall also note DOT regulations pertaining to the following modes of transportation:

(i) Rail—49 CFR Part 174, Subparts A-D and K.

(ii) Air—49 CFR Part 175, Subparts A-D and M.

(iii) Vessel—49 CFR Part 176, Subparts A-D and M.

(iv) Public Highway—49 CFR Part 177.

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(b) If DOT regulations are not applicable to a shipment of licensed material by rail, highway, or water because the shipment or the transportation of the shipment is not in interstate or foreign commerce, or to a shipment of licensed material by air because the shipment is not transported in civil aircraft, the licensee shall conform to the standards and requirements of the DOT specified in paragraph (a) of this section to the same extent as if the shipment or transportation were in interstate or foreign commerce or in civil aircraft. A request for modification, waiver, or exemption from those requirements, and any notification referred to in those requirements, must be filed with or made to the Director, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

§ 71.8 Information collection requirements: OMB approval.

49 FR 19633

(a) The Nuclear Regulatory Commission has submitted the information collection requirements contained in this part to the Office of Management and Budget (OMB) for approval as required by the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.). OMB has approved the information collection requirements contained in this part under control number 3150-0008.

(b) The approved information collection requirements contained in this part appear in §§ 71.5, 71.12, 71.31, 71.33, 71.35, 71.37, 71.85, 71.87, 71.89, 71.91, 71.93, 71.95, 71.97, 71.101, 71.103, 71.105, 71.107, 71.109, 71.111, 71.113, 71.115, 71.117, 71.119, 71.121, 71.123, 71.125, 71.127, 71.129, 71.131, 71.133, 71.135, and 71.137.

§ 71.6a Completeness and accuracy of information.

(a) Information provided to the Commission by an applicant for a license or by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the applicant or the licensee shall be complete and accurate in all material respects.

(b) Each applicant or licensee shall notify the Commission of information identified by the applicant or licensee as having for the regulated activity a significant implication for public health and safety or common defense and security. An applicant or licensee violates this paragraph only if the applicant or licensee fails to notify the Commission of information that the applicant or licensee has identified as having a significant implication for public health and safety or common defense and security. Notification shall be provided to the Administrator of the appropriate Regional Office within two working days of identifying the information. This requirement is not applicable to information which is already required to be provided to the Commission by other reporting or updating requirements.

52 FR 49362

Subpart B—Exemptions

§ 71.7 Specific exemptions.

On application of any interested person or on its own initiative, the Commission may grant any exemption from the requirements of the regulations in this part that it determines is authorized by law and will not endanger life or property or the common defense and security.

§ 71.8 [Reserved]

§ 71.9 Exemption of physicians.

Any physician licensed by a State of the United States to dispense drugs in the practice of medicine is exempt from § 71.5 with respect to transport by the physician of licensed material for use in the practice of medicine. However, any physician operating under this exemption must be licensed under 10 CFR Part 35.

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§ 71.10 Exemption for low level materials.

(a) A licensee is exempt from all requirements of this part with respect to shipment or carriage of a package containing radioactive material having a specific activity not greater than 0.002

microcurie/gram.

(b) A licensee is exempt from all requirements of this part, other than § 71.5 and § 71.88, with respect to shipment or carriage of the following packages:

(1) A package containing no more than a Type A quantity of radioactive material if the package contains no fissile material or if the fissile material exemption standards of § 71.53 are satisfied; or

(2) A package transported between locations within the United States which contains only americium or plutonium in special form with an aggregate radioactivity not to exceed 20 curies, if the package contains no fissile material or if the fissile material exemption standards of § 71.53 are satisfied.

§ 71.11 [Reserved]

Subpart C—General Licenses

§ 71.12 General license: NRC approved package.

(a) A general license is hereby issued to any licensee of the Commission to transport, or to deliver to a carrier for transport, licensed material in a package for which a license, certificate of compliance, or other approval has been issued by the NRC.

(b) This general license applies only to a licensee who has a quality assurance program approved by the Commission as satisfying the provisions of Subpart H of this part.

(c) This general license applies only to a licensee who:

(1) Has a copy of the specific license, certificate of compliance, or other approval of the package and has the drawings and other documents referenced in the approval relating to the use and maintenance of the packaging and to the actions to be taken prior to shipment;

(2) Complies with the terms and conditions of the license, certificate, or other approval, as applicable, and the applicable requirements of Subparts A, C, and H of this part; and

(3) Submits in writing to the Director, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555, prior to the licensee's first use of the package, the licensee's name and license number and the package identification number specified in the package approval.

(d) This general license applies only when the package approval authorizes use of the package under this general license.

(e) For previously approved Type B packages which are not designated as either B(U) or B(M) in the NRC Certificate of Compliance, this general license is subject to the additional restrictions of § 71.13.

§ 71.13 Previously approved Type B package.

(a) A Type B package previously approved by the NRC, but not designated as B(U) or B(M) in the NRC Certificate of Compliance, may be used under the general license of § 71.12 with the following additional limitations:

(1) Fabrication of the packaging was satisfactorily completed before August 31, 1986, as demonstrated by application of its model number in accordance with § 71.85(c); and

(2) The package may not be used for a shipment to a location outside the United States after August 31, 1986, except under special arrangement approved by DOT in accordance with 49 CFR 173.471.

(b) The NRC will approve modifications to the design and authorized contents of a Type B package previously approved by the NRC, but not designated as B(U) or B(M) in the NRC Certificate of Compliance, provided:

(1) The modifications are not significant with respect to the design, operating characteristics, or safe performance of the containment system when the package is subjected to the tests specified in §§ 71.71 and 71.73; and

(2) The modification to the package satisfies the requirements of this part.

(c) The NRC will revise the package identification number to designate previously approved Type B package designs as B(U) or B(M) after receipt of an application demonstrating that the design meets the requirements of this part.

§ 71.14 General license: DOT specification container.

(a) A general license is issued to any licensee of the Commission to transport or to deliver to a carrier for transport licensed material in a specification container for fissile material or for a Type B quantity of radioactive material as specified in the regulations of DOT in 49 CFR Parts 173 and 178.

(b) This general license applies only to a licensee who has a quality assurance program approved by the Commission as satisfying the provisions of Subpart H of this part.

(c) This general license applies only to a licensee who:

(1) Has a copy of the specification; and

(2) Complies with the terms and conditions of the specification and the applicable requirements of Subparts A, C, and H of this part.

(d) This general license is subject to the limitation that the specification container may not be used for a shipment to a location outside the United States after August 31, 1986, except under special arrangements approved by DOT in accordance with 49 CFR 173.472.

§ 71.16 General License: Use of foreign approved package.

(a) A general license is issued to any licensee of the Commission to transport or to deliver to a carrier for transport licensed material in a package the design of which has been approved in a foreign national competent authority certificate which has been revalidated by DOT as meeting the applicable requirements of 49 CFR 171.12.

(b) This general license applies only to shipments made to or from locations outside the United States.

(c) This general license applies only to a licensee who:

(1) Has a copy of the applicable certificate, the revalidation, and the drawings and other documents referenced in the certificate relating to the use and maintenance of the packaging and to the actions to be taken prior to shipment; and

(2) Complies with the terms and conditions of the certificate and revalidation and with the applicable requirements of Subparts A, C, and H of this part. With respect to the quality assurance provisions of Subpart H of this part, the licensee is exempt from design, construction, and fabrication considerations.

§ 71.18 General license: Type A, Fissile Class II package.

(a) A general license is issued to any licensee of the Commission to transport fissile material, or to deliver fissile material to a carrier for transport, without complying with the package standards of Subparts E and F of this part if the material is shipped as a Fissile Class II package.

(b) This general license applies only when a package contains no more than a Type A quantity of radioactive material, including only one of the following:

(1) Up to 40 grams of uranium-235; or

(2) Up to 30 grams of uranium-233; or

(3) Up to 25 grams of the fissile radionuclides of plutonium, except that for encapsulated plutonium-beryllium neutron sources in special form, an A_1 quantity of plutonium may be present; or

(4) A combination of fissile radionuclides in which the sum of the ratios of the amount of each radionuclide to the corresponding

maximum amounts in paragraphs (b) (1), (2), and (3) of this section does not exceed unity.

(c) This general license applies only when, except as specified below for encapsulated plutonium-beryllium sources, a package containing more than 15 grams of fissile radionuclides is labeled with a transport index not less than the number given by the following equation, where the package contains x grams of uranium-235, y grams of uranium-233 and z grams of the fissile radionuclides of plutonium:

$$\text{Minimum Transport Index} = (0.40x + 0.67y + z)\left(1 - \frac{15}{x+y+z}\right).$$

For a package in which the only fissile material is in the form of encapsulated plutonium-beryllium neutron sources in special form, the transport index based on criticality considerations may be taken as 0.026 times the number of grams of the fissile radionuclides of plutonium in excess of 15 grams. In all cases, the transport index must be rounded up to one decimal place, and may not exceed 10.0.

§ 71.20 General license: Restricted, Fissile Class II package.

(a) A general license is issued to any licensee of the Commission to transport fissile material, or to deliver fissile material to a carrier for transport, without complying with the package standards of Subparts E and F of this part if the material is shipped as a Fissile Class II package.

(b) This general license applies only when:

(1) The package contains no more than a Type A quantity of radioactive material, and

(2) Neither beryllium nor hydrogenous material enriched in deuterium is present; and

(3) The total mass of graphite present does not exceed 150 times the total mass of uranium-235 plus plutonium; and

(4) Substances having a higher hydrogen density than water, e.g., certain hydrocarbon oils, are not present, except that polyethylene may be used for packing or wrapping; and

(5) Uranium-233 is not present, and the amount of plutonium does not exceed 1% of the amount of uranium-235; and

(6) The amount of uranium-235 is limited as follows:

(i) If the fissile radionuclides are not uniformly distributed, the maximum amount of uranium-235 per package may not exceed the value given in Table I of this part, or

(ii) If the fissile radionuclides are distributed uniformly (i.e., cannot form a lattice arrangement within the packaging) the maximum amount of uranium-235 per package may not exceed the value given in Table II of this part; and

(7) The transport index of each package based on criticality considerations is taken as 10 times the number of grams of uranium-235 in the package divided by the maximum allowable number of grams per package in accordance with Table I or Table II of this part as applicable.

TABLE I.—PERMISSIBLE MASS OF URANIUM-235 PER FISSILE CLASS II PACKAGE APPLICABLE TO § 71.20(b)(6)(i)

[Nonuniform distribution]

Uranium enrichment in weight percent of uranium-235 not exceeding	Permissible maximum grams of uranium-235 per package
24	40
20	42
15	45
11	48
10	51
9.5	52
9	54
8.5	55
8	57
7.5	59
7	60
6.5	62
6	65
5.5	68
5	72
4.5	76
4	80
3.5	88
3	100
2.5	120
2	144
1.5	272
1.35	320
1	680
0.92	1,200

TABLE II.—PERMISSIBLE MASS OF URANIUM-235 PER FISSILE CLASS II PACKAGE APPLICABLE TO § 71.20(b)(6)(ii)

[Uniform distribution]

Uranium enrichment in weight percent of uranium-235 not exceeding	Permissible maximum grams of uranium-235 per package
4	84
3.5	92
3	112
2.5	148
2	240
1.5	440
1.35	800

§ 71.22 General license: Type A package, Fissile Class III shipment.

(a) A general license is issued to any licensee of the Commission to transport fissile material, or to deliver fissile material to a carrier for transport, without complying with the package standards of Subparts E and F of this part if limited material is shipped as a Fissile Class III shipment.

(b) This general license applies only when a package contains no more than a Type A quantity of radioactive material and no more than 400 grams total of the fissile radionuclides of plutonium encapsulated as plutonium-beryllium neutron sources in special form.

(c) This general license applies only when the fissile radionuclides in the Fissile Class III shipment exceeds none of the following:

(1) 500 grams of uranium-235; or

(2) 300 grams total of uranium-233, and the fissile radionuclides of plutonium; or

(3) A total quantity of uranium-233, uranium-235, and the fissile radionuclides of plutonium such that the sum of the ratios of the quantity of each radionuclide to the quantity specified in paragraphs (c)(1) and (c)(2) of this section exceeds unity; or

(4) 2500 grams total of the fissile radionuclides of plutonium encapsulated as plutonium-beryllium neutron sources in special form.

(d) This general license applies only when shipment of these packages is made under procedures specifically authorized by DOT in accordance with 49 CFR Part 173 of its regulations to prevent loading, transport or storage of these packages with other Fissile Class II packages or Fissile Class III shipments.

§ 71.24 General license: Restricted, Fissile Class III shipment.

(a) A general license is issued to any licensee of the Commission to transport fissile material, or to deliver fissile material to a carrier for transport, without complying with the package standards of Subparts E and F of this part if limited material is shipped as a Fissile Class III shipment.

(b) This general license applies only when:

(1) No package contains more than a Type A quantity of radioactive material; and

(2) The packaging does not incorporate lead shielding exceeding 5 cm in thickness, tungsten shielding, or uranium shielding; and

(3) Neither beryllium nor hydrogenous material enriched in deuterium is present; and

(4) The total mass of graphite present does not exceed 150 times the total mass of uranium-235 and plutonium; and

(5) Substances having a higher hydrogen density than water, e.g., certain hydrocarbon oils, are not present, except that polyethylene may be used for packing or wrapping; and

(6) For fissile contents containing no uranium-233 and less than 1% total plutonium:

(i) If the fissile radionuclides are not uniformly distributed, the maximum amount of uranium-235 per consignment does not exceed the value given in Table III of this part; or

(ii) If the fissile radionuclides are distributed uniformly and cannot form a lattice arrangement within the packaging, the maximum amount of uranium-235 per shipment does not exceed the value given in Table IV of this part; and

(7) For fissile contents containing uranium-233 or more than 1% plutonium, the total mass of fissile material per shipment is limited so that the sum of the number of grams of uranium-235 divided by 400, the number of grams of plutonium divided by 225, and the number of grams of uranium-233 divided by 250 does not exceed unity as expressed in the formula

$$\frac{\text{grams uranium}-235}{400\text{ grams}}+\frac{\text{grams plutonium}}{225\text{ grams}}+\frac{\text{grams uranium}-233}{250\text{ grams}}\leq 1;\text{ and}$$

(8) The transport must be direct to the consignee without any intermediate transit storage; and

(9) Shipment of these packages is made under procedures specifically authorized by DOT in accordance with 49 CFR Part 173 of its regulations to prevent loading, transport or storage of these packages with other Fissile Class II packages or Fissile Class III shipments.

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TABLE III.—PERMISSIBLE MASS OF URANIUM-235 PER FISSILE CLASS III SHIPMENT APPLICABLE TO § 71.24(b)(6)(i)

[Nonuniform distribution]

Uranium enrichment in weight percent of uranium-235 not exceeding	Permissible maximum grams of uranium-235 per consignment
20	520
15	580
11	600
10	640
9.5	664
9	675
8.5	690
8	710
7.5	730
7	750
6.5	780
6	810
5.5	860
5	900
4.5	960
4	1,000
3.5	1,100
3	1,250
2.5	1,300
2	2,050
1.5	3,400
1.35	4,000
1	8,500
0.92	15,500

TABLE IV.—PERMISSIBLE MASS OF URANIUM-235 PER FISSILE CLASS III SHIPMENT APPLICABLE TO § 71.24(b)(6)(ii)

[Uniform distribution]

Uranium enrichment in weight percent of uranium-235 not exceeding	Permissible maximum grams of uranium-235 per consignment
4	1,050
3.5	1,180
3	1,400
2.5	1,800
2	2,000
1.5	7,000
1.35	10,000

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Subpart D—Application for Package Approval

§ 71.31 Contents of application.

(a) An application for an approval under this part must include, for each proposed packaging design, the following information:

(1) A package description as required by § 71.33;

(2) A package evaluation as required by § 71.35;

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(3) A quality assurance program description as required by § 71.37;

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(4) In the case of fissile material, an identification of the proposed fissile class.

(b) Except as provided in § 71.13, an application for modification of a package design, whether for modification of the packaging or authorized contents, must include sufficient information to demonstrate that the proposed design satisfies the package standards in effect at the time the application is filed.

§ 71.33 Package description.

The application must include a description of the proposed package in sufficient detail to identify the package accurately and provide a sufficient basis for evaluation of the package. The description must include:

(a) With respect to the packaging:

(1) Classification as Type B(U), Type B(M), or fissile material packaging.

(2) Gross weight;

(3) Model number;

(4) Identification of the containment system;

(5) Specific materials of construction, weights, dimensions, and fabrication methods of:

(i) Receptacles;

(ii) Materials specifically used as nonfissile neutron absorbers or moderators;

(iii) Internal and external structures supporting or protecting receptacles;

(iv) Valves, sampling ports, lifting devices, and tie-down devices;

(v) Structural and mechanical means for the transfer and dissipation of heat; and

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(6) Identification and volumes of any receptacles containing coolant.

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(b) With respect to the contents of the package:

(1) Identification and maximum radioactivity of radioactive constituents;

(2) Identification and maximum quantities of fissile constituents;

(3) Chemical and physical form;

(4) Extent of reflection, the amount and identity of nonfissile materials used as neutron absorbers or moderators, and the atomic ratio of moderator to fissile constituents;

(5) Maximum normal operating pressure;

(6) Maximum weight;

(7) Maximum amount of decay heat; and

(8) Identification and volumes of any coolants.

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§ 71.35 Package evaluation.

The application must include:

(a) A demonstration that the package satisfies the standards specified in Subparts E and F of this part;

(b) For a Fissile Class II package, the allowable number of packages which may be transported in the same vehicle in accordance with § 71.59, and

(c) For a Fissile Class III shipment, any proposed special controls and precautions for transport, loading, unloading, and handling, and any proposed special controls in the event of accident or delay.

§ 71.37 Quality assurance.

(a) The applicant shall describe the quality assurance program (see Subpart H of this part) for the design, fabrication, assembly, testing, maintenance, repair, modification, and use of the proposed package.

(b) The applicant shall identify any established codes and standards proposed for use in package design, fabrication, assembly, testing, maintenance, and use. In the absence of any codes and standards, the applicant shall describe the basis and rationale used to formulate the package quality assurance program.

(c) The applicant shall identify any specific provisions of the quality assurance program which are applicable to the particular package design under consideration, including a description of the leak testing procedures.

§ 71.39 Requirement for additional information.

The Commission may at any time require additional information in order to enable it to determine whether a license, certificate of compliance, or other approval should be granted, denied, modified, suspended, or revoked.

Subpart E—Package Approval Standards

§ 71.41 Demonstration of compliance.

(a) The effects on a package of the tests specified in § 71.71 (Normal Conditions of Transport) and the tests specified in § 71.73 (Hypothetical Accident Conditions) must be evaluated by subjecting a sample package or scale model to test, or by other method of demonstration acceptable to the Commission, as appropriate for the particular feature being considered.

(b) Taking into account the type of vehicle, the method of securing or attaching the package, and the controls to be exercised by the shipper, the Commission may permit the shipment to be evaluated together with the transporting vehicle.

(c) Environmental and test conditions different from those specified in § 71.71 and § 71.73 may be approved by the Commission if the controls proposed to be exercised by the shipper are demonstrated to be adequate to assure the safety of the shipment.

§ 71.43 General standards for all packages.

(a) The smallest overall dimension of a package must not be less than 10 cm (four in.).

(b) The outside of a package must incorporate a feature, such as a seal, which is not readily breakable, and which, while intact, would be evidence that the package has not been opened by unauthorized persons.

(c) Each package must include a containment system securely closed by a positive fastening device which cannot be opened unintentionally.

(d) A package must be of materials and construction which assure that there will be no significant chemical, galvanic, or other reaction among the packaging components or between the package components and the package contents, including possible reaction resulting from inleakage of water to the maximum credible extent.

(e) A package valve or other device, the failure of which would allow radioactive contents to escape, must be protected against unauthorized operation and, except for a pressure relief device, must be provided with an enclosure to retain any leakage.

(f) A package must be designed, constructed, and prepared for shipment so that under the tests specified in § 71.71 (Normal Conditions of Transport) there would be no loss or dispersal of radioactive contents, no significant increase in external radiation levels, and no substantial reduction in the effectiveness of the packaging.

(g) A package must be designed, constructed, and prepared for transport so that in still air at 38°C (100°F) and in the shade, no accessible surface of a package would have a temperature exceeding 50°C (122°F) in a nonexclusive use shipment or 82°C (180°F) in an exclusive use shipment.

(h) A package must not incorporate a feature which is intended to allow continuous venting during transport.

§ 71.45 Lifting and tie-down standards for all packages.

(a) Any lifting attachment that is a structural part of a package must be designed with a minimum safety factor of three against yielding when used to lift the package in the intended manner, and must be designed so that failure of any lifting device under excessive load would not impair the ability of the package to meet other requirements of this subpart. Any other structural part of the package which could be used to lift the package must be capable of being rendered inoperable for lifting the package during transport or must be designed with strength equivalent to that required for lifting attachments.

(b) Tie-down devices:

(1) If there is a system of tie-down devices which is a structural part of the package, the system must be capable of withstanding, without generating stress in any material of the package in excess of its yield strength, a static force applied to the center of gravity of the package having a vertical component of two times the weight of the package with its contents, a horizontal component along the direction in which the vehicle travels of 10 times the weight of the package with its contents, and a horizontal component in the transverse direction of five times the weight of the package with its contents.

(2) Any other structural part of the package which could be used to tie down the package must be capable of being rendered inoperable for tying down the package during transport, or must be designed with strength equivalent to that required for tie-down devices.

(3) Each tie-down device which is a structural part of a package must be designed so that failure of the device under excessive load would not impair the ability of the package to meet other requirements of this part.

§ 71.47 External radiation standards for all packages.

A package must be designed and prepared for shipment so that the radiation level does not exceed 200 millirem per hour at any point on the external surface of the package and the transport index does not exceed 10 (See § 71.4 "Definitions"). For a package transported as exclusive use by rail, highway, or water, radiation levels external to the package may exceed those limits, but must not exceed any of the following:

(a) 200 millirem/hour on the accessible external surface of the package unless the following conditions are met, in which case the limit is 1000 millirem per hour:

(1) The shipment is made in a closed transport vehicle;

(2) Provisions are made to secure the package so that its position within the vehicle remains fixed during transportation; and

(3) There are no loading or unloading operations between the beginning and end of the transportation;

(b) 200 millirem/hour at any point on the outer surface of the vehicle, including the upper and lower surfaces, or, in the case of an open vehicle, at any point on the vertical planes projected from the outer edges of the vehicle, on the upper surface of the load, and on the lower external surface of the vehicle;

(c) 10 millirem/hour at any point two

meters from the vertical planes represented by the outer lateral surfaces of the vehicle, or, in the case of an open vehicle, at any point two meters from the vertical planes projected from the outer edges of the conveyance; and

(d) Two millirem/hour in any normally occupied positions of the vehicle, except that this provision does not apply to private motor carriers when persons occupying these positions are provided with special health supervision, personnel radiation exposure monitoring devices, and training in accordance with § 19.12 of this chapter.

§ 71.51 Additional requirements for Type B packages.

(a) A Type B package, in addition to satisfying the requirements of §§ 71.41–71.47 must be designed, constructed, and prepared for shipment so that under the tests specified in:

(1) Section 71.71 (Normal Conditions of Transport), there would be no loss or dispersal of radioactive contents, as demonstrated to a sensitivity of 10^{-6} A_2 per hour, no significant increase in external radiation levels, and no substantial reduction in the effectiveness of the packaging; and

(2) Section 71.73 (Hypothetical Accident Conditions), there would be no escape of krypton–85 exceeding 10,000 curies in one week, no escape of other radioactive material exceeding a total amount A_2 in one week, and no external radiation dose rate exceeding one rem per hour at one meter from the external surface of the package.

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(b) Compliance with the permitted activity release limits of paragraph (a) of this section must not depend upon filters or upon a mechanical cooling system.

§ 71.52 Exemption for low specific activity (LSA) packages.

A package need not satisfy the requirements of § 71.51 if it contains only low specific activity material and is transported as exclusive use, but is subject to §§ 71.41–71.47 of this part, including § 71.43(f).

§ 71.53 Fissile material exemptions.

The following packages are exempt from fissile material classification and from the fissile material standards of §§ 71.55–71.61, but are subject to all other requirements of this part:

(a) A package containing not more than 15 grams of fissile radionuclides. If material is transported in bulk, the quantity limitation applies to the conveyance; or

(b) A package containing irradiated natural or depleted uranium including the products of irradiation if the irradiation has taken place only in a thermal reactor; or

(c) A package containing homogenous hydrogenous solutions or mixtures where:

(1) The minimum ratio of the number of hydrogen atoms to the number of atoms of fissile radionuclides(H/X) is 5200;

(2) The maximum concentration of fissile radionuclides is five grams/liter; and

(3) The maximum mass of fissile radionuclides in the package is 800 grams, except for a mixture where the total mass of plutonium and uranium-233 exceeds one percent of the mass of uranium-235 the limit is 500 grams. If the material is transported in bulk, the quantity limitations apply to the vehicle, to a hold or compartment of an inland waterway craft, or to a hold, compartment, or defined deck area of a seagoing vessel; or

(d) A package containing uranium enriched in uranium-235 to a maximum of one percent by weight, and with a total plutonium and uranium-233 content of up to one percent of the mass of uranium-235. If the fissile radionuclides are distributed homogeneously throughout the package contents, and do not form a lattice arrangement within the package; or

(e) A package containing any fissile material if it does not contain more than five grams of fissile radionuclides in any 10-liter volume, and if the material is packaged so as to maintain this limit of fissile radionuclide concentration during normal transport; or

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(f) A package containing not more than one kilogram of plutonium of which not more than 20% by mass may consist of plutonium-239, plutonium-241, or any combination of those radionuclides; or

(g) A package containing liquid solutions of uranyl nitrate enriched in uranium-235 to a maximum of two percent by weight, with total plutonium and uranium-233 not more than one-tenth percent of the mass of uranium-235.

§ 71.55 General requirements for all fissile material packages.

(a) A package used for the shipment of fissile material must be designed and constructed in accordance with §§ 71.41–71.47. When required by the total amount of radioactive material, a package used for the shipment of fissile material must also be designed and constructed in accordance with § 71.51.

(b) Except as provided in paragraph (c) of this section, a package used for the shipment of fissile material must be so designed and constructed and its contents so limited that it would be subcritical if water were to leak into the containment system or liquid contents were to leak out of the containment system so that, under the following conditions, maximum reactivity of the fissile material would be attained:

(1) The most reactive credible configuration consistent with the chemical and physical form of the material;

(2) Moderation by water to the most reactive credible extent; and

(3) Close reflection by water on all sides.

(c) The Commission may approve exceptions to the requirements of paragraph (b) of this section if the package incorporates special design features that ensure that no single packaging error would permit leakage, and if appropriate measures are taken before each shipment to ensure the containment system does not leak.

(d) A package used for the shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in § 71.71 (Normal Conditions of Transport):

(1) The contents would be subcritical;

(2) The geometric form of the package contents would not be substantially altered;

(3) There would be no leakage of water into the containment system unless, in the evaluation of undamaged packages under §§ 71.57(a), 71.59(b)(1), and 71.61(a), it has been assumed that moderation is present to such an extent as to cause maximum reactivity consistent with the chemical and physical form of the material; and

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(4) There will be no substantial reduction in the effectiveness of the packaging, including:

(i) No more than five percent reduction in the total effective volume of the packaging on which nuclear safety is assessed;

(ii) No more than five percent reduction in the effective spacing between the fissile contents and the outer surface of the packaging; and

(iii) No occurrence of an aperture in the outer surface of the packaging large enough to permit the entry of a 10 cm (four in.) cube.

(e) A package used for the shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in § 71.73 (Hypothetical Accident Conditions), the package would be subcritical. For this determination, it must be assumed that:

(1) The fissile material is in the most reactive credible configuration consistent with the damaged condition of the package and the chemical and physical form of the contents;

(2) Water moderation occurs to the most reactive credible extent consistent with the damaged condition of the package and the chemical and physical form of the contents; and

(3) There is reflection by water on all sides, as close as is consistent with the damaged condition of the package.

§ 71.57 Specific standards for a Fissile Class I package.

A Fissile Class I package must be so designed and constructed and its contents so limited that:

(a) Any number of undamaged packages would be subcritical in any arrangement and with optimum interspersed hydrogenous moderation unless there is a greater amount of interspersed moderation in the packaging, in which case the greater amount may be assumed for this determination; and

(b) Two hundred fifty (250) packages, if each package were subjected to the tests specified in § 71.73 (Hypothetical Accident Conditions), would be subcritical if stacked together in any arrangement, closely reflected on all sides of the stack by water, and with optimum interspersed hydrogenous moderation.

§ 71.59 Specific standards for a Fissile Class II package.

(a) A Fissile Class II package must be controlled by the carrier during transport. To provide this control, the designer of a Fissile Class II package must determine the allowable number of packages of that design which can be safely transported in a vehicle under the conditions specified in this section. This allowable number of packages determines the minimum transport index which the shipper of the package marks on the package label when the package is shipped. By limiting to 50 the total number of transport indexes in a vehicle or storage area, the carrier provides adequate criticality control.

(b) A Fissile Class II package must be designed and constructed and its contents so limited, and the allowable number of these packages in a Fissile Class II shipment so determined, that:

(1) Five times the allowable number of undamaged packages would be subcritical if stacked together in any arrangement and closely reflected on all sides of the stack by water; and

(2) Twice the allowable number of packages, if each package were subjected to the tests specified in § 71.73 (Hypothetical Accident Conditions), would be subcritical if stacked together in any arrangement, closely reflected on all sides of the stack by water, and with optimum interspersed hydrogenous moderation.

(c) The transport index with respect to criticality control for each Fissile Class II package must be calculated by dividing the number 50 by the allowable number of Fissile Class II packages which may be transported together as determined under the limitations of paragraph (b) of this section. The transport index so determined must not exceed 10 and must be rounded up to the first decimal place.

§ 71.61 Specific standards for a Fissile Class III shipment.

A package for Fissile Class III shipment must be so designed and constructed and its contents so limited, and the number of packages in a Fissile Class III shipment must be so limited, that:

(a) Twice this number of undamaged packages would be subcritical if stacked together in any arrangement, assuming close reflection on all sides of the stack by water; and

(b) This number of packages would be subcritical if stacked together in any arrangement, closely reflected on all sides of the stack by water, and with optimum interspersed hydrogenous moderation. Except as permitted under § 71.41, each package must be considered to have been subjected to the tests specified in § 71.73 (Hypothetical Accident Conditions).

§ 71.63 Special requirements for plutonium shipments.

(a) Plutonium in excess of 20 curies per package must be shipped as a solid.

(b) Plutonium in excess of 20 curies per package must be packaged in a separate inner container placed within outer packaging that meets the requirements of Subparts E and F for packaging of material in normal form. If the entire package is subjected to the tests specified in § 71.71 (Normal Conditions of Transport), the separate inner container must not release plutonium, as demonstrated to a sensitivity of 10^{-6} A_2 per hour. If the entire package is subjected to the tests specified in § 71.73 (Hypothetical Accident Conditions), the separate inner container must restrict the loss of plutonium to not more than A_2 in one week. Solid plutonium in the following forms is exempt from the requirements of this paragraph:

(1) Reactor fuel elements;

(2) Metal or metal alloy; and

(3) Other plutonium bearing solids that the Commission determines should be exempt from the requirements of this section.

§ 71.65 Additional requirements.

The Commission may, by rule, regulation, or order, impose requirements upon any licensee in addition to those established in this part as it deems necessary or appropriate to protect health or to minimize danger to life or property.

Subpart F—Package and Special Form Tests [8]

§ 71.71 Normal conditions of transport.

(a) *Evaluation.* Evaluation of each package design under normal conditions of transport must include a determination of the effect on that design of the conditions and tests specified in this section. Separate specimens may be used for the free drop test, the compression test, and the penetration test if each specimen is subjected to the water spray test before being subjected to any of the other tests.

(b) *Initial conditions.* With respect to the initial conditions for the tests in this section, the demonstration of compliance with the requirements of this Part must be based on the ambient temperature preceding and following the tests remaining constant at that value between −29°C (−20°F) and +38°C (100°F) which is most unfavorable for the feature under consideration. The initial internal pressure within the containment system must be considered to be the maximum normal operating pressure, unless a lower internal pressure consistent with the ambient temperature considered to precede and follow the tests is more unfavorable.

(c) *Conditions and tests.* (1) *Heat.* An ambient temperature of 38°C (100°F) in still air, and insolation according to the following table:

INSOLATION DATA

Form and location of surface	Total insolation for a 12-hour period (g cal/cm²)
Flat surfaces transported horizontally	
—Base	None.
—Other surfaces	800.
Flat surfaces not transported horizontally	200.
Curved surfaces	400.

(2) *Cold.* An ambient temperature of −40°C (−40°F) in still air and shade.

(3) *Reduced external pressure.* An external pressure of 24.5 kilopascal (3.5 psi) absolute.

(4) *Increased external pressure.* An external pressure of 140 kilopascal (20 psi) absolute.

(5) *Vibration.* Vibration normally incident to transport.

(6) *Water spray.* A water spray that simulates exposure to rainfall of approximately five cm (two in.) per hour for at least one hour.

(7) *Free drop.* Between 1½ and 2½ hours after the conclusion of the water spray test, a free drop through the distance specified below onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which maximum damage is expected. For Fissile Class II packages, this free drop must be preceded by a

[8] The package standards related to the tests in this subpart are contained in Subpart E.

free drop from a height of 0.3 m (one ft.) on each corner or, in the case of a cylindrical Fissile Class II package, onto each of the quarters of each rim.

CRITERIA FOR FREE DROP TEST (WEIGHT/DISTANCE)

Package weight		Free drop distance	
Kilograms	Pounds	Meters	Feet
5 000 or less	(11,000)	1.2	(4)
5 000 to 10 000	(11 000 to 22 000)	0.9	(3)
10 000 to 15 000	(22 000 to 33,000)	0.6	(2)
More than 15,000	(More than 33 000)	0.3	(1)

(8) *Corner drop.* A free drop onto each corner of the package in succession, or in the case of a cylindrical package onto each quarter of each rim, from a height of 0.3 m (one ft.) onto a flat, essentially unyielding horizontal surface. This test applies only to fiberboard or wood rectangular packages not exceeding 50 kg (110 pounds) and fiberboard or wood cylindrical packages not exceeding 100 kg (220 pounds).

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(9) *Compression.* For packages weighing up to 5000 kg, the package must be subjected, for a period of 24 hours, to a compressive load applied uniformly to the top and bottom of the package in the position in which the package would normally be transported. The compressive load must be the greater of the following:

(i) The equivalent of five times the weight of the package; or

(ii) The equivalent of 12.75 kilopascal (1.85 lb/in²) multiplied by the vertically projected area of the package.

(10) *Penetration.* Impact of the hemispherical end of a vertical steel cylinder of 3.2 cm (1¼ in) diameter and six kg (13 lb) mass, dropped from a height of one m (40 in) onto the exposed surface of the package which is expected to be most vulnerable to puncture. The long axis of the cylinder must be perpendicular to the package surface.

§ 71.73 Hypothetical accident conditions.

(a) *Test procedures.* Evaluation for hypothetical accident conditions is to be based on sequential application of the tests specified in this section, in the order indicated, to determine their cumulative effect on a package or array of packages. An undamaged specimen must be used for the water immersion test specified in paragraph (c)(5) of this section.

(b) *Test conditions.* With respect to the initial conditions for the tests, except for the water immersion tests, to demonstrate compliance with the requirements of this part during testing, the ambient air temperature before and after the tests must remain constant at that value between −29°C (−20°F) and +38°C (100°F) which is most unfavorable for the feature under consideration. The initial internal pressure within the containment system must be the maximum normal operating pressure unless a lower internal pressure consistent with the ambient temperature assumed to precede and follow the tests is more unfavorable.

(c) *Tests.* Tests for hypothetical accident conditions must be conducted as follows:

(1) *Free Drop.* A free drop of the specimen through a distance of nine m (30 ft) onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which maximum damage is expected.

(2) *Puncture.* A free drop of the specimen through a distance of one m (40 in) in a position for which maximum damage is expected, onto the upper end of a solid, vertical, cylindrical, mild steel bar mounted on an essentially unyielding, horizontal surface. The bar must be 15 cm (six in) in diameter, with the top horizontal and its edge rounded to a radius of not more than six mm (¼ in) and of a length as to cause maximum damage to the package, but not less than 20 cm (eight in) long. The long axis of the bar must be vertical.

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(3) *Thermal.* Exposure of the whole specimen for not less than 30 minutes to a heat flux not less than that of a radiation environment of 800°C (1475°F) with an emissivity coefficient of at least 0.9. For purposes of calculation, the surface absorptivity must be either that value which the package may be expected to possess if exposed to a fire or 0.8, whichever is greater. In addition, when significant, convective heat input must be included on the basis of still, ambient air at 800°C (1475°F). Artificial cooling must not be applied after cessation of external heat input and any combustion of materials of construction must be allowed to proceed until it terminates naturally. The effects of solar radiation may be neglected prior to, during, and following the test.

(4) *Immersion—fissile material.* For fissile material, in those cases where water inleakage has not been assumed for criticality analysis, the specimen must be immersed under a head of water of at least 0.9 m (three ft) for a period of not less than eight hours and in the attitude for which maximum leakage is expected.

(5) *Immersion—all packages.* A separate, undamaged specimen must be subjected to water pressure equivalent to immersion under a head of water of at least 15 m (50 ft) for a period of not less than eight hours. For test purposes, an external pressure of water of 147 kilopascal (21 psi) gauge is considered to meet these conditions.

§ 71.75 Qualification of special form radioactive material.

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(a) Evaluation of the contents of a single package for qualification as special form must include a determination of the effect on a specimen of those contents of the tests specified in § 71.77.

(1) Specimens (solid radioactive material or capsules) to be tested must be as normally prepared for loading in a single package, with the radioactive material duplicated as closely as practicable.

(2) A different specimen may be used for each of the tests.

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(b) The specimen must not break or shatter when subjected to the impact, percussion, or bending tests.

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(c) The specimen must not melt or disperse when subjected to the heat test.

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(d) After each test, leak-tightness or indispersibility of the specimen must be determined by a method no less sensitive than the following leaching assessment procedure. For a capsule resistant to corrosion by water, and which has an internal void volume greater than 0.1 milliliters, an alternative to the leaching assessment is a demonstration of leak-tightness of 10^{-4} torr-l/s (1.3×10^{-4} atm cm³/s) (based on air at 25°C and one atmosphere differential pressure) for solid radioactive content, or 10^{-6} torr-l/s (1.3×10^{-6} atm cm³/s) for liquid or gaseous radioactive content.

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(1) The specimen must be immersed for seven days in water at ambient temperature. The water must have a pH of 6–8 and a maximum conductivity of 10 μmho/cm at 20°C (68°F). Encapsulated material is not subject to the seven-day requirement.

(2) The water with specimen must then be heated to a temperature of 50°±5°C (122°±9°F) and maintained at this temperature for four hours.

(3) The activity of the water must be determined at that time.

(4) The specimen must then be stored for at least seven days in still air of humidity not less than 90% and a temperature not less than 30°C (86°F).

(5) The specimen must then be immersed in water having a pH of 6–8 and a maximum conductivity of 10 μmho/cm at 20°C, and the water with specimen heated to 50°±5°C (122°±9°F) and maintained at this temperature for four hours.

(6) The activity of the water must be determined at that time.

(7) The activities determined in paragraphs (c)(3) and (c)(6) of this section must not exceed 0.05 μCi.

§ 71.77 Tests for special form radioactive material.

(a) *Impact test.* The specimen must fall onto a flat, horizontal, essentially unyielding surface from a height of not less than nine m (30 ft).

(b) *Percussion test.* The specimen must be placed on a sheet of lead which is supported by a smooth solid surface and struck by the flat face of a steel billet so as to produce an impact equivalent to that resulting from a free fall of 1.4 kg (three lb.) through one m (40 in.). The flat face of the billet must be 25 mm (one in.) in diameter with the edges rounded to a radius of three mm (0.12 in.)±0.3 mm (0.012 in.). The lead, of hardness number 3.5 to 4.5 on the Vickers scale and not more than 25 mm (one in.) thick, must cover an area greater than that covered by the specimen. A fresh surface of lead must be used for each impact. The billet must strike the specimen so as to cause maximum damage.

(c) *Bending test.* The test is applicable only to long, slender sources with both a minimum length of 10 cm (four in.) and a length to minimum width ratio not less than 10. The specimen must be rigidly clamped in a horizontal position so that one-half of its length protrudes from the face of the clamp. The orientation of the specimen must be such that the specimen will suffer maximum damage when its free end is struck by the flat face of a steel billet. The billet must strike the specimen so as to produce an impact equivalent to that resulting from a free vertical fall of 1.4 kg (three lb.) through one m (40 in.). The flat face of the billet must be 25 mm (one in.) in diameter with the edges rounded off to a radius of three mm (0.12 in.)±0.3 mm (0.012 in.).

(d) *Heat test.* The specimen must be heated to a temperature of not less than 800°C (1475°F) in an atmosphere which is essentially air, and held at that temperature for a period of 10 minutes and must then be allowed to cool.

Subpart G—Operating Controls and Procedures

§ 71.81 Applicability of operating controls and procedures.

A licensee subject to this part, who under a general or specific license transports licensed material or delivers licensed material to a carrier for transport, shall comply with the requirements of this Subpart G, with the quality assurance requirements of Subpart H of this part, and with the general provisions of Subpart A of this part.

§ 71.83 Assumptions as to unknown properties.

When the isotopic abundance, mass, concentration, degree of irradiation, degree of moderation, or other pertinent property of fissile material in any package is not known, the licensee shall package the fissile material as if the unknown properties have credible values that will cause the maximum nuclear reactivity.

§ 71.85 *Preliminary determinations.*

Prior to the first use of any packaging for the shipment of licensed material:

(a) The licensee shall ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects which could significantly reduce the effectiveness of the packaging;

(b) Where the maximum normal operating pressure will exceed 34.3 kilopascal (5 psi) gauge, the licensee shall test the containment system at an internal pressure at least 50% higher than the maximum normal operating pressure to verify the capability of that system to maintain its structural integrity at that pressure.

(c) The licensee shall conspicuously and durably mark the packaging with its model number, gross weight, and a package identification number assigned by the Nuclear Regulatory Commission. Prior to applying the model number, the licensee shall determine that the packaging has been fabricated in accordance with the design approved by the Commission.

§ 71.87 Routine determinations.

Prior to each shipment of licensed material, the licensee shall ensure that the package with its contents satisfies the applicable requirements of this part and of the license. The licensee shall determine that:

(a) The package is proper for the contents to be shipped;

(b) The package is in unimpaired physical condition except for superficial defects such as marks or dents;

(c) Each closure device of the packaging, including any required gasket, is properly installed and secured and free of defects;

(d) Any system for containing liquid is adequately sealed and has adequate space or other specified provision for expansion of the liquid;

(e) Any pressure relief device is operable and set in accordance with written procedures;

(f) The package has been loaded and closed in accordance with written procedures;

(g) For fissile material, any moderator or neutron absorber, if required, is present and in proper condition;

(h) Any structural part of the package which could be used to lift or tie down the package during transport is rendered inoperable for that purpose unless it satisfies the design requirements of § 71.45;

(i)(1) The level of non-fixed (removable) radioactive contamination on the external surfaces of each package offered for shipment is as low as reasonably achievable. The level of non-fixed radioactive contamination may be determined by wiping an area of 300 square centimeters of the surface concerned with an absorbent material, using moderate pressure, and measuring the activity on the wiping material. Sufficient measurements must be taken in the most appropriate locations to yield a representative assessment of the non-fixed contamination levels. Except as provided under paragraph (i)(2) of this section, the amount of radioactivity measured on any single wiping material when averaged over the surface wiped, must not exceed the limits given in Table V of this part at any time during transport. Other methods of assessment of equal or greater efficiency may be used. When other methods are used, the detection efficiency of the method used must be taken into account and in no case may the non-fixed contamination on the external surfaces of the package exceed ten times the limits listed in Table V.

TABLE V.—REMOVABLE EXTERNAL RADIOACTIVE CONTAMINATION WIPE LIMITS

Contaminant	Maximum permissible limits	
	μCi/cm²	dpm/cm²
Beta-gamma emitting radionuclides; all radionuclides with half-lives less than ten days; natural uranium; natural thorium; uranium-235; uranium-238; thorium-232; thorium-228 and thorium 230 when contained in ores or physical concentrates	10^{-5}	22
All other alpha emitting radionuclides	10^{-6}	2.2

(2) In the case of packages transported as exclusive use shipments by rail or highway only, the non-fixed radioactive contamination at any time during transport must not exceed ten times the levels prescribed in paragraph (i)(1) of this section. The levels at the beginning of transport must not exceed the levels prescribed in paragraph (i)(1) of this section;

(j) External radiation levels around the package and around the vehicle, if applicable, will not exceed the limits specified in § 71.47 at any time during transportation; and

(k) Accessible package surface

temperatures will not exceed the limits specified in § 71.43(g) at any time during transportation.

§ 71.84 Air transport of plutonium.

(a) Notwithstanding the provisions of any general licenses and notwithstanding any exemptions stated directly in this part or included indirectly by citation of 49 CFR Chapter I, as may be applicable, the licensee shall assure that plutonium in any form, whether for import, export, or domestic shipment, is not transported by air or delivered to a carrier for air transport unless:

(1) The plutonium is contained in a medical device designed for individual human application; or

(2) The plutonium is contained in a material in which the specific activity is not greater than 0.002 microcuries per gram of material and in which the radioactivity is essentially uniformly distributed; or

(3) The plutonium is shipped in a single package containing no more than an A_2 quantity of plutonium in any isotope or form and is shipped in accordance with § 71.5 of this part; or

(4) The plutonium is shipped in a package specifically authorized for the shipment of plutonium by air in the Certificate of Compliance for that package issued by the Commission.

(b) Nothing in paragraph (a) of this section is to be interpreted as removing or diminishing the requirements of § 73.24 of this chapter.

(c) There have been two orders issued by the NRC restricting the air shipment of plutonium in accordance with Pub. L. 94-79. The first order, issued on August 15, 1975 was superseded by the second order dated September 1, 1978, which has remained in effect since that time. As of the effective date of this rule, the outstanding order dated September 1, 1978 is revoked.

§ 71.89 Opening instructions.

Prior to delivery of a package to a carrier for transport, the licensee shall ensure that any special instructions needed to safely open the package have been sent to or otherwise made available to the consignee for the consignee's use in accordance with § 20.205 of this chapter.

§ 71.91 Records.

(a) Each licensee shall maintain for a period of three years after shipment a record of each shipment of licensed material not exempt under § 71.10, showing, where applicable:

(1) Identification of the packaging by model number.

(2) Verification that there are no significant defects in the packaging, as shipped;

(3) Volume and identification of coolant;

(4) Type and quantity of licensed material in each package, and the total quantity of each shipment;

(5) For each item of irradiated fissile material:

(i) Identification by model number and/or serial number;

(ii) Irradiation and decay history to the extent appropriate to demonstrate that its nuclear and thermal characteristics comply with license conditions; and

(iii) Any abnormal or unusual condition relevant to radiation safety;

(6) Date of the shipment;

(7) For Fissile Class III and for Type B packages, any special controls exercised;

(8) Name and address of the transferee;

(9) Address to which the shipment was made; and

(10) Results of the determinations required by § 71.87 and by the conditions of the package approval.

(b) The licensee shall make available to the Commission for inspection, upon reasonable notice, all records required by this part. Records are valid only if stamped, initialed, or signed and dated by authorized personnel or otherwise authenticated.

(c) Each licensee shall maintain sufficient written records to furnish evidence of the quality of packaging. The records to be maintained include results of the determinations required by § 71.85; design, fabrication, and assembly records; results of reviews, inspections, tests, and audits; results of monitoring work performance and materials analyses; and results of maintenance, modification, and repair activities. Inspection, test, and audit records must identify the inspector or data recorder, the type of observation, the results, the acceptability and the action taken in connection with any deficiencies noted. The records must be retained for three years after the life of the packaging to which they apply.

§ 71.93 Inspection and tests.

(a) The licensee shall permit the Commission at all reasonable times to inspect the licensed material, packaging, premises, and facilities in which the licensed material or packaging is used, produced, tested, stored, or shipped.

(b) The licensee shall perform, and permit the Commission to perform, tests as the Commission deems necessary or appropriate for the administration of the regulations in this chapter.

(c) The licensee shall notify the Regional Administrator of the appropriate Nuclear Regulatory Commission Regional Office listed in Appendix A of Part 73 of this chapter at least 45 days prior to fabrication of a package to be used for the shipment of licensed material having a decay heat load in excess of five kilowatts or with a maximum normal operating pressure in excess of 103 kilopascal (15psi) gauge.

§ 71.95 Reports.

The licensee shall report to the Director, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555, within 30 days:

(a) Any instance in which there is significant reduction in the effectiveness of any authorized packaging during use, and

(b) Details of any defects with safety significance in the packaging after first use, with the means employed to repair the defects and prevent their recurrence.

§ 71.97 Advance notification of shipment of nuclear waste.

(a) Except as specified in paragraph (b) of this section, prior to the transport or delivery to a carrier for transport of licensed material outside the confines of the licensee's plant or other place of use or storage, each licensee shall provide advance notification to the governor of a state, or the governor's designee, of the shipment to, through, or across the boundary of the state.

(b) Advance notification is required only when—

(1) The licensed material is required by this part to be in Type B packaging for transportation;

(2) The licensed material other than irradiated fuel is being transported to, through, or across state boundaries to a disposal site or to a collection point for transport to a disposal site;

(3) The quantity of licensed material in a single package exceeds:

(i) 5,000 curies of special form radionuclides;

(ii) 5,000 curies of uncompressed gases of Argon-41, Krypton-85m, Krypton-87, Xenon-131m, or Xenon-135;

(iii) 50,000 curies of Argon-37, or of uncompressed gases of Krypton-85 or Xenon-133, or of Hydrogen-3 as a gas, as luminous paint, or adsorbed on solid material;

(iv) 20 curies of other non-special form radionuclides for which A_2 is less than or equal to four curies; or

(v) 200 curies of other non-special form radionuclides for which A_2 is greater than four curies; and

(4) The quantity of irradiated fuel is less than that subject to advance notification requirements of 10 CFR Part 73.

(c) *Procedures for submitting advance notification.* (1) The notification must be made in writing to the office of each appropriate governor or governor's designee and to the Regional Administrator of the appropriate Nuclear Regulatory Commission Regional Office listed in Appendix A of Part 73 of this chapter.

(2) A notification delivered by mail must be postmarked at least seven days before the beginning of the seven-day period during which departure of the shipment is estimated to occur.

(3) A notification delivered by messenger must reach the office of the governor or of the governor's designee at least four days before the beginning of the seven-day period during which departure of the shipment is estimated to occur.

(i) A list of the names and mailing addresses of the governors' designees receiving advance notification of transportation of nuclear waste was published in the Federal Register on June 30, 1983 (48 FR 30221).

(ii) The list will be published annually in the Federal Register on or about June 30 to reflect any changes in information.

(iii) A list of the names and mailing addresses of the governors' designees is available upon request from the Director, Office of Governmental and Public Affairs, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

(4) The licensee shall retain a copy of the notification as a record for three years.

(d) *Information to be furnished in advance notification of shipment.* Each advance notification of shipment of nuclear waste must contain the following information:

(1) The name, address, and telephone number of the shipper, carrier, and receiver of the nuclear waste shipment;

(2) A description of the nuclear waste contained in the shipment, as required by the regulations of DOT in 49 CFR 172.202 and 172.203(d);

(3) The point of origin of the shipment and the seven-day period during which departure of the shipment is estimated to occur;

(4) The seven-day period during which arrival of the shipment at state boundaries is estimated to occur;

(5) The destination of the shipment, and the seven-day period during which arrival of the shipment is estimated to occur; and

(6) A point of contact with a telephone number for current shipment information.

(e) *Revision notice.* A licensee who finds that schedule information previously furnished to a governor or governor's designee in accordance with this section will not be met, shall telephone a responsible individual in the office of the governor of the State or of the governor's designee and inform that individual of the extent of the delay beyond the schedule originally reported. The licensee shall maintain a record of the name of the individual contacted for three years.

(f) *Cancellation notice.* (1) Each licensee who cancels a nuclear waste shipment for which advance notification has been sent, shall send a cancellation notice to the governor of each state or the governor's designee previously notified and to the Regional Administrator of the appropriate Nuclear Regulatory Commission Regional Office listed in Appendix A of Part 73 of this chapter.

(2) The licensee shall state in the notice that it is a cancellation and shall identify the advance notification which is being cancelled. The licensee shall retain a copy of the notice as a record for three years.

§ 71.99 Violations.

An injunction or other court order may be obtained prohibiting any violation of any provision of the Atomic Energy Act of 1954, as amended, (the Act) or Title II of the Energy Reorganization Act of 1974, as amended, or any regulation or order issued under the Acts. A court order may be obtained for the payment of a civil penalty imposed under section 234 of the Act for violation of sections 53, 57, 62, 63, 81, 82, 101, 103, 104, 107, or 109 of the Act, or section 206 of the Energy Reorganization Act of 1974, as amended, or any rule, regulation, or order issued under the Acts, or any term, condition, or limitation of any license issued under the Acts, or for any violation for which a license may be revoked under section 186 of the Act. Any person who willfully violates any provision of the Act or any regulation or order issued under the Acts may be guilty of a crime and, upon conviction, may be punished by fine or imprisonment or both, as provided by law.

Subpart H—Quality Assurance

§ 71.101 Quality assurance requirements.

(a) *Purpose.* This subpart describes quality assurance requirements applying to design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, and modification of components of packaging which are important to safety. As used in this subpart, "quality assurance" comprises all those planned and systematic actions necessary to provide adequate confidence that a system or component will perform satisfactorily in service. Quality assurance includes quality control, which comprises those quality assurance actions related to control of the physical characteristics and quality of the material or component to predetermined requirements.

(b) Each licensee shall establish, maintain, and execute a quality assurance program satisfying each of the applicable criteria of §§ 71.101 through 71.137 of this subpart and satisfying any specific provisions that are applicable to the licensee's activities, including procurement of packaging. The licensee shall apply each of the applicable criteria in a graded approach, i.e., to an extent that is consistent with its importance to safety.

(c) *Approval of program.* Prior to the use of any package for the shipment of licensed material subject to this subpart, each licensee shall obtain Commission approval of its quality assurance program. Each licensee shall file a description of its quality assurance program, including a discussion of which requirements of this subpart are applicable and how they will be satisfied, with the Director, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

(d) *Existing package designs.* The provisions of this paragraph deal with packages which have been approved for use in accordance with this part prior to January 1, 1979, and which have been designed in accordance with the provisions of this part in effect at the time of application for package approval. Those packages will be accepted as having been designed in accordance with a quality assurance program which satisfies the provisions of paragraph (b) of this section.

(e) *Existing packages.* The provisions of this paragraph deal with packages which have been approved for use in accordance with this part prior to January 1, 1979, have been at least partially fabricated prior to that date, and for which the fabrication is in accordance with the provisions of this part in effect at the time of application for approval of package design. These packages will be accepted as having been fabricated and assembled in accordance with a quality assurance program which satisfies the provisions of paragraph (b) of this section.

(f) *Previously approved programs.* A Commission-approved quality assurance program which satisfies the applicable criteria of Appendix B of Part 50 of this chapter and which is established, maintained, and executed with regard to transport packages will be accepted as satisfying the requirements of paragraph (b) of this section. Prior to first use, the licensee shall notify the Director, Office of Nuclear Material Safety and

Safeguards, U.S. Nuclear Regulatory Commisson, Washington, DC 20555, of its intent to apply its previously approved Appendix B program to transportation activities. The licensee shall identify the program by date of submittal to the Commission, Docket Number, and date of Commission approval.

§ 71.103 Quality assurance organization.

The licensee[1] shall be responsible for the establishment and execution of the quality assurance program. The licensee may delegate to others, such as contractors, agents, or consultants, the work of establishing and executing the quality assurance program, or any part of the quality assurance program, but shall retain responsibility for the program. The licensee shall clearly establish and delineate in writing the authority and duties of persons and organizations performing activities affecting the safety-related functions of structures, systems and components. These activities include performing the functions associated with attaining quality objectives and the quality assurance functions. The quality assurance functions are (a) assuring that an appropriate quality assurance program is established and effectively executed and (b) verifying, by procedures such as checking, auditing, and inspection, that activities affecting the safety-related functions have been correctly performed. The persons and organizations performing quality assurance functions must have sufficient authority and organizational freedom to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions. The persons and organizations performing quality assurance functions shall report to a management level which assures that the required authority and organizational freedom, including sufficient independence from cost and schedule when opposed to safety considerations, are provided. Because of the many variables involved, such as the number of personnel, the type of activity being performed, and the location or locations where activities are performed, the organizational structure for executing the quality assurance program may take various forms provided that the persons and organizations assigned the quality assurance functions have the required authority and organizational freedom. Irrespective of the organizational structure, the individual(s) assigned the responsibility for assuring effective execution of any portion of the quality assurance program at any location where activities subject to this section are being performed must have direct access to the levels of management necessary to perform this function.

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[1] While the term "licensee" is used in these criteria, the requirements are applicable to whatever design, fabrication, assembly, and testing of the package is accomplished with respect to a package prior to the time a package approval is issued.

§ 71.105 Quality assurance program.

(a) The licensee shall establish, at the earliest practicable time, consistent with the schedule for accomplishing the activities, a quality assurance program that complies with the requirements of §§ 71.101 through 71.137 of this subpart. The licensee shall document the quality assurance program by written procedures or instructions and shall carry out the program in accordance with those procedures throughout the period during which packaging is used. The licensee shall identify the material and components to be covered by the quality assurance program, the major organizations participating in the program, and the designated functions of these organizations.

53 FR 19748

(b) The licensee, through its quality assurance program, shall provide control over activities affecting the quality of the identified materials and components to an extent consistent with their importance to safety, and as necessary to assure conformance to the approved design of each individual package used for the shipment of radioactive material. The licensee shall assure that activities affecting quality are accomplished under suitably controlled conditions. Controlled conditions include the use of appropriate equipment; suitable environmental conditions for accomplishing the activity, such as adequate cleanliness; and assurance that all prerequisites for the given activity have been satisfied. The licensee shall take into account the need for special controls, processes, test equipment, tools and skills to attain the required quality, and the need for verification of quality by inspection and test.

(c) The licensee shall base the requirements and procedures of its quality assurance program on the following considerations concerning the complexity and proposed use of the package and its components:

(1) The impact of malfunction or failure of the item to safety;

(2) The design and fabrication complexity or uniqueness of the item;

(3) The need for special controls and surveillance over processes and equipment;

(4) The degree to which functional compliance can be demonstrated by inspection or test, and

(5) The quality history and degree of standardization of the item.

(d) The licensee shall provide for indoctrination and training of personnel performing activities affecting quality as necessary to assure that suitable proficiency is achieved and maintained. The licensee shall review the status and adequacy of the quality assurance program at established intervals. Management of other organizations participating in the quality assurance program shall regularly review the status and adequacy of that part of the quality assurance program which they are executing.

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§ 71.107 Package design control.

(a) The licensee shall establish measures to assure that applicable regulatory requirements and the package design, as specified in the license for those materials and components to which this section applies, are correctly translated into specifications, drawings, procedures, and instructions. These measures must include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from standards are controlled. Measures must be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the materials, parts, and components of the packaging.

48 FR 35949

(b) The licensee shall establish measures for the identification and control of design interfaces and for coordination among participating design organizations. These measures must include the establishment of written procedures among participating design organizations for the review, approval, release, distribution, and revision of documents involving design interfaces. The design control measures must provide for verifying or checking the adequacy of design, by methods such as design reviews, alternate or simplified calculational methods, or by a suitable testing program. For the verifying or checking process, the licensee shall designate individuals or groups other than those who were responsible for the original design, but who may be from the same organization. Where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, the licensee shall include suitable qualification testing of a prototype or sample unit under the most adverse design conditions. The licensee shall apply design control measures to items such as the following: criticality physics, radiation shielding, stress, thermal, hydraulic, and accident analyses; compatibility of materials; accessibility for inservice inspection, maintenance, and repair; features to facilitate decontamination; and delineation of acceptance criteria for inspections tests.

(c) The licensee shall subject design changes, including field changes, to design control measures commensurate with those applied to the original design. Changes in the conditions specified in the package approval require NRC approval.

§ 71.109 Procurement document control.

The licensee shall establish measures to assure adequate quality is required in the documents for procurement of material, equipment, and services, whether purchased by the licensee or by its contractors or subcontractors. To the extent necessary, the licensee shall require contractors or subcontractors to provide a quality assurance program consistent with the applicable provisions of this part.

§ 71.111 Instructions, procedures, and drawings.

The licensee shall prescribe activities affecting quality by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall require that these instructions, procedures, and drawings be followed. The instructions, procedures, and drawings must include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

§ 71.113 Document control.

The licensee shall establish measures to control the issuance of documents such as instructions, procedures, and drawings, including changes, which prescribe all activities affecting quality. These measures must assure that documents, including changes, are reviewed for adequacy, approved for release by authorized personnel, and distributed and used at the location where the prescribed activity is performed. These measures must assure that changes to documents are reviewed and approved.

§ 71.115 Control of purchased material, equipment, and services.

(a) The licensee shall establish measures to assure that purchased material, equipment, and services, whether purchased directly or through contractors and subcontractors, conform to the procurement documents. These measures must include provisions, as appropriate, for source evaluation and selection, objective evidence of quality furnished by the contractor or subcontractor, inspection at the contractor or subcontractor source, and examination of products upon delivery.

(b) The licensee shall have available documentary evidence that material and equipment conform to the procurement specifications prior to installation or use of the material and equipment. The licensee shall retain or have available this documentary evidence for the life of the package to which it applies. The licensee shall assure that the evidence is sufficient to identify the specific requirements met by the purchased material and equipment.

(c) The licensee or designee shall assess the effectiveness of the control of quality by contractors and subcontractors at intervals consistent with the importance, complexity, and quantity of the product or services.

§ 71.117 Identification and control of materials, parts, and components.

The licensee shall establish measures for the identification and control of materials, parts, and components. These measures must assure that identification of the item is maintained by heat number, part number, or other appropriate means, either on the item or on records traceable to the item, as required throughout fabrication, installation, and use of the item. These identification and control measures must be designed to prevent the use of incorrect or defective materials, parts, and components.

§ 71.119 Control of special processes.

The licensee shall establish measures to assure that special processes, including welding, heat treating, and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

§ 71.121 Internal inspection.

The licensee shall establish and execute a program for inspection of activities affecting quality by or for the organization performing the activity to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. The inspection must be performed by individuals other than those who performed the activity being inspected. Examination, measurements, or tests of material or products processed must be performed for each work operation where necessary to assure quality. If direct inspection of processed material or products is not carried out, indirect control by monitoring processing methods, equipment, and personnel must be provided. Both inspection and process monitoring must be provided when quality control is inadequate without both. If mandatory inspection hold points, which require witnessing or inspecting by the licensee's designated representative and beyond which work should not proceed without the consent of its designated representative, are required, the specific hold points must be indicated in appropriate documents.

§ 71.123 Test control.

The licensee shall establish a test program to assure that all testing required to demonstrate that the packaging components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements of this part and the requirements and acceptance limits contained in the package approval. The test procedures must include provisions for assuring that all prerequisites for the given test are met, that adequate test instrumentation is available and used, and that the test is performed under suitable environmental conditions. The licensee shall document and evaluate the test results to assure that test requirements have been satisfied.

§ 71.125 Control of measuring and test equipment.

The licensee shall establish measures to assure that tools, gages, instruments, and other measuring and testing devices used in activities affecting quality are properly controlled, calibrated, and adjusted at specified times to maintain accuracy within necessary limits.

§ 71.127 Handling, storage, and shipping control.

The licensee shall establish measures to control, in accordance with instructions, the handling, storage, shipping, cleaning, and preservation of materials and equipment to be used in packaging to prevent damage or deterioration. When necessary for particular products, special protective environments, such as inert gas atmosphere, and specific moisture content and temperature levels must be specified and provided.

§ 71.129 Inspection, test, and operating status.

(a) The licensee shall establish measures to indicate, by the use of markings such as stamps, tags, labels, routing cards, or other suitable means, the status of inspections and tests performed upon individual items of the packaging. These measures must provide for the identification of items which have satisfactorily passed required inspections and tests where necessary to preclude inadvertent bypassing of the inspections and tests.

(b) The licensee shall establish measures to identify the operating status of components of the packaging, such as tagging valves and switches, to prevent inadvertent operation.

§ 71.131 Nonconforming materials, parts, or components.

The licensee shall establish measures to control materials, parts, or components which do not conform to the licensee's requirements in order to prevent their inadvertent use or installation. These measures must include, as appropriate, procedures for identification, documentation, segregation, disposition, and notification to affected organizations. Nonconforming items must be reviewed and accepted, rejected, repaired, or reworked in accordance with documented procedures.

§ 71.133 Corrective action.

The licensee shall establish measures to assure that conditions adverse to quality, such as deficiencies, deviations, defective material and equipment, and nonconformances, are promptly identified and corrected. In the case of a significant condition adverse to quality, the measures must assure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken must be documented and reported to appropriate levels of management.

§ 71.135 Quality assurance records.

The licensee shall maintain sufficient written records to describe the activities affecting quality. The records must include the instructions, procedures, and drawings required by § 71.111 to prescribe quality assurance activities and must include closely related specifications such as required qualifications of personnel, procedures, and equipment. The records must include the instructions or procedures which establish a records retention program that is consistent with applicable regulations and designates factors such as duration, location, and assigned responsibility. The licensee shall retain these records for three years beyond the date when the licensee last engages in the activity for which the quality assurance program was developed. If any portion of the written procedures or instructions is superseded, the licensee shall retain the superseded material for three years after it is superseded.

§ 71.137 Audits.

The licensee shall carry out a comprehensive system of planned and periodic audits to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program. The audits must be performed in accordance with written procedures or checklists by appropriately trained personnel not having direct responsibilities in the areas being audited. Audited results must be documented and reviewed by management having responsibility in the area audited. Follow-up action, including re-audit of deficient areas, must be taken where indicated.

Appendix A—Determination of A_1 and A_2

I. Single radionuclides.

(1) For a single radionuclide of known identity, the values of A_1 and A_2 are taken from Table A-1 if listed there. The values A_1 and A_2 in Table A-1 are also applicable for radionuclides contained in (α, n) or (γ, n) neutron sources.

(2) For any single radionuclide whose identity is known but which is not listed in Table A-1, the values of A_1 and A_2 are determined according to the following procedure:

(a) If the radionuclide emits only one type of radiation, A_1 is determined according to the rules in paragraphs (i), (ii), (iii) and (iv) of this paragraph. For radionuclides emitting different kinds of radiation, A_1 is the most restrictive value of those determined for each kind of radiation. However, in both cases, A_1 is restricted to a maximum of 1000 Ci. If a parent nuclide decays into a shorter lived daughter with a half-life not greater than 10 days, A_1 is calculated for both the parent and the daughter, and the more limiting of the two values is assigned to the parent nuclide.

(i) For gamma emitters, A_1 is determined by the expression:

$$A_1 = \frac{9}{\Gamma} \text{ curies}$$

where Γ is the gamma-ray constant corresponding to the dose in R/h at 1 m per Ci; the number 9 results from the choice of 1 rem/h at a distance of 3 m as the reference dose-equivalent rate.

(ii) For X-ray emitters, A_1 is determined by the atomic number of the nuclide:

for $Z < 55$—$A_1 = 1000$ Ci
for $Z > 55$—$A_1 = 200$ Ci
where Z is the atomic number of the nuclide.

(iii) For beta emitters, A_1 is determined by the maximum beta energy (E_{max}) according to Table A-2.

(iv) For alpha emitters, A_1 is determined by the expression:

$A_1 = 1000\ A_2$

where A_2 is the value listed in Table A-3.

(b) A_2 is the more restrictive of the following two values:

(i) The corresponding A_1; and

(ii) The value A_2 obtained from Table A-3.

(3) For any single radionuclide whose identity is unknown, the value of A_1 is taken to be two Ci and the value of A_2 is taken to be 0.002 Ci. However, if the atomic number of the radionuclide is known to be less than 82, the value of A_1 is taken to be 10 Ci and the value of A_2 is taken to be 0.4 Ci.

II. Mixtures of radionuclides, including radioactive decay chains.

(1) For mixed fission products the following activity limits may be assumed if a detailed analysis of the mixture is not carried out:

$A_1 = 10$ Ci
$A_2 = 0.4$ Ci

(2) A single radioactive decay chain is considered to be a single radionuclide when the radionuclides are present in their naturally occurring proportions and no daughter nuclide has a half-life either longer than 10 days or longer than that of the parent nuclide. The activity to be taken into account and the A_1 or A_2 value from Table A-1 to be applied are those corresponding to the parent nuclide of that chain. When calculating A_1 or A_2 values, radiation emitted by daughters must be considered. However, in the case of radioactive decay chains in which any daughter nuclide has a half-life either longer than 10 days or greater than that of the parent nuclide, the parent and daughter nuclides are considered to be mixtures of different nuclides.

(3) In the case of a mixture of different radionuclides, where the identity and activity of each radionuclide are known, the permissible activity of each radionuclide $R_1, R_2 \ldots R_n$ is such that $F_1 + F_2 + \ldots F_n$ is not greater than unity, where

$$F_1 = \frac{\text{Total activity of } R_1}{A_i(R_1)}$$

$$F_2 = \frac{\text{Total activity of } R_2}{A_i(R_2)}$$

$$F_n = \frac{\text{Total activity of } R_n}{A_i(R_n)}$$

$A_i(R_1, R_2 \ldots R_n)$ is the value of A_1 or A_2 as appropriate for the nuclide $R_1, R_2 \ldots R_n$.

(4) When the identity of each radionuclide is known but the individual activities of some of the radionuclides are not known, the formula given in paragraph (3) is applied to establish the values of A_1 or A_2 as appropriate. All the radionuclides whose individual activities are not known (their total activity will, however, be known) are classed in a single group and the most restrictive value of A_1 and A_2 applicable to any one of them is used as the value of A_1 or A_2 in the denominator of the fraction.

(5) Where the identity of each radionuclide is known but the individual activity of none of the radionuclides is known, the most restrictive value of A_1 or A_2 applicable to any one of the radionuclides present is adopted as the applicable value.

(6) When the identity of none of the nuclides is known, the value of A_1 is taken to be two Ci and the value of A_2 is taken to be 0.002 Ci. However, if alpha emitters are known to be absent, the value of A_2 is taken to be 0.4 Ci.

TABLE A-1.—A_1 AND A_2 VALUES FOR RADIONUCLIDES

(See footnotes at end of table)

Symbol of radionuclide	Element and atomic number	A_1 (Ci)	A_2 (Ci)	Specific activity (Ci/g)
227_{Ac}	Actinium (89)	1000	0.003	7.2×10
228_{Ac}		10	4	2.2×10^6
105_{Ag}	Silver (47)	40	40	3.1×10^4
$110m_{Ag}$		7	7	4.7×10^3
111_{Ag}		100	20	1.6×10^5
241_{Am}	Americium (95)	8	0.008	3.2
243_{Am}		8	0.008	1.9×10^{-1}
37_{Ar} (compressed or uncompressed)*	Argon (18)	1000	1000	1.0×10^5
41_{Ar} (uncompressed)*		20	20	4.3×10^7
41_{Ar} (compressed)*		1	1	4.3×10^7
73_{As}	Arsenic (33)	1000	400	2.4×10^4
74_{As}		20	20	1.0×10^5
76_{As}		10	10	1.6×10^6
77_{As}		300	20	1.1×10^6
211_{At}	Astatine (85)	200	7	2.1×10^6
193_{Au}	Gold (79)	200	200	9.3×10^5
196_{Au}		30	30	1.2×10^5
198_{Au}		40	20	2.4×10^5
199_{Au}		200	25	2.1×10^5
131_{Ba}	Barium (56)	40	40	8.7×10^4
133_{Ba}		40	10	4.0×10^2
140_{Ba}		20	20	7.3×10^4
7_{Be}	Beryllium (4)	300	300	3.5×10^5
206_{Bi}	Bismuth (83)	5	5	9.9×10^4
207_{Bi}		10	25	2.2×10^1
210_{Bi} (RaE)		100	4	1.2×10^5
212_{Bi}		6	6	1.5×10^7
249_{Bk}	Berkelium (97)	1000	1	1.8×10^3
77_{Br}	Bromine (35)	70	25	7.1×10^5

TABLE A-1.—A_1 AND A_2 VALUES FOR RADIONUCLIDES—Continued

(See footnotes at end of table)

Symbol of radionuclide	Element and atomic number	A_1(Ci)	A_2(Ci)	Specific activity (Ci/g)
^{82}Br		6	6	1.1×10^{6}
^{11}C	Carbon (6)	20	20	8.4×10^{8}
^{14}C		1000	60	4.6
^{45}Ca	Calcium (20)	1000	25	1.9×10^{4}
^{47}Ca		20	20	5.9×10^{5}
^{109}Cd	Cadmium (48)	1000	70	2.6×10^{3}
^{115m}Cd		30	30	2.6×10^{4}
^{115}Cd		80	20	5.1×10^{5}
^{139}Ce	Cerium (58)	100	100	6.5×10^{3}
^{141}Ce		300	25	2.8×10^{4}
^{143}Ce		60	20	6.6×10^{5}
^{144}Ce		10	7	3.2×10^{3}
^{249}Cf	Californium (98)	2	0.002	3.1
^{250}Cf		7	0.007	1.3×10^{2}
^{252}Cf		2	0.009	6.5×10^{2}
^{36}Cl	Chlorine (17)	300	10	3.2×10^{-2}
^{38}Cl		10	10	1.3×10^{8}
^{242}Cm	Curium (96)	200	0.2	3.3×10^{3}
^{243}Cm		9	0.009	4.2×10
^{244}Cm		10	0.01	8.2×10
^{245}Cm		6	0.006	1.0×10^{-1}
^{246}Cm		6	0.006	3.5×10^{-1}
^{56}Co	Cobalt (27)	5	5	3.0×10^{4}
^{57}Co		90	90	8.5×10^{3}
^{58m}Co		1000	1000	5.9×10^{6}
^{58}Co		20	20	3.1×10^{4}
^{60}Co		7	7	1.1×10^{3}
^{51}Cr	Chromium (24)	600	600	9.2×10^{4}
^{129}Cs	Cesium (55)	40	40	7.6×10^{5}
^{131}Cs		1000	1000	1.0×10^{5}
^{134m}Cs		1000	10	7.4×10^{6}
^{134}Cs		10	10	1.2×10^{3}
^{135}Cs		1000	25	8.8×10^{-4}
^{136}Cs		7	7	7.4×10^{4}
^{137}Cs		30	10	9.8×10
^{64}Cu	Copper (29)	80	25	3.8×10^{6}
^{67}Cu		200	25	7.9×10^{5}
^{165}Dy	Dysprosium (66)	100	20	8.2×10^{6}
^{166}Dy		1000	200	2.3×10^{5}
^{169}Er	Erbium (68)	1000	25	8.2×10^{4}
^{171}Er		50	20	2.4×10^{6}
^{152m}Eu	Europium (63)	30	30	2.2×10^{6}
^{152}Eu		20	10	1.9×10^{2}
^{154}Eu		10	5	1.5×10^{2}
^{155}Eu		400	60	1.4×10^{3}
^{18}F	Fluorine (9)	20	20	9.3×10^{7}
^{52}Fe	Iron (26)	5	5	7.3×10^{6}
^{55}Fe		1000	1000	2.2×10^{3}
^{59}Fe		10	10	4.9×10^{4}
^{67}Ga	Gallium (31)	100	100	6.0×10^{5}
^{68}Ga		20	20	4.0×10^{7}
^{72}Ga		7	7	3.1×10^{6}
^{153}Gd	Gadolinium (64)	200	100	3.6×10^{3}
^{159}Gd		300	20	1.1×10^{6}
^{68}Ge	Germanium (32)	20	10	7.0×10^{3}
^{71}Ge		1000	1000	1.6×10^{5}
^{3}H	Hydrogen (1) see T-Tritium			
^{181}Hf	Hafnium (72)	30	25	1.6×10^{4}
^{197m}Hg	Mercury (80)	200	200	6.6×10^{5}
^{197}Hg		200	200	2.5×10^{5}
^{203}Hg		80	25	1.4×10^{4}
^{166}Ho	Holmium (67)	30	30	6.9×10^{5}
^{123}I	Iodine (53)	60	60	1.9×10^{6}
^{125}I		1000	70	1.7×10^{4}
^{126}I		40	10	7.8×10^{4}
^{129}I		1000	2	1.6×10^{-4}
^{131}I		40	10	1.2×10^{5}
^{132}I		7	7	1.1×10^{7}

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TABLE A-1.—A_1 AND A_2 VALUES FOR RADIONUCLIDES—Continued

(See footnotes at end of table)

Symbol of radionuclide	Element and atomic number	A_1(Ci)	A_2(Ci)	Specific activity (Ci/g)
^{133}I		30	10	1.1×10^{6}
^{134}I		8	8	2.7×10^{7}
^{135}I		10	10	3.5×10^{6}
^{111}In	Indium (49)	30	25	4.2×10^{5}
^{113m}In		60	60	1.6×10^{7}
^{114m}In		30	20	2.3×10^{4}
^{115m}In		100	20	6.1×10^{6}
^{190}Ir	Iridium (77)	10	10	6.2×10^{4}
^{192}Ir		20	10	9.1×10^{3}
^{194}Ir		10	10	8.5×10^{5}
^{42}K	Potassium (19)	10	10	6.0×10^{6}
^{43}K		20	10	3.3×10^{6}
^{85m}Kr (uncompressed)*	Krypton (36)	100	100	8.4×10^{6}
^{85m}Kr (compressed)*		3	3	8.4×10^{6}
^{85}Kr (uncompressed)*		1000	1000	4.0×10^{2}
^{85}Kr (compressed)*		5	5	4.0×10^{2}
^{87}Kr (uncompressed)*		20	20	2.8×10^{7}
^{87}Kr (compressed)*		0.6	0.6	2.8×10^{7}
^{140}La	Lanthanum (57)	30	30	5.6×10^{5}
LSA	Low specific activity material—see § 71.4			
^{177}Lu	Lutetium (71)	300	25	1.1×10^{5}
MFP	Mixed fission products	10	0.4	
^{28}Mg	Magnesium (12)	6	6	5.2×10^{6}
^{52}Mn	Manganese (25)	5	5	4.4×10^{5}
^{54}Mn		20	20	8.3×10^{3}
^{56}Mn		5	5	2.2×10^{7}
^{99}Mo	Molybdenum (42)	100	20	4.7×10^{5}
^{13}N	Nitrogen (7)	20	10	1.5×10^{9}
^{22}Na	Sodium (11)	5	5	6.3×10^{3}
^{24}Na		5	5	8.7×10^{6}
^{93m}Nb	Niobium (41)	1000	200	1.1×10^{3}
^{95}Nb		20	20	3.9×10^{4}
^{97}Nb		20	20	2.6×10^{7}
^{147}Nd	Neodymium (60)	100	20	8.0×10^{4}
^{149}Nd		30	20	1.1×10^{7}
^{59}Ni	Nickel (28)	1000	900	8.1×10^{-2}
^{63}Ni		1000	100	4.6×10
^{65}Ni		10	10	1.9×10^{7}
^{237}Np	Neptunium (93)	5	0.005	6.9×10^{-4}
^{239}Np		200	25	2.3×10^{5}
^{185}Os	Osmium (76)	20	20	7.3×10^{3}
^{191}Os		600	200	4.6×10^{4}
^{191m}Os		200	200	1.2×10^{6}
^{193}Os		100	20	5.3×10^{5}
^{32}P	Phosphorus (15)	30	30	2.9×10^{5}
^{230}Pa	Protactinium (91)	20	0.8	3.2×10^{4}
^{231}Pa		2	0.002	4.5×10^{-2}
^{233}Pa		100	100	2.1×10^{4}
^{201}Pb	Lead (82)	20	20	1.7×10^{6}
^{210}Pb		100	0.2	8.8×10
^{212}Pb		6	5	1.4×10^{6}
^{103}Pd	Palladium (46)	1000	700	7.5×10^{4}
^{109}Pd		100	20	2.1×10^{6}
^{147}Pm	Promethium (61)	1000	25	9.4×10^{2}
^{149}Pm		100	20	4.2×10^{5}
^{210}Po	Polonium (84)	200	0.2	4.5×10^{3}
^{142}Pr	Praseodymium (59)	10	10	1.2×10^{6}
^{143}Pr		300	20	6.6×10^{4}
^{191}Pt	Platinum (78)	100	100	2.3×10^{5}
^{193m}Pt		200	200	2.0×10^{5}
^{197m}Pt		300	20	1.2×10^{7}
^{197}Pt		300	20	8.8×10^{5}
^{238}Pu	Plutonium (94)	3	0.003	1.7×10
^{239}Pu		2	0.002	6.2×10^{-2}
^{240}Pu		2	0.002	2.3×10^{-1}
^{241}Pu		1000	0.1	1.1×10^{2}
^{242}Pu		3	0.003	3.9×10^{-3}
^{223}Ra	Radium (88)	50	0.2	5.0×10^{4}

TABLE A-1.—A_1 AND A_2 VALUES FOR RADIONUCLIDES—Continued

(See footnotes at end of table)

Symbol of radionuclide	Element and atomic number	A_1(Ci)	A_2(Ci)	Specific activity (Ci/g)
^{224}Ra		6	0.5	1.6×10^5
^{226}Ra		10	0.05	1.0
^{228}Ra		10	0.05	2.7×10^2
^{81}Rb	Rubidium (37)	30	25	8.2×10^6
^{86}Rb		30	30	8.1×10^4
^{87}Rb		Unlimited.	Unlimited.	6.6×10^{-8}
Rb (natural)		Unlimited.	Unlimited.	1.8×10^{-8}
^{186}Re	Rhenium (75)	100	20	1.9×10^5
^{187}Re		Unlimited.	Unlimited.	3.8×10^{-8}
^{188}Re		10	10	1.0×10^7
Re (natural)		Unlimited.	Unlimited.	2.4×10^{-8}
^{103m}Rh	Rhodium (45)	1000	1000	3.2×10^7
^{105}Rh		200	25	8.2×10^5
^{222}Rn	Radon (86)	10	2	1.5×10^5
^{97}Ru	Ruthenium (44)	80	80	5.5×10^6
^{103}Ru		30	25	3.2×10^4
^{105}Ru		20	20	6.6×10^6
^{106}Ru		10	7	3.4×10^3
^{35}S	Sulphur (16)	1000	60	4.3×10^4
^{122}Sb	Antimony (51)	30	30	3.9×10^5
^{124}Sb		5	5	1.8×10^4
^{125}Sb		40	25	1.4×10^3
^{46}Sc	Scandium (21)	8	8	3.4×10^4
^{47}Sc		200	20	8.2×10^5
^{48}Sc		5	5	1.5×10^6
^{75}Se	Selenium (34)	40	40	1.4×10^4
^{31}Si	Silicon (14)	100	20	3.9×10^7
^{147}Sm	Samarium (62)	Unlimited.	Unlimited.	2.0×10^{-8}
^{151}Sm		1000	90	2.6×10
^{153}Sm		300	20	4.4×10^5
^{113}Sn	Tin (50)	60	60	1.0×10^4
^{119m}Sn		100	100	4.4×10^3
^{125}Sn		10	10	1.1×10^5
^{85m}Sr	Strontium (38)	80	80	3.2×10^7
^{85}Sr		30	30	2.4×10^4
^{87m}Sr		50	50	1.2×10^7
^{89}Sr		100	10	2.9×10^4
^{90}Sr		10	0.4	1.5×10^2
^{91}Sr		10	10	3.6×10^6
^{92}Sr		10	10	1.3×10^7
T (uncompressed)*	Tritium (1)	1000	1000	9.7×10^3
T (compressed)*		1000	1000	9.7×10^3
T (activated luminous paint)		1000	1000	9.7×10^3
T (adsorbed on solid carrier)		1000	1000	9.7×10^3
T (tritiated water)		1000	1000	9.7×10^3
T (other forms)		20	20	9.7×10^3
^{182}Ta	Tantalum (73)	20	20	6.2×10^3
^{160}Tb	Terbium (65)	20	10	1.1×10^4
^{96m}Tc	Technetium (43)	1000	1000	3.8×10^7
^{96}Tc		6	6	3.2×10^5
^{97m}Tc		1000	200	1.5×10^4
^{97}Tc		1000	400	1.4×10^{-3}
^{99m}Tc		100	100	5.2×10^6
^{99}Tc		1000	25	1.7×10^{-2}
^{125m}Te	Tellurium (52)	1000	100	1.8×10^4
^{127m}Te		300	20	4.0×10^3
^{127}Te		300	20	2.6×10^6
^{129m}Te		30	10	2.5×10^4
^{129}Te		100	20	2.0×10^7
^{131m}Te		10	10	8.0×10^5
^{132}Te		7	7	3.1×10^5
^{227}Th	Thorium (90)	200	0.2	3.2×10^4
^{228}Th		6	0.008	8.3×10^2
^{230}Th		3	0.003	1.9×10^{-2}
^{231}Th		1000	25	5.3×10^5
^{232}Th		Unlimited.	Unlimited.	1.1×10^{-7}
^{234}Th		10	10	2.3×10^4
Th (natural)		Unlimited.	Unlimited.	2.2×10^{-7}

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TABLE A-1.—A_1 AND A_2 VALUES FOR RADIONUCLIDES—Continued

(See footnotes at end of table)

Symbol of radionuclide	Element and atomic number	A_1(Ci)	A_2(Ci)	Specific activity (Ci/g)
$_{\mathrm{Th}}$ (irradiated)**				
200$_{\mathrm{Tl}}$	Thallium (81)	20	20	5.8×10^{5}
201$_{\mathrm{Tl}}$		200	200	2.2×10^{5}
202$_{\mathrm{Tl}}$		40	40	5.4×10^{4}
204$_{\mathrm{Tl}}$		300	10	4.3×10^{2}
170$_{\mathrm{Tm}}$	Thulium (69)	300	10	6.0×10^{3}
171$_{\mathrm{Tm}}$		1000	100	1.1×10^{3}
230$_{\mathrm{U}}$	Uranium (92)	100	0.1	2.7×10^{4}
232$_{\mathrm{U}}$		30	0.03	2.1×10
233$_{\mathrm{U}}$		100	0.1	9.5×10^{-3}
234$_{\mathrm{U}}$		100	0.1	6.2×10^{-3}
235$_{\mathrm{U}}$		100	0.2	2.1×10^{-6}
236$_{\mathrm{U}}$		200	0.2	6.3×10^{-5}
238$_{\mathrm{U}}$		Unlimited.	Unlimited.	3.3×10^{-7}
$_{\mathrm{U}}$ (natural)		Unlimited.	Unlimited.	(SEE TABLE A-4)
$_{\mathrm{U}}$ (enriched) <20%		Unlimited.	Unlimited.	(SEE TABLE A-4)
20% or greater		100	0.1	(SEE TABLE A-4)
$_{\mathrm{U}}$ (depleted)		Unlimited.	Unlimited.	(SEE TABLE A-4)
$_{\mathrm{U}}$ (irradiated)***				
48$_{\mathrm{V}}$	Vanadium (23)	6	6	1.7×10^{5}
181$_{\mathrm{W}}$	Tungsten (74)	200	100	5.0×10^{3}
185$_{\mathrm{W}}$		1000	25	9.7×10^{3}
187$_{\mathrm{W}}$		40	20	7.0×10^{5}
127$_{\mathrm{Xe}}$ (uncompressed)*	Xenon (54)	70	70	2.8×10^{4}
127$_{\mathrm{Xe}}$ (compressed)*		5	5	2.8×10^{4}
131m$_{\mathrm{Xe}}$ (compressed)*		10	10	1.0×10^{5}
131m$_{\mathrm{Xe}}$ (uncompressed)*		100	100	1.0×10^{5}
133$_{\mathrm{Xe}}$ (uncompressed)*		1000	1000	1.9×10^{5}
133$_{\mathrm{Xe}}$ (compressed)*		5	5	1.9×10^{5}
135$_{\mathrm{Xe}}$ (uncompressed)*		70	70	2.5×10^{6}
135$_{\mathrm{Xe}}$ (compressed)*		2	2	2.5×10^{6}
87$_{\mathrm{Y}}$	Yttrium (39)	20	20	4.5×10
90$_{\mathrm{Y}}$		10	10	2.5×10^{5}
91m$_{\mathrm{Y}}$		30	30	4.1×10^{7}
91$_{\mathrm{Y}}$		30	30	2.5×10^{4}
92$_{\mathrm{Y}}$		10	10	9.5×10^{6}
93$_{\mathrm{Y}}$		10	10	3.2×10^{6}
169$_{\mathrm{Yb}}$	Ytterbium (70)	80	80	2.3×10^{4}
175$_{\mathrm{Yb}}$		400	25	1.8×10^{5}
65$_{\mathrm{Zn}}$	Zinc (30)	30	30	8.0×10^{3}
69m$_{\mathrm{Zn}}$		40	20	3.3×10^{5}
69$_{\mathrm{Zn}}$		300	20	5.3×10^{7}
93$_{\mathrm{Zr}}$	Zirconium (40)	1000	200	3.5×10^{-3}
95$_{\mathrm{Zr}}$		20	20	2.1×10^{4}
97$_{\mathrm{Zr}}$		20	20	2.0×10^{6}

* For the purpose of Table A-1, compressed gas means a gas at a pressure which exceeds the ambient atmospheric pressure at the location where the containment system was closed.

** The values of A_1 and A_2 must be calculated in accordance with the procedure specified in Appendix A, paragraph II(3), taking into account the activity of the fission products and of the uranium-233 in addition to that of the thorium.

*** The values of A_1 and A_2 must be calculated in accordance with the procedure specified in Appendix A, paragraph II(3), taking into account the activity of the fission products and plutonium isotopes in addition to that of the uranium.

TABLE A-2

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RELATIONSHIP BETWEEN A_1 AND E_{max} FOR BETA EMITTERS

E_{max}(MeV)	A_1(Ci)
< 0.5	1000
0.5 - < 1.0	300
1.0 - < 1.5	100
1.5 - < 2.0	30
≥ 2.0	10

TABLE A-3

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RELATIONSHIP BETWEEN A_3 AND THE ATOMIC NUMBER OF THE RADIONUCLIDE

A_3			
Atomic Number	Half-life less than 1000 days	Half-life 1000 days to 10^6 years	Half-life greater than 10^6 years
1 to 81	3 Ci	.05 Ci	3 Ci
82 and above	.002 Ci	.002 Ci	3 Ci

TABLE A-4.—ACTIVITY-MASS RELATIONSHIPS FOR URANIUM/THORIUM

Thorium and uranium enrichment[1] wt % ^{235}U present	Specific activity	
	Ci/g	g/Ci
0.45	5.0×10^{-7}	2.0×10^{6}
0.72 (natural)	7.06×10^{-7}	1.42×10^{6}
1.0	7.6×10^{-7}	1.3×10^{6}
1.5	1.0×10^{-6}	1.0×10^{6}
5.0	2.7×10^{-6}	3.7×10^{5}
10.0	4.8×10^{-6}	2.1×10^{5}
20.0	1.0×10^{-5}	1.0×10^{5}
35.0	2.0×10^{-5}	5.0×10^{4}
50.0	2.5×10^{-5}	4.0×10^{4}
90.0	5.8×10^{-5}	1.7×10^{4}
93.0	7.0×10^{-5}	1.4×10^{4}
95.0	9.1×10^{-5}	1.1×10^{4}
Natural Thorium	2.2×10^{-7}	4.6×10^{6}

[1] The figures for uranium include representative values for the activity of the uranium-234 which is concentrated during the enrichment process. The activity for Thorium includes the equilibrium concentration of Thorium-228.