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National Spent Nuclear Fuel Program

NSNFP Activities in Support of Repository Licensing for Disposal of DOE SNF

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U.S. Department of Energy
Assistant Secretary for Environmental Management
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ABSTRACT

The U.S. Department of Energy (DOE) Office of Civilian Radioactive Waste Management is in the process of preparing the Yucca Mountain license application for submission to the Nuclear Regulatory Commission as the nation's first geologic repository for spent nuclear fuel (SNF) and high-level waste. Because the DOE SNF will be part of the license application, there are various components of the license application that will require information relative to the DOE SNF. The National Spent Nuclear Fuel Program (NSNFP) is the organization that directs the research, development, and testing of treatment, shipment, and disposal technologies for all DOE SNF. This report documents the work activities conducted by the NSNFP and discusses the relationship between these NSNFP technical activities and the license application. A number of the NSNFP activities were performed to provide risk insights and understanding of DOE SNF disposal as well as to prepare for anticipated questions from the regulatory agency.

NSNFP Activities in Support of Repository Licensing for Disposal of DOE SNF Report Contributors

The information contained in this report was prepared by a number of people working on the U.S. Department of Energy Spent Nuclear Fuel Program. (Listed in alphabetical order.)

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FIGURES

ACRONYMS

ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
CFR	Nuclear Regulatory Commission
DOE	U.S. Department of Energy
DOE sites	Refers to the three sites indicated in the DOE Record of Decision to regionalize spent nuclear fuel management: Hanford Site, Idaho National Engineering and Environmental Laboratory (previously Idaho National Engineering Laboratory), and Savannah River Site
EM	Office of Environmental Management
FEP	features, events, and processes
HIC	high-integrity can
HLW	high-level waste
INEEL	Idaho National Engineering and Environmental Laboratory
M&O	management and operating (contractor)
MCO	multi-canister overpack
MGR	monitored geologic repository
NRC	Nuclear Regulatory Commission
NSNFP	National Spent Nuclear Fuel Program
QA	quality assurance
RW	Office of Civilian Radioactive Waste Management
SNF	spent nuclear fuel
SSC	systems, structures, and components
TRIGA	Training Research Isotopes—General Atomic
TSPA	total system performance assessment

NSNFP Activities in Support of Repository Licensing for Disposal of DOE SNF

1. INTRODUCTION

The National Spent Nuclear Fuel Program (NSNFP) was formally established[a] in 1995 as the result of the spent nuclear fuel (SNF) settlement agreement[1] between the State of Idaho, U.S. Department of Energy (DOE), and the United States Navy. Since its inception, the NSNFP has directed the research, development, and testing of treatment, shipment, and disposal technologies for all DOE SNF. The program operates under the direction of the Manager, DOE Idaho Operations Office (NE-ID).[2] The NSNFP was directed by the DOE Office of Environmental Management (EM) to coordinate the functions necessary to disposition DOE SNF. The NSNFP has been coordinating with the DOE SNF sites (Hanford, the Idaho National Engineering and Environmental Laboratory [INEEL], and the Savannah River Site) to achieve its goal of dispositioning DOE SNF in a monitored geologic repository (MGR). NSNFP is working closely with the DOE Office of Civilian Radioactive Waste Management (RW) to meet this goal.

The primary purpose of this report is to present the most significant NSNFP technical activities that have been completed by the NSNFP in support of the repository license application at Yucca Mountain for DOE SNF. Secondly, this report will discuss the relationship between the NSNFP technical activities and the license application.

Several of the NSNFP activities were performed to provide risk insights and a comprehensive understanding of DOE SNF disposal as well as to prepare for questions from the regulatory agency. Thus, when appropriate, this report will also identify how the NSNFP activities:

- Help determine the important DOE SNF disposal parameters, models, and assumptions
- Support the comprehensive understanding of the impact of including DOE SNF in the repository system.

1.1 Background

After President Bush signed the House Joint Resolution 87[3] on July 23, 2002, RW proceeded with the license application process for the proposed MGR at Yucca Mountain, Nevada. Figure 1-1 is a summary diagram providing a high-level timeline and milestones for RW to move forward with the license application. The lower half of Figure 1-1 highlights the NSNFP activities that support the disposition of DOE SNF in the repository. These activities correspond to the RW milestones at the top of the figure. Basically, the process entails:

- Submitting a license application
- Receiving a construction authorization from the Nuclear Regulatory Commission (NRC)
- Submitting a license amendment to receive and possess source, special nuclear, or byproduct material, including SNF and high-level waste (HLW)
- Amending the license for permanent closure
- Requesting an amendment to terminate the license (to close the repository).

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a. NSNFP activities were initiated in 1993, and the NSNFP organization was formalized in the 1995 settlement agreement.

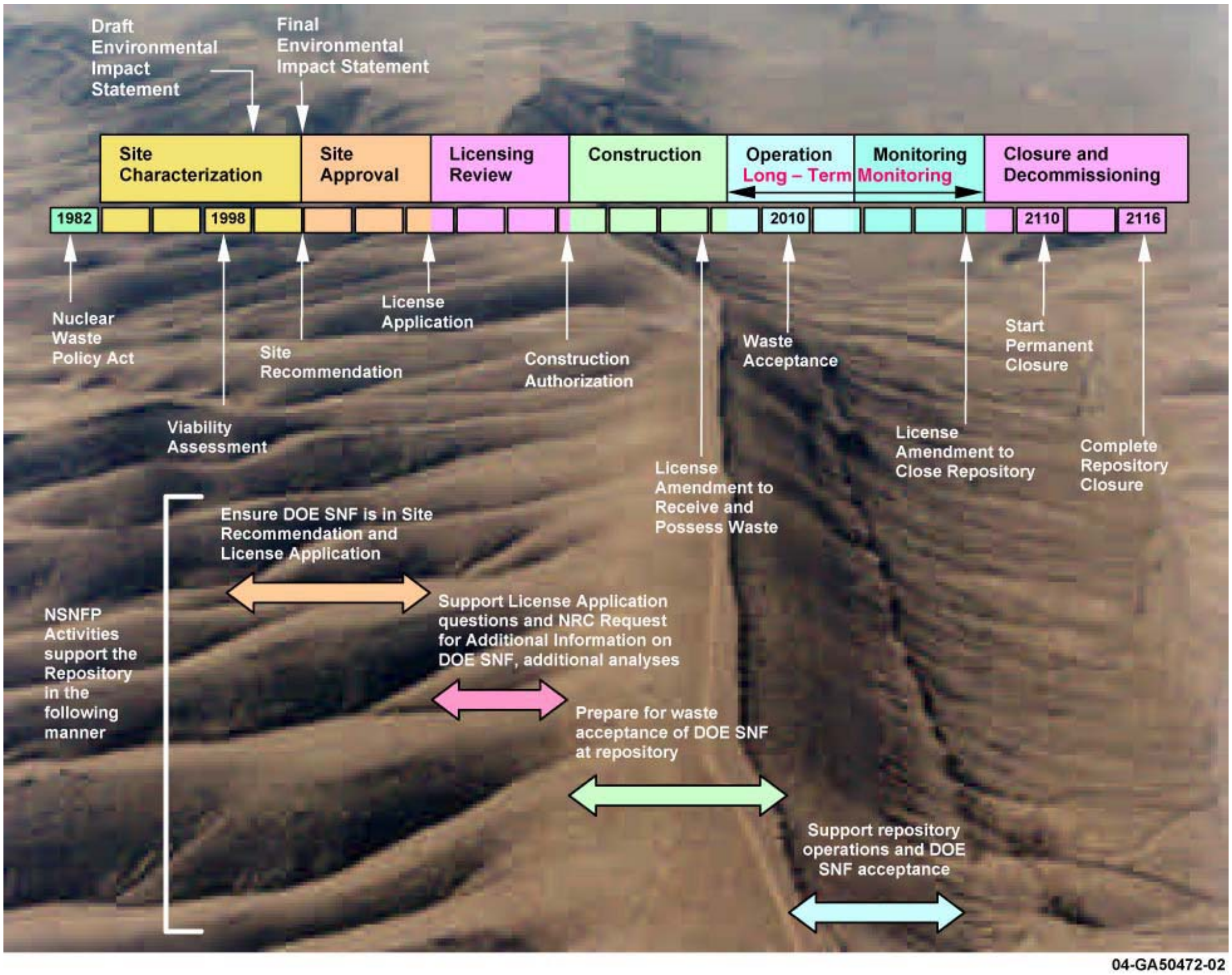

Figure 1-1. Repository timeline and milestones including the NSNFP activities supporting the milestones.

In July 2003, based on the final rule 10 CFR Part 63, the NRC issued the *Yucca Mountain Review Plan*.[4] This plan provides guidance for the NRC staff to evaluate the DOE license application for the proposed Yucca Mountain repository. Figure 1-2 lists the detailed review topics and areas defined in the *Yucca Mountain Review Plan*.

In anticipation of the license application review by the NRC, RW has issued the *Yucca Mountain Project Licensing Strategy*.[5] This document describes the overall approach RW will use during the licensing process for a geological repository at Yucca Mountain, Nevada. Figure 1-3 lists the RW key documents that support the license application as described in the *Yucca Mountain Project Licensing Strategy*.

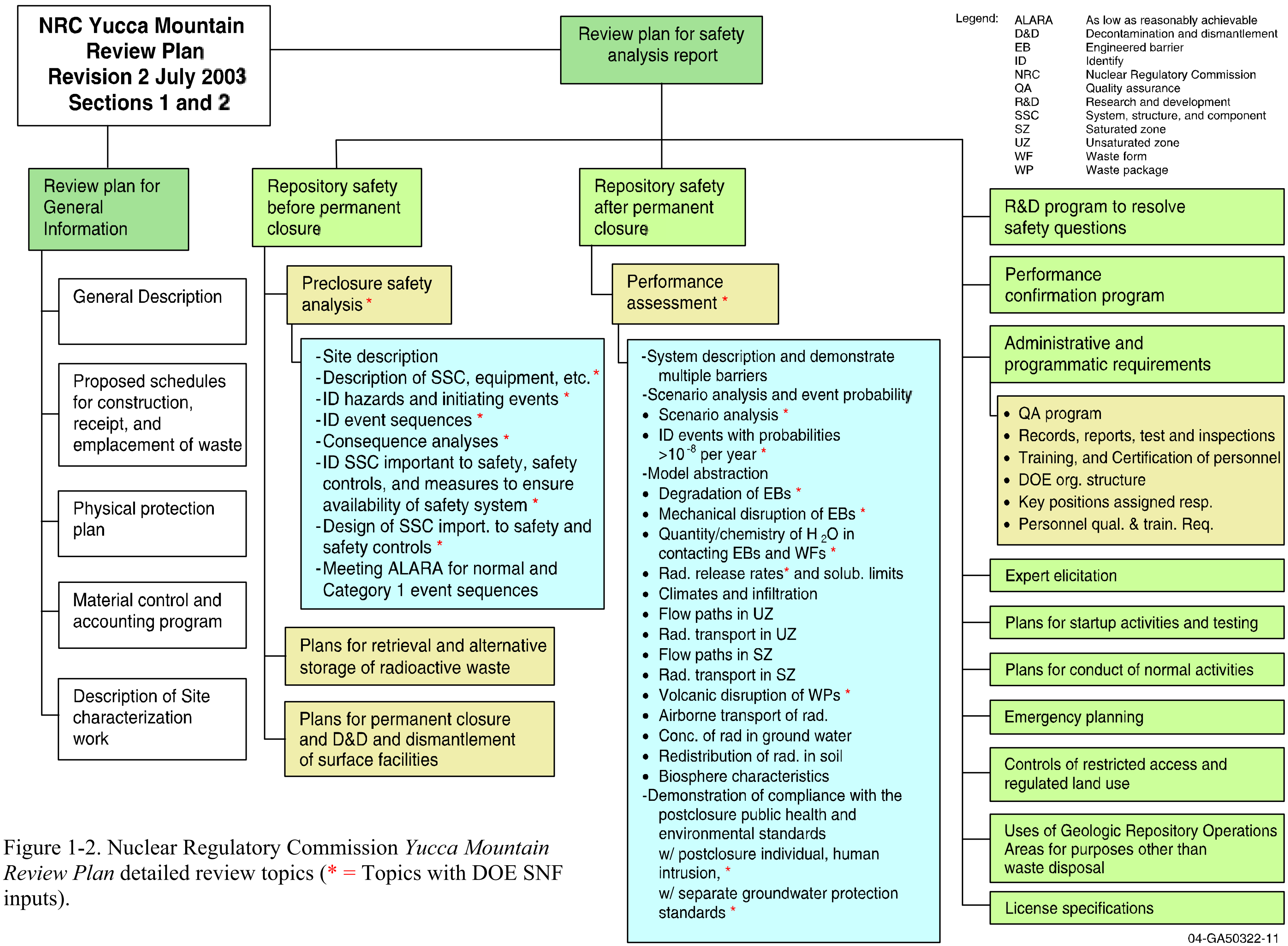

Figure 1-2. Nuclear Regulatory Commission *Yucca Mountain Review Plan* detailed review topics (* = Topics with DOE SNF inputs).

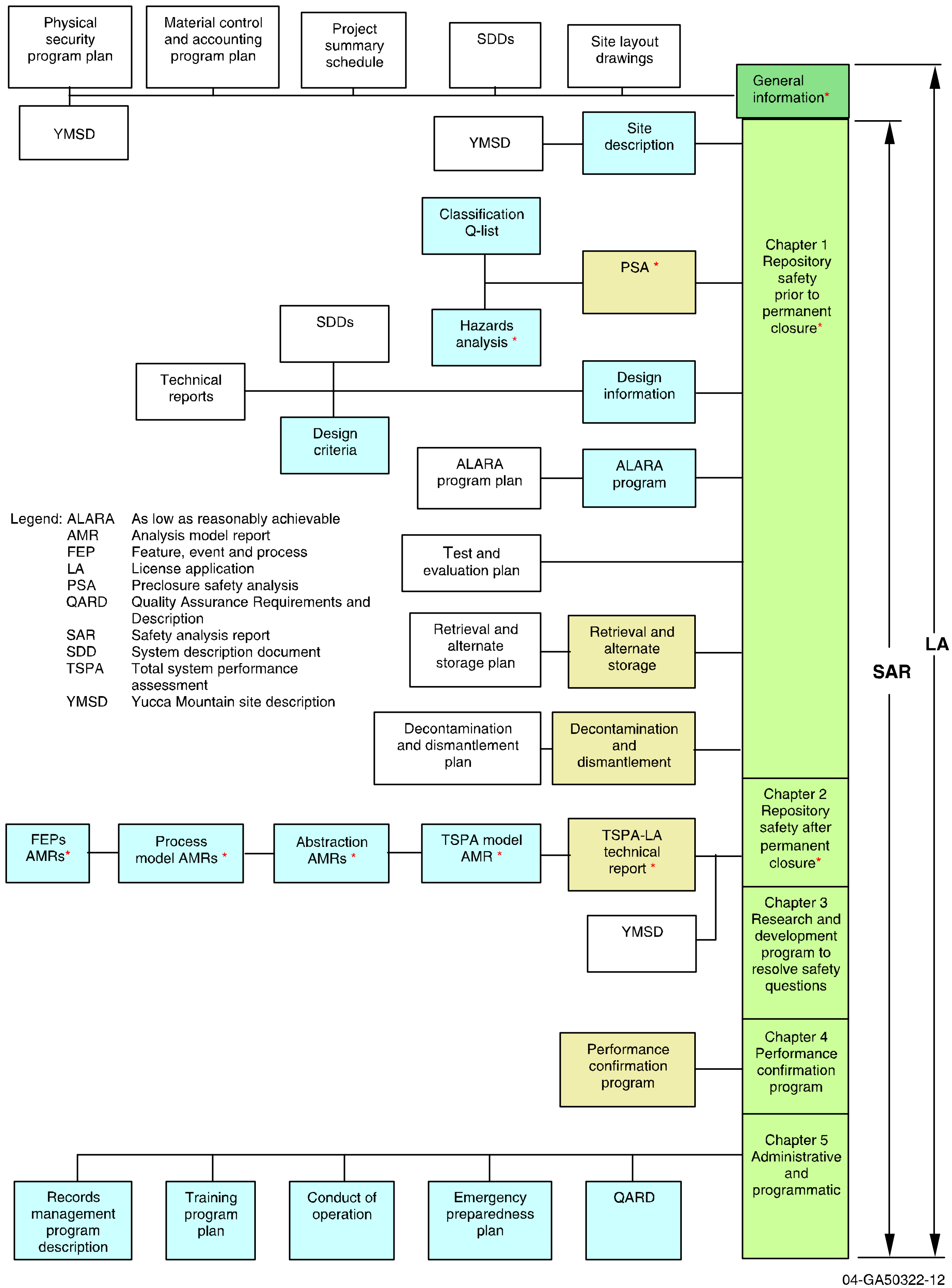

Figure 1-3. Key documents that support the licensing application—from *Yucca Mountain Project Licensing Strategy* (* = Documents with DOE SNF input).

As part of the disposal technologies responsibility, the NSNFP performs a number of technical activities to ensure that all the DOE SNF destined for direct disposition is considered and included in the nation's first MGR planning and licensing process. As required, these activities were performed either by the NSNFP under the NSNFP quality assurance (QA) program procedures or the RW's QA program procedures. To date, these NSNFP activities have supported the inclusion of DOE SNF in the following repository activities.

- Viability assessment (1998)
- Environmental Impact Statement (2002)
- Site recommendation (2002)
- License application (planned December 2004).

A number of NSNFP technical activities provided the necessary input to RW's key licensing documents to ensure that DOE SNF types have been evaluated adequately in the repository license application. The required areas of DOE SNF input for the NRC review topics in Figure 1-2 and key RW documents in Figure 1-3 are identified by an asterisk (∗). These NSNFP activities are summarized in this report.

DOE RW plans to submit a license application to the NRC by December 2004. The NSNFP and the DOE SNF sites have been working with RW and its management and operating (M&O) contractor to understand and complete the various technical activities that are required to support the inclusion of the DOE SNF into the license application.

Since 1995, a number of the activities were performed to support the viability of the Yucca Mountain site as a repository and in support of the site recommendation process outlined by law. Viability assessment and site recommendation activities provided insights to the behaviors of DOE SNF, and they will be discussed here if they are used as part of the supporting evidence in the license application. However, this report discusses primarily the activities that support development of the license application.

1.2 Scope

This report discusses activities that support the disposition of all the DOE SNF that is destined for the repository. Currently, there are approximately 2,400 MTHM DOE SNF (excluding naval fuel) identified in the NSNFP report, *Source Term Estimates for DOE Spent Nuclear Fuels,* DOE/SNF/REP-078, Revision 1.[6]

This report has been organized to provide a summary of the significant NSNFP activities performed to date and a summary of the DOE SNF licensing strategy. When appropriate, detailed discussion of the NSNFP activities in support of the DOE SNF licensing strategy are covered in the appendixes.

2. NSNFP ACTIVITY SUMMARY

Since 1995, the NSNFP has directed research, development, and testing activities to address treatment, storage, shipment, and disposal technology needed for disposition of all DOE SNF. This section briefly describes the key technical activities performed since the inception of the NSNFP. These activity summaries provide a historic perspective as to how NSNFP activities evolved through the repository viability assessment, site recommendation, to the current license application process.

The current NSNFP activities are based on the DOE SNF strategy covered in Section 3. Detailed discussions of the current DOE SNF licensing strategy activities include preclosure safety (Appendix A) , postclosure TSPA (Appendix B), and pre- and postclosure criticality activities (Appendix C). Each of the following subsections provides a short description, the reason for the work, and its status or outcome. Figure 2-1 illustrates the major activities described in this section.

2.1 Project Integration and Interfaces

2.1.1 DOE SNF Licensing Strategy

The overall objective of the DOE SNF licensing strategy is to ensure safety of repository personnel, the environment, and the public while minimizing the need for additional characterization of DOE SNF. This is accomplished through reliance on the systems, structures, and components (SSCs) and barriers[b] for preclosure safety and postclosure waste isolation. This reliance on SSCs and barriers rather than DOE SNF characteristics is the basic philosophy applied to the DOE SNF licensing strategy. A well developed licensing strategy allows the DOE Office of Environmental Management (EM) to safely dispose of DOE SNF that is destined for the MGR without expensive treatment or processing. Further discussions of the DOE SNF licensing strategy are covered in Section 3.

2.1.2 NSNFP Quality Assurance

The NSNFP instituted a QA Program when it initiated SNF research and development activities in 1993. The NSNFP implemented and maintains an effective QA program in all aspects of its work that may affect the safety and protection of workers, the public, or the environment. As part of the NSNFP QA program, the NSNFP has the responsibility to verify that each of the DOE sites implemented the RW *Quality Assurance Requirements and Description*[7] for their activities. As directed by EM, this responsibility was transferred to the EM/RW QA Oversight Team[8] in April 2004.

Today, the NSNFP QA program is prescribed in the *NSNFP Quality Assurance Program Plan*[9] (QAPP). The QAPP describes the NSNFP QA policy, the NSNFP organization structure, the internal and external QA interfaces, the general QA program principles applicable to the scope for the NSNFP mission, and the roles and responsibilities of the NSNFP with respect to QA. The NSNFP adopts the principles defined by the *Quality Assurance Requirements and Description* for engineering and design-related activities intended to guide the development of a path forward for successful disposition of DOE SNF. It includes a formal document control process and training system. Work performed by the NSNFP that is relied on to develop design requirements and to demonstrate DOE SNF compliance with repository acceptance requirements is subject to the *Quality Assurance Requirements and Description*. The NSNFP implements QA requirements by complying with NSNFP procedures.

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b. Any material, structure, or feature that for a period to be determined by the NRC prevents or substantially reduces the rate of movement of water or radionuclides from the Yucca Mountain repository to the accessible environment or prevents the release of radionuclides from the waste, 10 CFR 63.2.

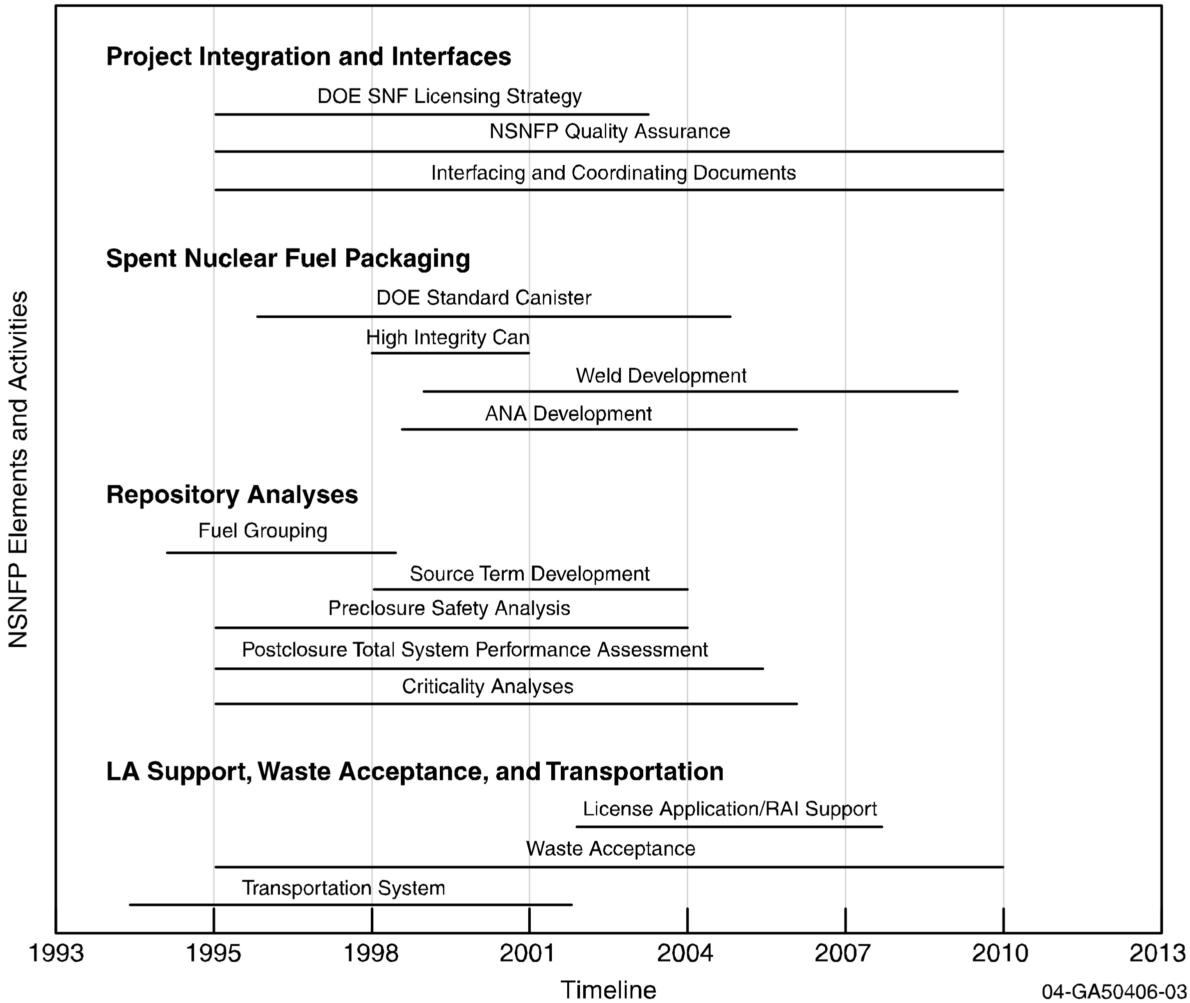

Figure 2-1. NSNFP elements and activities.

2.1.3 Memorandum of Agreement

Through a Memorandum of Agreement (MOA) for Acceptance of Department of Energy Spent Nuclear Fuel and High-Level Radioactive Waste,[10] RW and EM seek to achieve safe and timely disposal of DOE SNF and HLW by identifying data needs, interfaces and acceptance criteria and developing compliance procedures needed to support both the geologic repository construction authorization and license application to the Nuclear Regulatory Commission (NRC) and the transportation system necessary to transfer DOE SNF and HLW to an RW facility.

This MOA states the policy that "EM and RW will cooperate to ensure that all current and future activities relating to acceptance of DOE SNF and HLW continue to be performed in a safe, secure, cost-effective manner, in accordance with applicable requirements, and in a manner that contributes toward a public understanding and acceptance of DOE goals and activities. EM and RW will provide mutual support for budget justification to the Office of Management and Budget, and hearings before Congress to implement the MOA, to the extent consistent with their individual missions."

2.1.4 Interface Control Document

The integrated Interface Control Document Volume (1)[11] was initially issued in 1999 and then updated in 2002. It records and implements interface agreements associated with the receipt and handling by the MGR of DOE SNF, naval SNF, and HLW.

In addition, this Interface Control Document included agreements between RW, EM, and the Naval Nuclear Propulsion Program on each organization's responsibilities for control of the mechanical and physical interfaces. The integrated Interface Control Document Volume (1) was prepared based on the RW/EM relationship that was defined in the memoranda of agreement between EM and RW (see Reference 10.)

2.1.5 Master Logic Schedule

To coordinate the activities being performed to disposition DOE SNF, the NSNFP produced a Master Logic Schedule and interface schedule. The Master Logic Schedule is updated annually to help the DOE SNF sites, the NSNFP, and RW to evaluate and integrate the activities required to dispose of all DOE SNF at the MGR.

2.1.6 Guidance Document

The NSNFP also developed a guidance document as an early attempt to define the minimum data needed for acceptance of DOE SNF in the proposed MGR. The guidance document identified an extensive list of DOE SNF data initially considered necessary for waste acceptance. However, with the development of the DOE standard canister[c] and the DOE SNF licensing strategy, the minimum data needed for DOE SNF have been significantly reduced, and the guidance document is no longer required to support waste acceptance.

2.1.7 High Priority Performance Parameters

In 1995, based on 10 CFR Part 60, the NSNFP identified a number of parameters that were considered important to the acceptance of DOE SNF in the proposed MGR. These parameters included pressurization, radioactive releases from particulates, and combustibility. Together with the RW M&O, a number of evaluations were conducted to determine the impact of these parameters. However, the publication of NRC regulation 10 CFR Part 63 in 2001 (which is performance based) and the development of the DOE standard canister changed the licensing focus. The need to consider these performance parameters changed from licensing analysis to providing risk insights.

2.1.8 Technology Integration Plan

In the early 1990s, each DOE site was developing its technology development needs for interim storage of its SNF. Funds were being requested on an individual site basis to address these needs, sometimes resulting in dual funding for the same technology development. One of the first tasks assigned to the NSNFP in 1994 was to establish a common and consistent technical basis for technology development; integrate the DOE complexwide efforts; and develop a timely, cost-effective technical solution for DOE SNF management. The NSNFP provided this information to the DOE SNF management through the development of the *DOE-Owned Spent Nuclear Fuel Technology Integration Plan*.[12]

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c. The term DOE standard canister refers to both the DOE standardized canister and the multi-canister overpack.

This report documented the systematic methodology used to a) identify the SNF technology needs, b) identify the associated cost and schedules, and c) prioritize the identified technology development needs. The technology needs identified ranged from resolving existing storage issues to new treatment technologies needed to prepare SNF for disposal. DOE SNF site management and EM headquarters staff used this document for fiscal planning of technology development needs after it was issued.

2.2 Spent Nuclear Fuel Packaging

As part of the initial plan to address interim storage, spent fuel packaging needed to be integrated with repository acceptance. The NSNFP began to integrate interim storage plans with repository acceptance criteria resulting in road-ready packaging concepts.

2.2.1 DOE Standard Canisters

DOE plans to package SNF into two types of stainless steel canisters. Most of the fuel will be packages in a DOE standardized canister. The N-reactor and a small amount of other SNF will be packaged into the MCO. The remainder of this report will refer to both as standard canisters.

The NSNFP initiated research and development activities on a single multi-purpose canister to package all the DOE SNF types. In 1997, the NSNFP began to develop and test a standardized canister concept and provide guidance relative to the internal basket. The DOE standardized canister was developed to minimize fuel handling during interim storage, transport, and final disposal operations. The canister incorporates an energy absorbing skirt that protects the heads and shell during a potential drop or sudden impact. The standardized canister is sufficient to withstand operational loads and accidental drops while maintaining containment.

In parallel, Hanford was developing a multi-canister overpack (MCO) to repackage N-reactor fuel and move it from water-filled basins near the Columbia River to a dry storage facility. The MCO was initially designed for interim storage to satisfy a near-term need to move the N-reactor fuel. The NSNFP performed an early review of the MCO and determined that a minor change to the MCO internal baskets design would increase the probability that it could be transported in a horizontal configuration to the repository at a later date. The NSNFP has demonstrated that the MCO can meet the operational loads and accidental drops at the repository surface facility and will be analyzing the MCO for transportation loads in the near future.

The overall integrity of the DOE standard canister was demonstrated through analytical modeling and multiple drop tests at varying impact angles followed by helium leak testing. The DOE standard canister is a key component in the licensing strategy for DOE SNF. A more detailed description of the canister and its development is found in Appendix A.

2.2.1.1 Material Interaction. For the DOE standard canisters, internal interactions considered the chemical, physical, and thermodynamic properties of the materials to be stored inside the canisters. Degradation mechanisms that were considered include: electrochemical interactions, such as general corrosion, pitting corrosion, and stress corrosion cracking; mechanical forces, such as overpressurization; and metallurgical degradation such as hydrogen embrittlement, liquid metal embrittlement, and thermal effects due to welding. It was concluded that there are no significant degradation mechanisms that would fail a standardized canister or MCO at nominal repository temperatures of 200°C or even as high as 350°C. It was further concluded that neither the proposed DOE standardized canister constructed of Type 316L stainless steel nor the MCO made with Type 304L stainless steel is expected to be susceptible to liquid metal embrittlement due to the presence of cesium and rubidium from the fuel. The canister shells are immune to stress corrosion cracking from cesium/rubidium hydroxide based on the experiments performed by the NSNFP. After drying and inerting, the DOE standard canister degradation is considered negligible.

In 2001, the NSNFP initiated drying studies and began to generate consensus standards for dryness of SNF for packaging SNF in the DOE standard canisters. The standard guide on drying[13] behavior of SNF is intended to provide DOE sites and regulators with consistent guidance for the evaluation of SNF dryness.

2.2.1.2 High Strain Testing. The use of finite element analysis for elastic structural response has been successfully used in numerous industries including the nuclear power industry. Full confidence exists in the technology's ability to provide acceptable answers. Additional testing was necessary to obtain a similar level of confidence for plastic analysis technology.

The NSNFP has embarked on an effort to develop material data to support modification to the stress-strain curve to account for strain rate effects under moderate strain rates. This material data will be based on limited dynamic material testing at strain levels and strain rates approximating what the DOE standard canister is expected to experience during accidental drop events. These strain and strain-rate values will be determined from the applicable canister analytical evaluations performed to date, which used plastic analysis techniques using ABAQUS/Explicit software.

Material testing will be performed using an impact test machine developed by the INEEL. These impact responses will then be compared to the stress-strain results of a quasi-static tensile test of the same material. These data will be used to confirm the plastic analysis values used in the canister analysis.

2.2.1.3 Thermal Analysis. Two thermal analyses were performed in 1998 to evaluate the allowable decay heat values that could be supported by the DOE standardized canisters and the NSNFP conceptual transportation cask. The first analysis determined the allowable decay heat value that could be supported by the DOE standardized canister oriented vertically in ambient air at 21°C (70°F). The second analysis determined the allowable decay heat value such that under steady state conditions with nine DOE standardized canisters arranged inside a transportation cask, a maximum temperature of 316°C (600°F) occurs in the stainless steel wall of any DOE standard canisters. These analyses determined what kind of administrative controls are needed during the transportation of the DOE standardized canister in a cask. The results of the analyses are documented in a letter report.[14]

2.2.2 High Integrity Can

The NSNFP began the development of a small diameter, high integrity can (HIC) in 1998. The intent of the HIC was to provide containment of items that may not meet the requirements for storage, transportation, or disposal. Specifically, the NSNFP developed the HIC for the handling and packaging of failed fuel and other radioactive materials such as: particulate, sectioned pieces, rubble, melted or highly degraded elements, unclad uranium alloys, and chemically reactive fuels. A feasibility study[15] was performed to evaluate the development of a HIC. Developers completed the first HIC in 2001 for the packaging of sectioned TRIGA rods. With the development of the DOE standard canisters, the need for additional protection by the HIC is no longer required.

2.2.3 Weld Development

Closure welding is the final step in sealing the DOE standardized canister. To optimize its volume, the canister does not incorporate shielding. Without shielding, the DOE standardized canister must be closed and welded remotely. In 2000, the NSNFP began to develop a remote welding technology to support the DOE standardized canister closure process. This technology includes a welding and nondestructive examination process to perform and inspect closure welds. The method is being developed to meet the American Society of Mechanical Engineers (ASME) fabrication code requirements. It is also being developed to minimize weld heat input and metallurgical structure interruption, and to minimize radiation exposure to operation personnel. Prototype demonstrations of the system have begun.

2.2.4 Advanced Neutron Absorber Development

The NSNFP began developing a neutron absorbing structural material in 1999 to support nuclear criticality safety for interim storage, transport, and final disposal of SNF. Researchers are developing a corrosion-resistant, nickel-chromium-molybdenum alloy containing gadolinium for criticality control in the DOE standard canister. Gadolinium is a neutron-absorbing element that has the highest available neutron absorption cross section. The gadolinium must be alloyed into a corrosion-resistant structural metal that will meet ASME, Section 2 code requirements to be used as a structural material.

This alloy will be used for the internal baskets of the DOE standardized canister to provide structural support and geometry control and to ensure nuclear criticality safety. Use of poison inside the canister allows higher fissile loading per canister. This will reduce the number of canisters and waste packages needed for DOE SNF disposal. Researchers are working to define the chemistry ranges and minimum mechanical properties for the ASME code case.

2.3 Repository Analyses

For the DOE SNF to be included in the MGR baseline, a series of analyses were required to show that DOE SNF does not impact repository performance. This section discusses the NSNFP activities that support repository analyses.

2.3.1 SNF Database

In 1993, the NSNFP created the Spent Fuel Database to provide a single source of DOE EM site data to make management decisions for handling, storage, transport, and disposal of SNF. The Spent Fuel Database contains quantitative and characteristic SNF information for all DOE-owned or managed SNF and provides the ability to search on a wide range of parameters. The database is updated periodically and a reference version was generated to support the license application.

2.3.2 Fuel Grouping

The DOE SNF inventory is diverse. This large number of DOE SNF types posed a challenge in the number of analyses needed. The necessary analyses were simplified by using the information from the Spent Fuel Database to group the DOE SNF and to facilitate preclosure safety analysis (also known as design basis event), postclosure total system performance assessment (TSPA) analysis, and criticality analysis. DOE SNF was grouped by similar materials of construction, fuel meats, enrichments, etc. This resulted in a small number of representative groups for analyses. The results are discussed in Appendixes A and B.

2.3.3 Source Term Development

The source term development activity provided an estimate of radionuclide inventories that are used to support determination of the radiological doses associated with preclosure and postclosure repository safety, the decay heat for thermal analyses of casks and storage canisters, and photon emission spectra for shielding calculations. The development, application, and results of the methodology employed for the estimates are discussed in Appendix D.

2.3.4 Preclosure Safety Analysis (Design Basis Events)

Design basis events analysis was performed to identify risks and to establish the associated administrative and operational controls needed to ensure safety during SNF receipt and handling (repository preclosure operations). In 1998, the NSNFP began participating in repository design basis events analysis with the objective of demonstrating that DOE SNF will not adversely affect repository safety during credible repository events. The NSNFP has demonstrated that the DOE standard canister provides radionuclide containment during credible preclosure events. The NSNFP performed analyses that provide risk insights associated with receipt and handling of DOE SNF. The preclosure safety strategy for DOE SNF and associated analyses are discussed in Appendix A.

2.3.4.1 Chemical Reactivity Analysis. Other preclosure safety considerations included the possibility of a fire resulting from a pyrophoric reaction involving uranium metal SNF. Fuels with damaged cladding can accumulate uranium hydride on their surface if they react with water while in storage. The repository staff conducted simplified analyses to assess the potential impacts of this event sequence. These analyses conservatively assumed ignition and complete combustion of an MCO full of N-reactor SNF in the repository surface facilities. No credit was taken for high-efficiency particulate air filtration or other natural and engineered barriers to mitigate the consequences. Because these analyses indicated that the potential energy and radiological materials released from this event were unacceptable, chemical reactivity analyses were undertaken to develop a more accurate assessment of the potential for pyrophoric reaction associated with receipt and handling of uranium metal SNF. The analysis results showed that for a sustained reaction to occur, two holes must be present in the SNF canister to allow a flow path through the SNF. The holes allow the inflow of reactant gas and outflow of reaction products, necessary to support the reaction. Holes of less than 0.75 inches retarded gas flow sufficiently to make the reaction self-controlled and not sustainable. This analysis, coupled with the no canister breach approach to the license, eliminated chemical reactivity as a licensing issue. Chemical reactivity analysis is further discussed in Appendix D.

2.3.5 Postclosure Total Systems Performance Assessment

The TSPA activities were performed to confirm that DOE SNF can safely be emplaced in a geologic repository. The TSPA activities predict how well the repository's engineered and natural barriers will contain the DOE SNF over the 10,000-year regulatory period. The NSNFP performed iterative analyses to:

- Determine if DOE SNF would affect any of the repository features, events, and processes (FEPs)
- Retain and evaluate only those FEPs with probabilities $>10^{-8}$ per year, or significantly change radiological exposure or radionuclide releases
- Determine if the repository can still meet environmental safety and health requirements by including the various DOE SNF types.

The NSNFP has performed TSPA activities in support of both the site recommendation and license application processes. The postclosure safety strategy and related analyses are presented in Appendix B.

2.3.5.1 Features, Events, and Processes. 10 CFR 63.114(e) and (f) require that the repository performance assessment provides the technical basis for either inclusion or exclusion of specific FEPs.

Specific FEPs must be included in TSPA analyses if the magnitude and time of the resulting radiological exposures would be significantly changed by their omission. The repository performance assessment provides the technical basis for either inclusion or exclusion of degradation, deterioration, or alteration processes of engineered barriers, including those processes that would adversely affect the performance of natural barriers. Degradation, deterioration, or alteration processes of engineered barriers were evaluated in detail if the magnitude and time of the resulting radiological exposures would be significantly changed by their omission. RW completed this screening for the commercial SNF.

The DOE SNF FEPs activity evaluated the impact of the addition of DOE SNF into the proposed repository. This activity identified the FEPs that should be analyzed to provide the basis for RW to determine if the FEPs need inclusion in the performance assessment. Appendix B further discusses the FEPs activity for DOE SNF.

2.3.5.2 Corrosion Data Report. The behaviors of the DOE SNF in the TSPA were investigated through the use of a literature search to help understand how the various DOE SNF types may behave under repository conditions. The results of the literature search were published in two reports.[16,17] These reports, in concert with the other postclosure activities, will provide NRC the confidence that EM understands the behaviors of DOE SNF. Further discussion of this and other related works are covered in Appendix B.

2.3.5.3 Release Rate Testing. From 1998 through 2002, the NSNFP provided support to the repository license application process by determining the expected release rate of radionuclides from SNF. These release rates support the prediction of the radiation dose rate at the repository boundary and assessment of the potential impact to criticality safety. This work was performed to establish a technical basis for the degradation and release rate predictions for performance assessment of the repository system.

Through 2002, the NSNFP performed tests on mixed-oxide fuel, uranium metal fuel, and aluminum clad fuel. Three primary tests supported the fuel degradation and release rate studies: drip testing, flow-through testing, and batch testing. These tests[18] provided data related to corrosion mechanisms, reaction products, and the retention and release of selected radionuclides. The data obtained from these tests were used to increase confidence in SNF performance assessment models. Release rate testing is further discussed in Appendix B.

2.3.6 Criticality Analyses

Criticality analyses were needed to support the repository license application and ensure safe DOE SNF handling and transport. In 1994, the NSNFP initiated criticality analyses in an effort to ensure safe handling during storage, transport, and disposal of DOE SNF and to define the appropriate criticality controls needed in the waste package design. Criticality analyses have been key in the development of the DOE standard canister and internal baskets. Initial analyses also led to the development of the advanced neutron absorber material. Analytical bases were established to group the many SNF types and focus analyses on the most reactive SNF types in each of nine fuel groups.

These criticality analyses have contributed to the development of safe fuel packaging guidelines by establishing fissile material limits and neutron poisoning requirements. These guidelines[19] form the basis for safe interim storage, transport, and repository emplacement. Appendix C further discusses the criticality activities.

2.4 License Application Support, Waste Acceptance, and Transportation

2.4.1 License Application/Request for Additional Information Support

Since FY 2003, the NSNFP has been coordinating the license application review for the EM organization. RW and its M&O contractor developed a license application review process to ensure the technical accuracy of the license application. The NSNFP coordinates the process by providing the license application materials to designated EM personnel for review when each section or chapter becomes available. These include the federal and contractor personnel at each DOE site, and EM Headquarters personnel in both the SNF and HLW programs. The NSNFP collects all the comments from the reviewers, resolves inconsistent comments, consolidates comments, and submits them for resolution to RW. The NSNFP represents EM and participates in the EM comment resolutions.

The NSNFP will continue this support through completion of the license application. The NSNFP will answer NRC's request for additional information regarding the DOE SNF. The NSNFP will perform additional DOE SNF analysis as required to address NRC's requests.

2.4.2 Waste Acceptance

The Nuclear Waste Policy Act limits repository inventories to those wastes that meet the legal definition of SNF or HLW. It is critical that this fundamental requirement flows down to the Waste Acceptance element and be a central discriminator in what types of waste are accepted into the MGR.

2.4.2.1 Integration with RW. As part of the process under agreements between EM and RW, RW has committed to identify acceptance criteria. The NSNFP supported RW's efforts in developing performance-based acceptance criteria. By developing a risk-informed strategy, the NSNFP was able to eliminate reliance on the waste form and the characterization requirements that have no basis in law, safe conduct of operations, or long-term performance. The acceptance of DOE SNF is not expected to require extensive measurement and remediation. Therefore, in most cases, the requirements are focused on SSCs important to safety and waste isolation.

2.4.2.2 Integration with EM Sites. Site compliance plans describe how a specific EM site will demonstrate compliance with each requirement in the RW acceptance criteria. These documents include descriptions of the tests, analyses, and process controls to be performed by EM, including the identification of records to be provided to demonstrate compliance with the specifications. The NSNFP coordinated the preparation of these plans and evaluated their adequacy.

2.4.2.3 Safeguards. From 1998 through 2004, EM and RW examined safeguards and security issues related to the disposal of DOE SNF. The RW approach to implementing safeguards and security at the MGR is based on:

> …a demonstration that candidate materials are no more attractive from a theft standpoint than commercial spent nuclear fuel or vitrified HLW and, thus, are adequately protected.
>
> From a licensing perspective, four characteristics are important to demonstrating the unattractiveness of a candidate material from the standpoint of theft:
>
> - Size, including overall weight
> - Fissile material content

- Relative difficulty of separation

- Homogeneity and concentration of special nuclear material content.[20]

The NSNFP applied these characteristics to DOE SNF to determine the relative attractiveness of DOE SNF relative to commercial SNF. An initial screening task used an expert elicitation process to examine the last two characteristics i.e., relative difficulty of separation, and homogeneity and concentration. This produced a fuel specific determination of difficulty of separation relative to commercial SNF. Subsequent to this, the NSNFP participated in a second expert elicitation workshop where overall attractiveness was determined based on size, fissile mass, and ease of separation. It was concluded that DOE SNF when packaged in large sealed canisters, such as DOE standard canisters, is no more attractive for theft than commercial SNF.

2.4.2.4 Multi-Detector Analysis System. In 1997, the NSNFP began research and development on a technology to characterize SNF for fissile mass, radiation source term, and fissile isotopic content. This work was performed to provide confirmatory data for some DOE SNF types. The NSNFP began to investigate a method to directly measure properties of SNF. The Multi-Detector Analysis System integrated a large detector array, fast electronics, high-speed data acquisition, real-time analysis, and archival mass storage methods to characterize SNF and remote-handled transuranic waste without special calibration standards or a priori knowledge.

Although the prototype system was successfully used to identify spontaneous fission in natural sources, the Multi-Detector Analysis System research was discontinued in 2001. This decision was based on signal source strength, mobility, and signal to noise ratio issues that made the system impractical for systematic SNF analysis. In addition, the NSNFP licensing strategy was evolving to rely on the robust containment properties of the DOE standard canister. Therefore, detailed fuel-specific characterization was determined to be unnecessary for repository waste acceptance.

2.4.3 Transportation System

In 1994, the NSNFP began development of a transportation system for the safe, efficient, shipment of DOE SNF. The transportation system was being developed according to 10 CFR Part 71 transportation system requirements to be licensed by the NRC. The NSNFP developed this system to accommodate the wide variety of DOE SNF types. A cask system was being developed with nine interchangeable basket configurations to allow for custom fuel shipments that met the needs of each DOE SNF site. The system would minimize cask-handling operations and personnel exposure, and it would reduce load and transfer times.

In 2001, DOE consolidated the responsibility for SNF transportation within RW, and work on this effort was discontinued by the NSNFP. RW is currently in the process of establishing a contract for the transportation of all SNF, both commercial and DOE SNF, to the repository. At the request of RW, some of the NSNFP development work is being provided for their use.

2.5 The NSNFP Today

By completing the work described above, the NSNFP has developed an understanding of what is important in terms of preparing the DOE SNF for final MGR disposal and interim storage. Today, the NSNFP continues to work with the EM sites to effectively manage and package DOE SNF for interim management. For long-term success, the NSNFP has identified a number of program elements that will ensure the final MGR disposition of all DOE SNF. The first program element is to ensure a well-developed DOE SNF licensing strategy that includes all DOE SNF. The second element is to complete all necessary DOE SNF analyses needed to support the final repository license application. The third element is to provide licensing application/request for additional information support, transportation, and waste acceptance support to ensure DOE SNF is included in the initial "receive and possess" license amendment and that DOE SNF is transportable using the final RW cask design. Section 3 provides additional details on the DOE SNF licensing strategy. The appendixes discuss the analyses supporting the DOE SNF licensing strategy.

3. DOE SNF LICENSING STRATEGY

At the March 2002 quarterly meeting between the RW and the EM, the RW management and operations contractor and the NSNFP were directed to reach consensus on the technical bases and approach for including DOE SNF in the license application. Consensus was reached through a series of interactions held between June and August 2002.[21] These interactions addressed preclosure safety, criticality safety, and postclosure performance assessment. As the repository design matures, the DOE SNF licensing strategy may be updated as needed to ensure all DOE SNF destined for direct disposition is accepted in the MGR.

In accordance with the NRC regulation, this licensing strategy uses a risk-informed, performance-based decision-making process to identify SSCs that are important-to-safety and waste isolation. Additional measures are also considered and included, as necessary, to provide defense-in-depth. This section is an overview of the licensing strategy for DOE SNF in the repository-licensing basis.

Risk is a couplet involving both likelihood and consequences. Therefore, each safety function has both preventive and mitigative elements. Prevention is preferable to mitigation because it results in the elimination of the risk rather than merely reduction of undesirable consequences. The safety case diagram, Figure 3-1, has separate branches for the preclosure and postclosure program phases. For each safety function, there is at least one preventive and one mitigative control strategy. For each control strategy, there is one or more SSC and barriers. SSCs and barriers are shaded in Figure 3-1 to highlight those that are the primary focus of this strategy. The SSCs and barriers in darker shading are important to safety or important to waste isolation, and the SSCs and barriers in lighter shading are considered to be additional measures.

The analyses used to demonstrate compliance with 10 CFR 63.111(b) and 10 CFR 63.113(b) will only credit SSCs and barriers that are important to safety or important to waste isolation. The NSNFP's objective is to have the DOE SNF analyzed in the license application such that no fuel will be rejected.

3.1 Preclosure Strategy

For preclosure safety, the maximum allowable radiological consequences for an event (or sequence of events) is based on the estimated frequency of occurrence. Category 1 event sequence is a series of actions and/or occurrences that are expected to occur one or more times before permanent closure of the geologic repository operations area that could potentially lead to exposure of individuals to radiation. Category 2 event sequence is a series of actions and/or occurrences that have at least one chance in 10,000 of occurring before permanent closure. Beyond Category 2 event sequence is a series of actions and/or occurrences that have less than one chance in 10,000 of occurring before permanent closure. The maximum allowable consequences for the Category 1 and Category 2 events are specified in 10 CFR part 63.111. There are no requirements for Beyond Category 2 events.

The key to the preclosure strategy for both radionuclide confinement and criticality control is the DOE standard canisters and their ability to maintain confinement under credible conditions and accidents.[23] As long as the DOE standard canisters maintain confinement, there will be no radionuclide release, and there can be no criticality. Many of the NSNFP activities have focused on demonstrating the reliability of the DOE standard canisters. This strategy has minimal reliance on fuel-specific DOE SNF characterization information. Existing DOE SNF information is sufficient for facility design of the shielding and ventilation systems and for Beyond Category 2 analyses. Even in the highly unlikely event

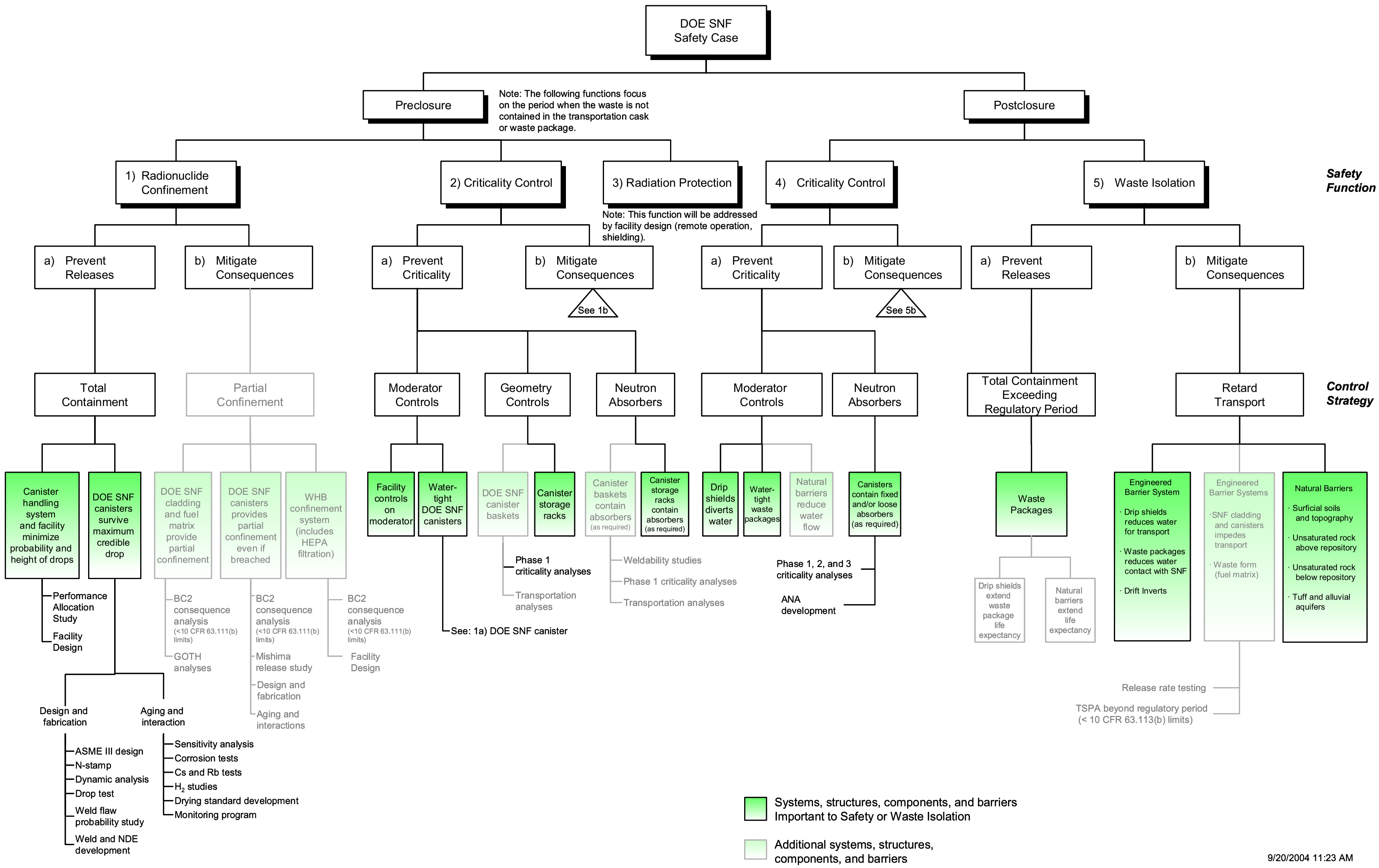

Figure 3-1. DOE SNF safety case strategy.

the DOE standard canister breaches,[d] any releases would be expected to be well below allowable limits based on retention within the DOE standard canister, credit for cladding or fuel matrix, etc.

The MGR facilities will be designed to provide gamma and neutron radiation shielding as required for worker protection. Because there is nothing unique about the DOE SNF that requires special radiation protection measures, a strategy for radiation protection specific to DOE SNF is not needed, and this safety function is not addressed further.

3.2 Postclosure Strategy

The DOE SNF postclosure licensing strategy, which is the same as for the commercial SNF, relies on the combination of engineered and natural barriers to meet performance objectives of 10 CFR 63.113(b). As part of the design process, RW has identified specific SSCs as being important to waste isolation. These SSCs either prevent or substantially reduce movement of water or prevent or substantially reduce releases of radionuclides from the waste. These barriers work in concert with one another to ensure the regulatory objectives are met. For example, the drip shield would remain intact beyond the regulatory period and prevent groundwater from dripping onto the waste packages, thereby limiting transport to diffusion even if a waste package was to fail.

Reasonable values and approaches will be used in evaluating the postclosure safety aspects to allow an expected or “realistic” value to be determined. Simple, bounding evaluations will be used when this approach does not overly constrain the design or the operations of the facility. To represent individual DOE SNF waste packages in the licensing bases, DOE’s intention is to use bounding analyses to demonstrate that the performance objectives cannot be exceeded in the event DOE SNF waste packages were to fail during the regulatory time period. DOE SNF information related to total key radionuclides (see Reference 6) and a reasonable range of total number of canisters[24] will be used to represent DOE SNF in the nominal base case.

For DOE SNF, an instantaneous release rate will be used with nominal radionuclide inventory to show DOE SNF is still well below limits. This instantaneous release model is conservative because the DOE standard canister, cladding, and realistic degradation rates would delay the release. However, these are not credited in the licensing bases.

3.3 DOE SNF Licensing Strategy Summary

This strategy places primary reliance on SSCs and barriers, mainly the DOE standard canisters and the engineered and natural barriers. This strategy places minimal reliance on additional DOE SNF characterization. In addition to demonstrating compliance with regulatory limits, the strategy provides additional measures consistent with the defense-in-depth philosophy. Analyses and testing described in the appendixes support this strategy.

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d. An opening in a transportation cask, spent nuclear fuel canister, disposal canister, waste package, or drip shield caused by corrosion or mechanical stress.

4. QUALITY ASSURANCE

This document was developed and is controlled in accordance with NSNFP procedures. Unless noted otherwise, information must be evaluated for adequacy relative to its specific use if relied on to support design or decisions important to safety or waste isolation.

The NSNFP procedures applied to this activity implement DOE/RW-0333P, *Quality Assurance Requirements and Description*, and are part of the NSNFP QA Program. The NSNFP QA Program has been assessed and accepted by representatives of the Office of Quality Assurance within the Office of Civilian Radioactive Waste Management for the work scope of the NSNFP. The NSNFP work scope extends to the work represented in this report.

The current, principal NSNFP procedures applied to this activity include the following:

- NSNFP Program Management Procedure (PMP) 6.01, "Review and Approval of NSNFP Internal Documents"
- NSNFP PMP 6.03, "Managing Document Control and Distribution"
- NSNFP Program Support Organization (PSO) 3.04, "Engineering Documentation."

5. COMPUTER CODE/SOFTWARE

Microsoft® WORD and EXCEL 2000 SR1 programs loaded on a DELL OptiPlex GX260 were used to generate this report and various tables in this report. No other computer software was used in the preparation of this report.

6. CONCLUSION AND SUMMARY

This report covers the technical activities of the NSNFP in support of the licensing process. Since the inception of the NSNFP, the proposed MGR has moved from viability assessment (determine its viability as a repository), through site recommendation (the study of its adequacy to be recommended by the Secretary of DOE as a repository site), to the current preparation of a license application. During this period, the NRC has promulgated the new regulation 10 CFR 63 for the MGR to protect the public and the environment.

The NSNFP has refined and adjusted its technical activities within an evolving regulatory environment to ensure that DOE SNF continues to be part of these licensing processes. The DOE SNF was included as part of the repository waste form when DOE SNF was included in the 1998 RW baseline. The DOE SNF was included in the viability assessment in 1998 and the site recommendation in 2001. And the DOE SNF will be included in the license application planned in December 2004. The DOE SNF licensing strategy defines the acceptance path for DOE SNF.

The NSNFP will continue to evaluate the needs of the repository program and work with the sites to ensure the proper technical activities are completed so that all DOE SNF planned for direct disposition is accepted into the repository.

7. REFERENCES

1. U.S. Department of Energy, Consent Order (PSC 1995) for spent nuclear fuel among the State of Idaho, the U.S. Navy, and the U.S. Department of Energy, October 1995.

2. U.S. Department of Energy Environmental Management, *Program Management Plan*, DOE/SNF/PP-033, Revision 2, January 2004.

3. House Joint Resolution Bill H. J. RES 87 Introduced April 11, 2002, Passage on May 8, 2002 (Passed by the Yeas and Nays: 306–117).

4. Nuclear Regulatory Commission, *Yucca Mountain Review Plan*, NUREG-1804, Revision 2, July 2003.

5. U.S. Department of Energy Office of Civilian Radioactive Waste Management, *Yucca Mountain Project Licensing Strategy*, YMP/02-01, Revision 0, ICN 01, published in August 2002.

6. U.S. Department of Energy Environmental Management, *Source Term Estimates for DOE Spent Nuclear Fuels,* DOE/SNF/REP-078, Revision 1,

7. U.S. Department of Energy Office of Civilian Radioactive Waste Management, *Quality Assurance Requirements and Description*, Revision 13, DOE/RW-0333P, April 28, 2003.

8. Bob L. Blyth (DOE Idaho) letter to Larry D. Vaughn (EM 2.2), "Formal Transfer of National Spent Nuclear Fuel Quality Assurance Oversight Responsibilities (NSNFP-QA-04-023)," April 1, 2004.

9. U.S. Department of Energy Environmental Management, *NSNFP Quality Assurance Program Plan*, Revision 1, DOE/SNF/QAPP-001, August 22, 2002.

10. U.S. Department of Energy, *Memorandum of Agreement for Acceptance of Department of Energy Spent Nuclear Fuel and High-Level Radioactive Waste*, between EM and RW, Revision 1, Washington, D.C., January 1999.

11. U.S. Department of Energy Office of Civilian Radioactive Waste Management, *Integrated Interface Control Document Volume 1: U.S. Department of Energy Spent Nuclear Fuel and High-Level Radioactive Waste to the Monitored Geologic Repository*, Revision 1, March 2002.

12. U.S. Department of Energy Environmental Management, *DOE-Owned Spent Nuclear Fuel Technology Integration Plan,* DOE/SNF/PP-002, Rev. 1, dated May 1996.

13. American Society for Testing and Materials, "Standard Guide for Drying Behavior of Spent Nuclear Fuel," ASTM Z 8523-02, BALLOT DRAFT, No. 7, January 4, 2004.

14. G. L. Hawkes (INEEL) letter report to Nancy Smith (INEEL), "Letter report on DOE–SNF Canister Thermal Analysis," GLH-02-98, June 18, 1998.

15. U.S. Department of Energy Environmental Management, *Feasibility Study for the High Integrity Can*, DOE/SNF/REP-016, Revision 0, April 1998.

16. U.S. Department of Energy Environmental Management, *Review of Oxidation Rates of DOE Spent Nuclear Fuel Part 1:Metallic Fuel*, DOE/SNF/REP-0054, September 2000.

17. U.S. Department of Energy Environmental Management, *Review of Oxidation Rates of DOE Spent Nuclear Fuel Part II: Nonmetallic Fuel,* DOE/SNF/REP-068, July 2003.

18. U.S. Department of Energy Environmental Management, *Review of DOE Spent Nuclear Fuel Release Rate Test Results*, DOE/SNF/REP-073, Revision 0, August 2003.

19. U.S. Department of Energy Environmental Management, *Qualifying 'Other' DOE Fuels for Criticality Safety in Repository Packaging*, DOE/SNF/REP-090, Revision 0, June 2004.

20. U.S. Department of Energy, "A Strategic Approach to Integrating the Long-Term Management of Nuclear Materials—The Department of Energy's Integrated Nuclear Materials Management Plan, A Report to Congress," June 2000.

21. N. Williams (BSC) letter to J. Zeigler (DOE-RW), "Meeting Notes Between the National Spent Fuel Program (NSNFP), Bechtel SAIC Company, LLC (BSC), and the U.S. Department of Energy (DOE) on DOE Spent Nuclear Fuel Licensing Approach," September 24, 2002.

22. 10 CFR Parts 2, 19, 20, 21, etc., *Disposal of High-Level Radioactive Waste in a Geologic Repository at Yucca Mountain, Nevada; Final Rule*, November 2, 2001.

23. U.S. Department of Energy Environmental Management, FY 1999 Drop Testing Report for the Standardized 18-inch DOE SNF Canister, EDF-NSNF-007, Revision 2, 2002.

24. U.S. Department of Energy Environmental Management, "Determination of the Range of DOE SNF Canisters," EDF-NSNF-033, Revision 1, 2003.

Appendix A

Preclosure Safety Activities

Appendix A

Preclosure Safety Activities

For preclosure safety, 10 CFR Part 63.111 implements a risk-informed approach to safety by establishing maximum allowable radiological consequences for an event (or sequence of events) based on its estimated frequency of occurrence. The risk thresholds established by 10 CFR Part 63.111 are summarized in Table A-1.

This regulatory framework acknowledges that risks may be addressed by preventing the occurrence of events, by mitigating their consequence, or both. For the U.S. Department of Energy (DOE) spent nuclear fuel (SNF), other than those that will be handled and disposed of as commercial SNF, the strategy for demonstrating compliance with preclosure safety requirements relies on prevention. Radiological releases from DOE SNF are prevented by reliance on DOE standard canisters[a] to contain DOE SNF during credible accident scenarios. Reasons for selecting a preclosure licensing strategy based on prevention include:

- Minimization of the need for qualified fuel-specific data—Relying on a sealed canister as an engineered barrier will provide confinement during credible preclosure events, thereby reducing the need for fuel-specific information and avoiding the costs and radiological exposures associated with fuel characterization activities.

- Compliance with ALARA principles—In addition to avoiding the personnel exposure associated with fuel characterization activities needed for qualifying fuel-specific data to support dose calculations, reliance on an engineered barrier eliminates doses associated with radiological releases that may accompany unplanned events.

- Standardization—DOE SNF includes a broad range of fuel configurations that have been handled and stored at various DOE facilities for several decades. Because of the diverse configurations and storage history of this SNF, standard canisters are needed to simplify equipment interfaces needed for handling and transportation.

- Simplification of analyses and controls—Reliance on an engineered barrier designed and tested to current standards eliminates the analytical and administrative burden associated with licensing, certifying, and accepting several hundred different fuel configurations.

The key to the preclosure strategy for both radionuclide confinement and criticality control is the DOE standard canister and its ability to maintain confinement under credible conditions and accidents.[1] The strategy for radionuclide confinement relies on the fact that all DOE SNF is shipped in DOE standard canisters that will preclude any radiological release for all Category 1 and 2 events. Table A-2 identifies the systems, structures, and components (SSCs) and barriers for this strategy. Many National Spent Nuclear Fuel Program (NSNFP) activities have focused on demonstrating the reliability of the DOE standard canister. This strategy has minimal reliance on fuel-specific DOE SNF characterization information. Existing DOE SNF information is sufficient for facility design of the shielding and ventilation systems and for Beyond Category 2 analyses. Even in the highly unlikely event that the DOE standard canister breaches, Beyond Category 2 analyses have shown that any release would be expected to be well below allowable limits. However, Beyond Category 2 analysis will not be part of the licensing bases.

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a. The term DOE standard canister refers to both the DOE standardized canister and the multi-canister overpack.

Table A-1. Preclosure regulatory criteria.

Probability of Occurrence during Repository Preclosure Period (P)	Annual Frequency of Occurrence[a] (P_A)	Event Category	Maximum Allowable Consequence
P>1	P_A > 1E-2/yr	1	Occupational dose limits per 10 CFR 20
1E-4<P<1	1E-6/yr > P_A < 1E-2/yr	2	5 Rem (TEDE) or 15 Rem (lens of eye) or 50 Rem (DDE+CDE to any organ except lens of eye) or 50 Rem (SDE)
P<1E-4	P_A<1E-6/yr	Beyond Category 2	No requirements specified

CDE = committed dose equivalent
DDE = deep dose equivalent
SDE = skin dose equivalent
TEDE = total effective dose equivalent

a. The annual frequency of occurrence was computed by uniformly distributing the allowable occurrences over a 100-year period.

Table A-2. Systems, structures, and components for preclosure radionuclide confinement.

SSC and Barrier	Category	Description
DOE standard canisters (DOE SCs)	Important to safety	For DOE SCs transfer operations in the Canister Transfer System, the DOE SC will be designed and demonstrated to provide total containment under all credible conditions and events including the maximum credible drop.
Canister handling system and associated facility	Important to safety	The canister handling system and facility design (bridge crane, lifting fixtures, etc.) will both minimize the likelihood of DOE SC drops as well as minimize the height of a potential drop.
DOE SNF cladding and fuel matrix	Potential additional measures but no credits taken in the license application at this time	The cladding and matrix will provide partial confinement in the extremely unlikely event of a DOE SC breach. The extent of confinement varies across the fuel types.
Standard canisters (SCs)	Additional measure	Though important to safety with respect to ensuring that a breach is beyond Beyond Category 2, the DOE SC will also mitigate radiological consequences by retaining much of its contents in the extremely unlikely event of a drop and breach.
Canister Handling Facility (CHF) confinement system	Additional measure	The CHF confinement system will mitigate any release that may occur within the CHF. The mitigation includes settling, plate out, and HEPA filtration.

Because the strategy does not rely on DOE SNF characteristics to demonstrate compliance with 10 CFR Part 63.111(b), no additional characterization will be required. The only DOE SNF requirement will be that DOE SNF be shipped in DOE standard canisters.

Section A-1 summarizes the NSNFP activities and products that provide the technical justification for the licensing strategy based on preventing a breach of a canister containing DOE SNF. An unplanned radiological release from canisters containing DOE SNF can be prevented by relying on engineered solutions that minimize challenges to the canister integrity (i.e., a drop) and preventing a canister breach in the event of a challenge. Implementation of this strategy for prevention of a drop and breach renders any preclosure event not to be credible.

Section A-2 provides an overview of NSNFP activities and information that support Beyond Category 2 consequence analyses. Although not directly relied on to support the licensing case, consequence analyses have nonetheless been performed for a breach, even though not credible. These Beyond Category 2 analyses quantify the available safety margin and provide additional insights into the risks associated with receiving and handling DOE SNF.

A-1. DOE SNF LICENSING CASE—PREVENTION OF RADIOLOGICAL RELEASE

A-1.1 Performance Allocation

Personnel at Yucca Mountain evaluated 27 design alternatives intended to reduce the risk associated with handling DOE standard canisters.[2] The evaluation focused on minimizing the possibility of breaching a canister, preventing and mitigating the design basis events, developing alternative handling concepts, considering cost and reliability issues, and other risks associated with system designs. The study concluded that the design alternatives that eliminate drop events exceeding the design bases of the canisters minimize the risk to workers, the public, and facility operations.

Because of the many design alternatives capable of preventing canister breach and the resultant benefits, system level requirements to preclude breach of a DOE standard canister were recommended. Additional trade studies were performed and documented in Reference 3. Eight specific facility design alternatives were evaluated. Four were eliminated based on cost/benefit considerations. Two of the remaining four were considered to be outside the control of the Civilian Radioactive Waste Management System program and were thus not considered viable scenarios to pursue. The remaining two alternatives were used to develop and recommend specific design criteria by allocating performance to the canister and canister handling systems.

Two additional activities demonstrated that facility and equipment design, coupled with canister reliability, could be successfully relied on to reduce the likelihood of a canister breach during the preclosure period to below the Beyond Category 2 threshold (i.e., <1E-4). The NSNFP estimated the canister critical flaw size and the likelihood of a flaw equal to or greater than the critical flaw size to go undetected during the fabrication and canister acceptance process.[4] The Yucca Mountain Project evaluated a range of canister and Canister Transfer System performance failure probabilities to determine their effect on the frequency of a radionuclide release.[5] The evaluation focused on a particular initiating event—a drop of a canister due to failure of a Canister Transfer System bridge crane, which was determined to be the bounding event for the Canister Transfer System. These evaluations confirmed that reasonably achievable performance requirements for the Canister Transfer System and DOE standard canisters could reduce the likelihood of a release to below the Beyond Category 2 threshold.

Based on the results of these activities, the recommended design criteria for the DOE standard canisters were translated into the Waste Acceptance System Requirements Document (WASRD).[6] The WASRD requires all DOE SNF that cannot be handled interchangeably with standard commercial pressurized water reactor or boiling water reactor fuels to be placed in DOE standard canisters. Other DOE standard canister requirements are specified including survivability requirements that, when coupled with the design controls imposed on canister handling equipment, will ensure that 10 CFR Part 63 limits are satisfied. The WASRD requirements, as written, allow canister acceptance based either on certification that a given dose would not be exceeded if breached or on certification that the probability of canister breach is Beyond Category 2.

These WASRD requirements essentially establish the design requirements for repository canister handling equipment and the survivability requirements for DOE standard canisters. Yucca Mountain Project canister handling system designs must ensure that the specified drops are not exceeded. And the DOE standard canister design must ensure that the likelihood of a breach, if dropped from within these limits, is sufficiently small that, when coupled with the likelihood of a drop, the combined probability renders the event not credible (i.e., Beyond Category 2).

A-1.2 DOE Standard Canister Design and Testing

In 1997, the NSNFP began the development of the DOE standardized canister concept. The objective was to have a canister into which DOE SNF would be placed and never reopened. The canister would be:

- Used within a facility (probably HEPA-filtered) when handled by itself
- Placed within a storage facility or cask during interim storage
- Placed within a transportation package for movement to the national repository
- Ultimately loaded inside a waste package at the repository for permanent disposal.

Canister requirements were based on preliminary waste acceptance criteria for the proposed repository. They included normal operating conditions (e.g., heat generation rates, internal pressure) and off-normal event conditions, which included. an accidental drop during canister handling. It was recognized early that for repository-defined normal operating conditions, the canister could be shown to meet the criteria of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code[7] due to its robust design. Additional analyses and testing have been performed by the NSNFP in order to demonstrate canister integrity under all credible scenarios during repository preclosure operations. This assumption was further refined during development of the licensing strategy for DOE SNF when the robustness of the standard canister was demonstrated. The licensing approach relies on the low probability of failure of the DOE standard canister during surface facility operations. Although conditions during interim storage and transportation are not completely known, based on industry experience, it is assumed that the canister can be designed to ensure repository performance objectives will not be compromised by storage and age-related degradation.

In a drop event, the material stresses were expected to exceed the ASME Level D stress limits. It was also expected that the canister components made of ductile materials (e.g., stainless steel) could experience significant plastic deformation before rupture would occur. During 1998 and 1999, the NSNFP performed a number of drop tests of test specimens and canisters to show that the DOE standardized canister could maintain containment during a variety of accidental drop events. These tests were also used to validate the use of finite element analytical methods in predicting canister response during drop events and to calculate the material deformations and strains. The following sections will show that considerable plastic straining of the canister stainless steel components occurred without a single breach during the testing. These sections also provide:

- A background on the development of the DOE standard canister concept
- A general description of the DOE standardized canister design configuration, the ASME calculation performed on the DOE standardized canister under drop conditions
- A discussion on the drop testing of small and full-scale 304L stainless steel specimens performed in 1998
- A summarization of results of the drop testing of nine DOE standardized test canisters under a variety of drop conditions
- A summary of what these results mean
- An overview of the current status of the DOE standardized canister and multi-canister overpack (MCO) development activities.

A-1.2.1 Background of the Development of DOE Standard Canisters

In November and December 1995, a working group was established to develop a path forward for dispositioning DOE SNF at the Savannah River Site. During subsequent discussions, which included technical staffs from the three major DOE sites (Hanford, the Idaho National Engineering and Environmental Laboratory [INEEL], Savannah River Site), DOE-Office of Environmental Management (EM) Headquarters staff, and the Yucca Mountain Site Characterization Office (YMSCO), a proposed DOE high-level waste (HLW) waste package was presented to the participants. This waste package was shown with five 24-in.-diameter HLW canisters clustered in a circle, with an open space in the middle. The working group proposed that DOE SNF could be disposed in the center location with little added cost. This configuration would also facilitate criticality safety. NSNFP and EM management recognized the potential for reducing the EM disposal costs and reducing handling costs at both the DOE sites and the repository. In a video conference held on January 23, 1996, DOE and contractor participants discussed the formation of a Standard Canister Working Group. A charter and mission statement for this group was discussed, and a kickoff meeting was organized.

The first Working Group meeting was held on April 14, 1996, in Atlanta, Georgia. INEEL, Oak Ridge National Laboratory, and YMSCO representatives presented their proposed plans for packaging DOE SNF for disposal. The YMSCO presentation was particularly significant because it identified a canister that was proposed for placement of DOE SNF in the center position of the HLW waste package as had been discussed at the January 23 video conference (see Figure A-1). The available space was a little over 17 in. in diameter and either 120 or 180 in. long. For the remainder of 1996, the concept of codisposal of DOE SNF with HLW was refined, and initial total system performance assessment and criticality analyses were performed. No identifiable issues were raised and subsequently in February 1997, the NSNFP formally initiated a DOE standardized canister development work task.

In 1998, the NSNFP worked with the YMSCO staff to increase the waste package diameter slightly to allow the use of standard 18-in. diameter pipe for the DOE standardized canister. A large diameter (24-in.) standardized canister was not an issue because it was the same diameter as the HLW canister. A study was conducted that concluded all DOE SNF could be disposed in either the 18-in. or 24-in.-diameter DOE standardized canister having either 10 or 15-ft lengths. The proposed DOE SNF canister design has been baselined in the repository program and has been used in all analyses including

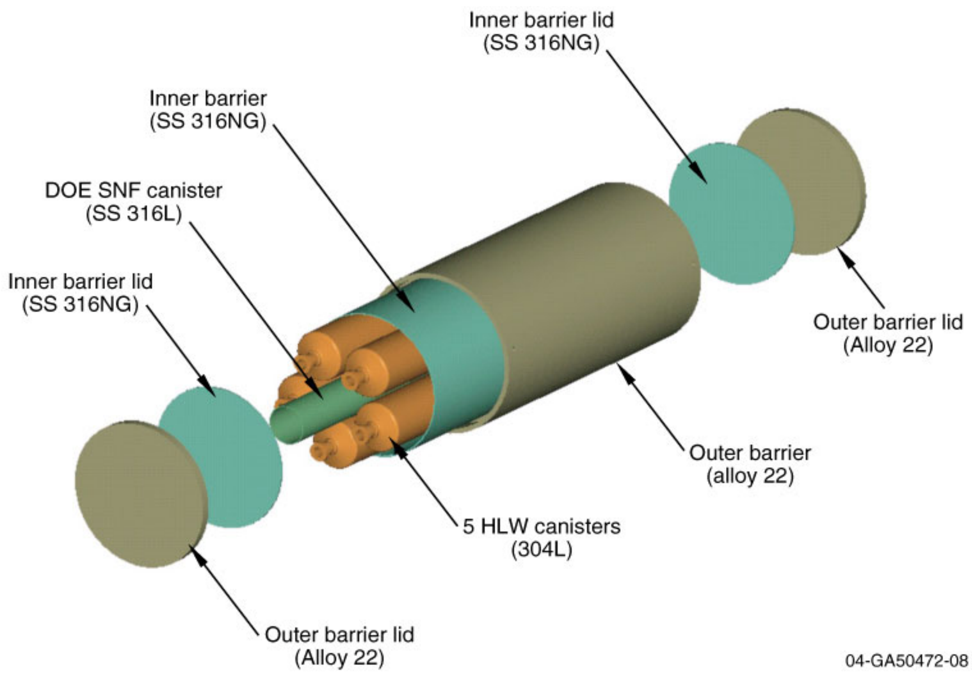

Figure A-1. DOE HLW/DOE SNF waste package.

the final environmental impact statement and site recommendation. Currently, the INEEL, Savannah River Site, and Hanford (except for fuels to be placed in MCOs) have all indicated their intent to use DOE standardized canisters for packaging of their SNF for disposal. Moreover, Argonne National Laboratory has plans to use the DOE standardized canister for packaging the HLW from their electrometallurgical treatment process. The decision to use the DOE standardized canister will limit the handling procedures and handling tools required at the repository for all DOE SNF types.

In parallel, Hanford was developing a MCO to repackage N-reactor fuel and move it from water-filled basins near the Columbia River to a dry storage facility. The MCO was initially designed for interim storage to satisfy a near-term need to move the N-reactor fuel. The NSNFP performed an early review of the MCO and determined that a minor change to the design of the MCO internal baskets would increase the probability that it could be transported in a horizontal configuration to the repository at a later date. The NSNFP has demonstrated that the MCO can meet the operational loads and the identified accidental drops at the repository surface facility and will be analyzing the MCO for transportation loads in the near future.

A-1.2.2 DOE Standardized Canister Design

The DOE standardized canister design resulted in four unique sizes.

1. 18-in. (457-mm) diameter, 3/8-in. (9.53-mm)-thick walls, 10-ft (3.05-m) total length, 5,005-lb (2,270-kg) total weight

2. 18-in. (457-mm) diameter, 3/8-in. (9.53-mm)-thick walls, 15-ft (4.57-m) total length, 6,000-lb (2,722-kg) total weight

3. 24-in. (610-mm) diameter, 1/2-in. (12.7-mm)-thick walls, 10-ft (3.05-m) total length, 8,996-lb (4,081-kg) total weight

4. 24-in. (610-mm) diameter, 1/2-in. (12.7-mm)-thick walls, 15-ft (4.57-m) total length, 10,000-lb (4,536-kg) total weight.

A skirt with a lifting ring is attached to each canister end. Canister shells, dished heads, skirts, and lifting rings are made of either SA-312 or SA-240 316L stainless steel. Figures A-2 and A-3 show this design for the 18-in. (457-mm)-diameter canister.

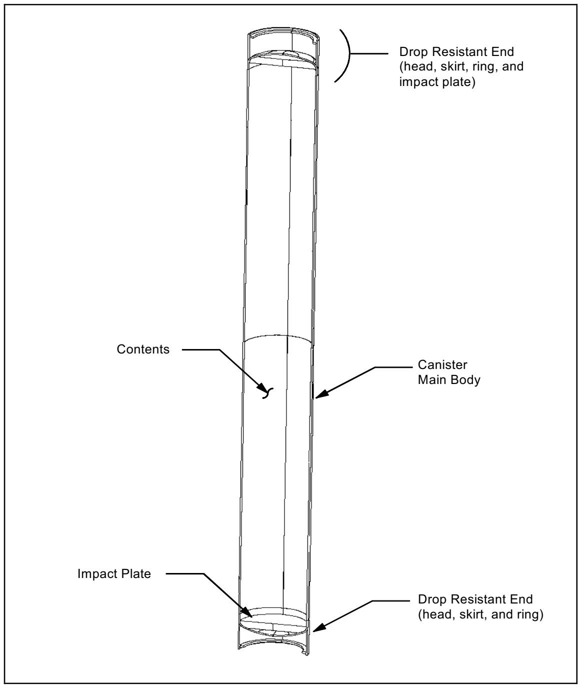

Figure A-2. DOE standardized canister overall design (section view).

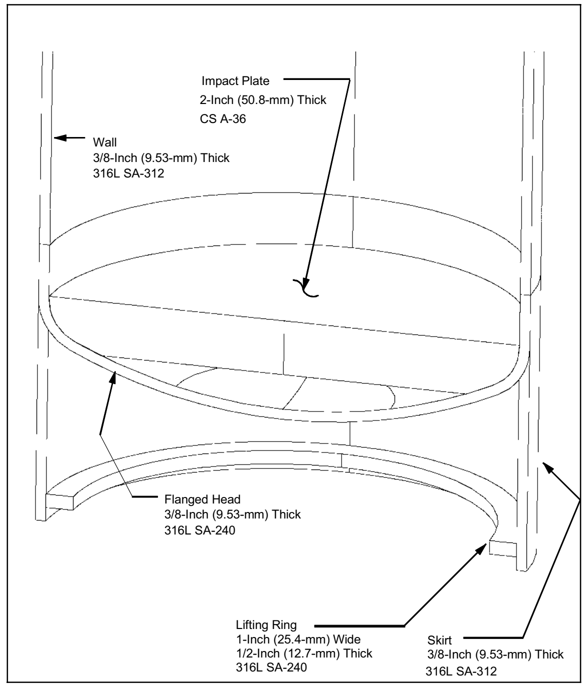

Figure A-3. 18-in. (457-mm) canister lower end (section view).

A-1.2.3 ASME Code Calculations—Allowable Drop Height

Drop analyses of a capacity-loaded 18-in. (457-mm)-diameter, 15-ft (4.57-m)-long, DOE standardized canister were performed to determine the maximum height that the canister could be dropped under the most unfavorable orientation such that the containment boundary would meet the requirements of the ASME B&PV Code, Section III. These analyses were detailed in an internal memo.[8]

First, the allowable stress levels from the ASME B&PV Code, Section III, Appendix F, "Rules for Evaluation of Service Loadings with Level D Service Limits," (see Reference 7) were determined. The general primary membrane stress intensity limit for the canister material (316L stainless steel) was 44.8 ksi (309 MPa), and the maximum primary membrane plus bending stress intensity limit was 57.6 ksi (397 MPa).

Next, the worst-case drop angle was determined to be 10 degrees off horizontal (canister impacting a rigid flat surface). At this orientation, the DOE standardized canister would first impact the bottom end, then rotate to "slap down" the top end. A finite element model was used to calculate the stresses in the canister starting with a 10-ft (3.05-m) drop. The resulting stresses did not meet the acceptance criteria, so successive runs were conducted until a 1-ft (0.3-m) drop height case was reached. At this point, the acceptance criteria were met for the bottom-end impact, but were still exceeded for the subsequent top-end slapdown. Specifically, the average stress across eight equally spaced sections through the thickness was 49.8 ksi (343 MPa) (greater than the 44.8 ksi [309 MPa] allowed) with an outer surface peak of 67.7 ksi (467 MPa) (greater than the 57.6 ksi [397 MPa] allowed), occurring in the top head at the knuckle. (Because the drop event is a rapidly occurring dynamic event, these stress levels occurred for only a few milliseconds.)

It was concluded that the ASME Code methodology of stress limits inadequately addressed canister accidental drop events. This was not surprising because the ASME Code methodology is based on situations where pressure, mechanical, or thermal loads are applied in a static or slowly varying multiple-second timeframe, and not on dynamic impact cases where the loads are applied for a few milliseconds and then never repeated. Drop events are better defined by strains.

A-1.2.4 Summary of 1998 Testing of Small and Full-Sized Standardized Canister Specimens

Eleven test specimens were drop tested in 1998 by the NSNFP at the INEEL. These drop tests are summarized in an INEEL letter report (see Reference 8).

A-1.2.4.1 5-Inch-Diameter Drop Test Specimens

Six 5-in. (127-mm)-diameter test specimens were constructed using approximately 3-ft (0.91-m)-long, thin walled tubes. A flat plate was welded in the top and bottom of each tube to represent heads. A relatively thick interior plate was welded 3 in. (76.2 mm) from the bottom head to support the contents, which consisted of lengths of No. 4 rebar. All specimen materials, excluding the interior rebar, were 304L stainless steel. The weight of each test specimen was just over 110 lb (50 kg). These test specimens were initially oriented at 15 degrees off vertical, and were dropped at heights of 10 ft (3.05 m), 15 ft (4.57 m, 2 specimens), 20 ft (6.10m), 25 ft (7.62 m), and 30 ft (9.14 m) onto a “rigid” surface (2-in. [50.8-mm] steel plate on a thick concrete pad). The test and analytical results are summarized as follows:

- The specimen damage was confined to the volume between the bottom head and the contents support plate. The bulk of the deformation occurred in the tube wall as a single outward bulge.

- Finite element models (using ABAQUS/Explicit Version 5.8-1) predicted the deformed shape of the dropped specimens very well. The calculated peak equivalent plastic strain levels increased with drop height, reaching a maximum of 86% on a surface and 26% at mid-thickness (giving an average of about 50% through the wall) on the highest drop. Pressure testing of all specimens after drop testing indicated that the pressure boundary had been maintained (25 psig [172 kPa] pressure was held steady for 1 hour without loss).

- A comparison of drop height versus resulting equivalent plastic strains in the test specimens showed that large increases in drop height (5-ft [1.52-m] increments) corresponded to small increases in material strains (about 10% peak strain per increment). Figure A-4 illustrates the strain results of the 5-in. (127-mm)-diameter drop tests versus drop height.

A-1.2.4.2 18-Inch-Diameter Short Drop Test Specimen— One 18-in. (457-mm)-diameter short test specimen was constructed using a 5-ft (1.52-m)-long thin wall tube. A flat plate was welded to the tube top to form the top head. An 18-in. (457-mm)-radius shallow head was inserted 7 in. (177.8 mm) into the bottom of the tube and welded in place. The contents consisted of No. 4 rebar and a lightweight interior structure. All specimen materials, excluding the contents, were 304L stainless steel. The total weight of the specimen was about 1,000 lb (453.6 kg). This specimen was oriented at 32 degrees

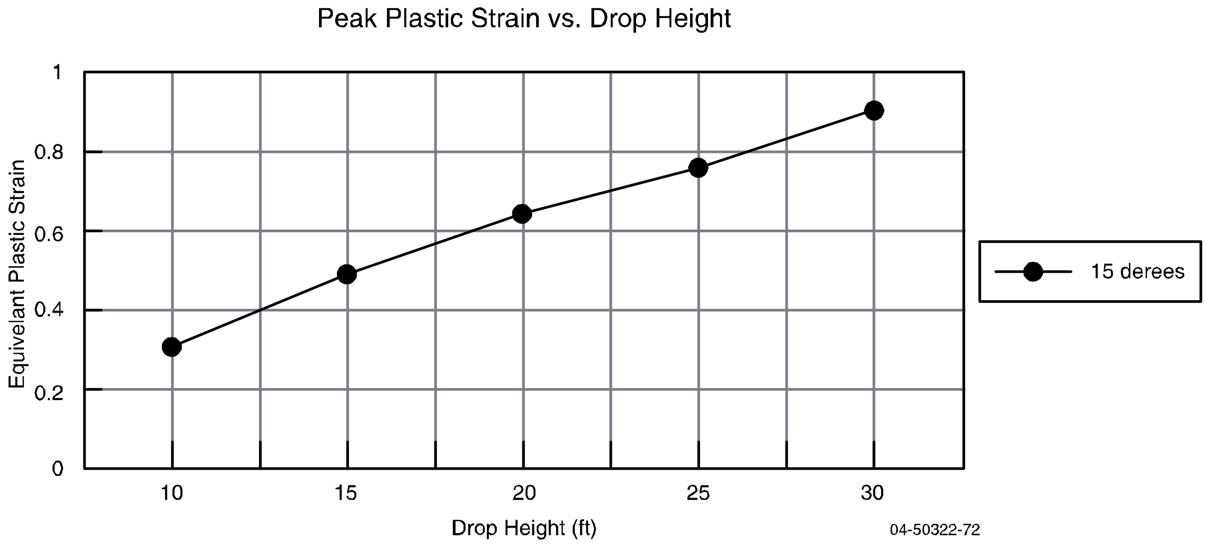

Figure A-4. 5-in. (127-mm) specimens, peak plastic strain versus drop height.

off-vertical at impact and was dropped from a height of 30 ft (9.14 m) onto a “rigid” surface (a 2-in. [50.8-mm] steel plate on a thick concrete pad). The test and analytical results are summarized as follows:

- The most significant specimen damage was confined to the skirt as expected. The skirt folded inward toward the lower head, bending the skirt walls.

- A finite element model of the specimen predicted the deformed shape very well. The peak equivalent plastic strain in the pressure boundary was calculated in the model at 13% on a surface and 3% at mid-thickness (giving an average of about 7% through the wall). Post-drop pressure testing showed that the pressure boundary had been maintained (25 psig [172 kPa] and held steady for 1 hour without loss).

A-1.2.4.3 Representative Drop Test Specimens—Two full-scale representative drop test specimens were constructed of 304L stainless steel, an 18-in. (457-mm)-diameter and a 24-in. (610-mm) diameter canister, both 15 ft (4.57 m) in total length. The contents of the representative test specimens consisted of rebar, steel sections filled with concrete, etc. The total loaded weight of the 18-in. (457-mm) test specimen was 5,690 lb (2,581 kg), while the 24-in. (610-mm) test specimen was 9,790 lb (4,441 kg). Each test specimen was oriented such that the center-of-gravity was over the impacting corner (6 degrees off-vertical for the 18-in. [457-mm] test specimen, 9 degrees off-vertical for the 24-in. [610-mm] test specimen) and dropped from a height of 30 ft (9.14-m) onto a “rigid” surface (a 2-in. [50.8-mm] steel plate on a thick concrete pad). The test and analytical results are summarized as follows:

- The visual damage to the representative test specimens was limited to the lifting rings and skirts on the impacting end of each specimen as expected.

- Pre- and postdrop test finite element models showed a good match in overall deformation shape to the actual dropped test specimens. The peak equivalent plastic strain in the pressure boundary of the 18-in. (457-mm) test specimen was calculated in the model at 3% on a surface and 1% at mid-thickness; giving an average of about 2% through the wall. The 24-in. (610-mm) test specimen had 4% on a surface and 1% at mid-thickness, giving an average of about 2% through the wall. Postdrop pressure testing showed that the pressure boundary had been maintained.

After the 18-in. (457-mm) test specimen was dropped at 6 degrees off-vertical, it was dropped 30 ft (9.14 m) again in a horizontal orientation. A postdrop pressure test showed that the pressure boundary was still maintained.

A-1.2.4.4 Puncture Drop Tests—Following the drop tests of the 18-in. (457-mm) and 24-in. (610-mm)-diameter representative drop test specimens discussed in the previous section and the subsequent pressure tests, these two test specimens were subjected to a drop onto a puncture bar. With the test specimen in a horizontal orientation, the puncture testing consisted of a 40-in. (1-m) drop through the air onto a 6-in. (152.5-mm)-diameter, 12-in. (304.8-mm)-tall, solid steel bar. The puncture bar was welded to a flat-horizontal, 2-in. (50.8-mm) thick, steel plate placed on a concrete pad. The test specimens were modified by removing the impacted end (from the previous drop tests) so that contents could be arranged to provide an empty volume where the puncture post would impact. The specimen end was not reattached for this test. The 18-in. (457-mm)-diameter test specimen weighed 5,775 lb (2,620 kg), and the 24-in. (610-mm)-diameter test specimen weighed 9,820 lb (4,454 kg) for this puncture drop test. Additional weights were added to the test specimens. The test and analytical results are summarized as follows:

- The results showed that the two test specimens did experience significant deformation from impact with the puncture post.

- The finite element models showed a good match in deformation shape to the test specimens after impacting the puncture bar. The peak equivalent plastic strain in the pressure boundary of the 18-in. (457-mm) test specimen was calculated in the model at 24% on a surface and 6% at mid-thickness, giving an average of about 14% through the wall, while the 24-in. (610-mm) test specimen had 15% on a surface and 7% at mid-thickness, giving an average of about 10% through the wall.

Because the test specimens' lower ends were removed after the previous drop testing, no pressure testing after the puncture drop was possible. However, visual examinations clearly indicated that there was no significant damage to the pressure retention capability of either test specimen.

A-1.2.4.5 Conclusions to 1998 Testing—The drop testing and finite element analyses on the 5-in. (127-mm) diameter specimens, the 18-in. (457-mm) diameter short specimen, and the 18-in. (457-mm) and 24-in. (610-mm) representative test specimens showed two important things. First, all the test specimens retained their pressure boundary after the drop testing even though the calculated material strains were higher than allowed by the ASME B&PV Code. Second, finite element methods using plastic analyses could accurately calculate the deformed shape of each test specimen.

A-1.2.5 Summary of 1999 Testing of Nine Representative Test Standardized Canisters

Nine representative test canisters (all 18-in. [457-mm] diameters) were drop tested in 1999 by the NSNFP at Sandia National Laboratories. These drop tests are summarized in an NSNFP report (see Reference 1).

All the representative test canister pressure boundary components (body and heads), skirts, and lifting rings were made of 316L stainless steel. A summary of the test canister configurations and intended impact orientations is given in Table A-3.

Figures A-5 and A-6 show the configuration of internal components in these representative test canisters.

Table A-3. Representative test canister configurations and orientations.

Canister No.	Length (ft)	Desired Impact Angle Off-Vertical (degrees)	Total Weight (lb)	Drop Height (ft)	Contents[a]
18-15-00-01	15	0	6033	30	S, S-W
18-15-06-02	15	6[b]	5948	30	S, S-W
18-15-90-03	15	90	5995	30	S, S-W
18-15-45-04	15	45	5995	30	S, S-W
18-15-80-05	15	80	5965	30	S, S-W
18-10-90-06	10	90	3802	30	HICs
18-10-90-07	10	90	2997	30	Shippingport
18-15-PW-08	15	0	5972	2	S-W
18-15-PP-09	15	90	6085	40 in.	S-W

1 ft = 0.3 m, 1 lb = 0.454 kg

a. S = sleeve, S-W = spoked wheel divider. HICs and Shippingport fuel were simulated. Contents included rebar for all canisters.

b. Center-of-gravity-over-corner orientation.

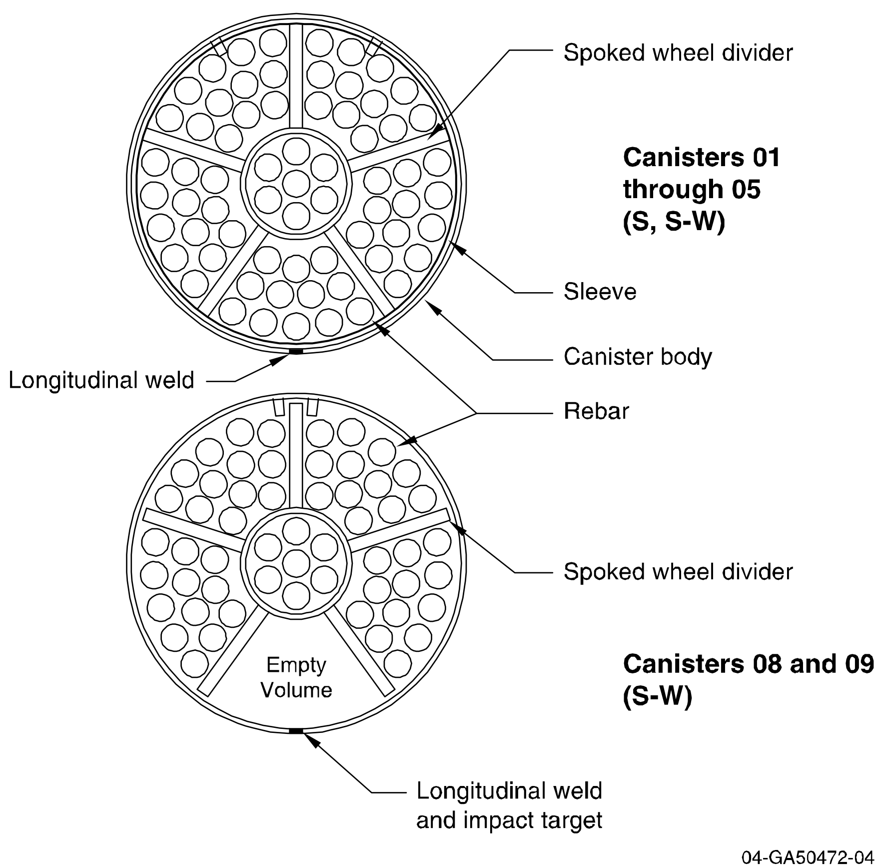

Figure A-5. Canister internal configurations.

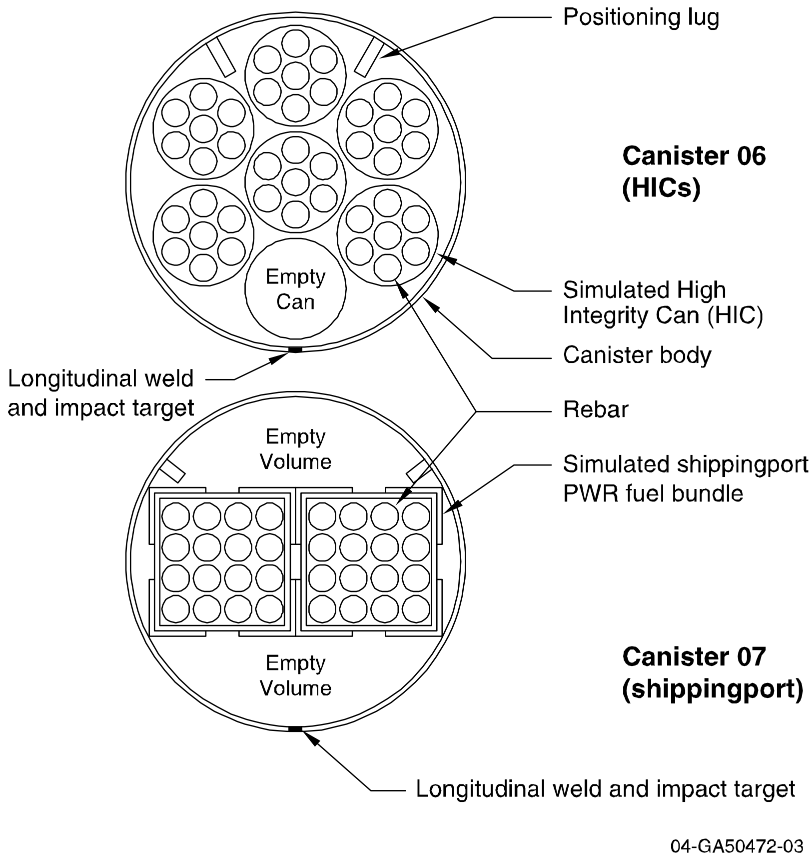

Figure A-6. Canister internal configurations.

The target at the drop test facility included a flat 4-in.-thick (at the thinnest location) steel plate imbedded in heavily reinforced concrete (about 2 million lb [910,000 kg] total weight). The design of the facility provided the desired "essentially unyielding surface."

Test Canisters 01 through 07 were dropped onto this flat surface from 30 ft (9.14 m). Test Canister 08 simulated a drop event onto a repository waste package. This was simulated by dropping the canister 24 in. (609.6 mm) onto a vertically oriented 2-in. (50.8-mm)-thick plate. Because the canister was not centered over the plate, it then rotated to impact another vertically oriented 2-in. (50.8-mm)-thick plate set 78 in. (1,981 mm) away (on the other side of the waste package). Test Canister 09 was dropped from 40 in. (1 m) onto a 6-in. (152.5-mm)-diameter steel bar welded to the steel surface. The drop heights and puncture bar dimensions were chosen to follow that specified by the Code of Federal Regulations (CFR)[9] for transportation packages.

The test and analytical results are as follows:

- The finite element models accurately predicted the actual deformed shape of the test canisters after the drop events. An example of this is shown in Figures A-7 and A-8.

The calculated material strains in the representative test canister components are summarized in Table A-4.

- An average through-wall thickness strain of about 34% corresponds to Canister 05 (highest containment boundary strain experienced by test canisters) in Table A-4.

Figure A-7. Test Canister 04 deformed end.

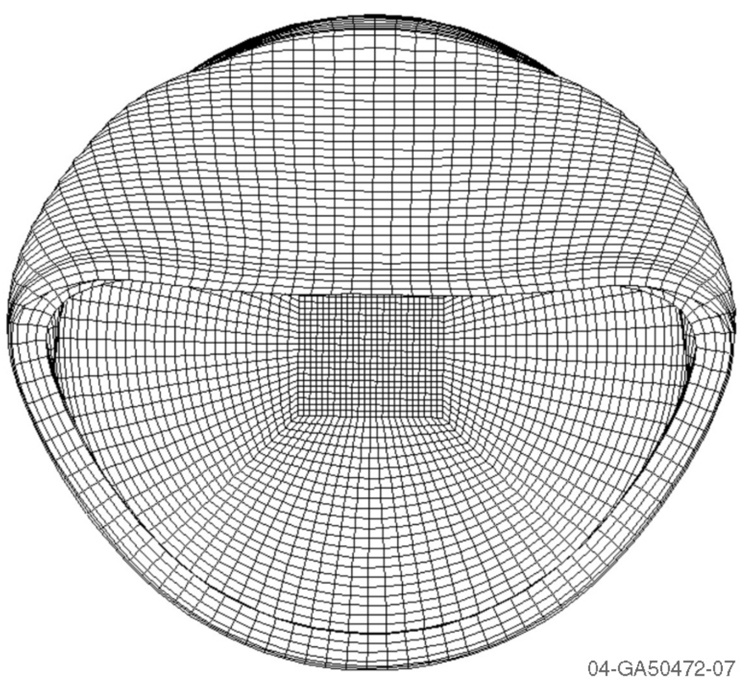

Figure A-8. Test Canister 04 deformed finite element model.

Table A-4. Calculated peak equivalent plastic strains.

Representative Test Canister	Peak Equivalent Plastic Strain (%)					
	Pressure/Containment Boundary Components			Skirts and Lifting Rings		
	Outside	Middle	Inside	Outside	Middle	Inside
18-15-00-01	7	3	6	91	17	75
18-15-06-02	9	3	10	107	21	94
18-15-90-03	40	15	26	10	10	10
18-15-45-04	33	9	36	52	33	84
18-15-80-05	57	19	42	24	20	19
18-10-90-06	44	17	31	21	10	18
18-10-90-07	62[a]	22[a]	42[a]	11	10	10
18-15-PW-08	20[b]	7[b]	18[b]	38[b]	38[b]	38[b]
18-15-PP-09	39	14	40	—	—	—

a. Peak strains due to conservative modeling of internals as discussed in report. Actual peak straining estimated below that reported for Canister 18-10-90-06.

b. Reported peak pressure/containment boundary strains due to impact with second vertical plate. Peak skirt and lifting ring strains due to impact with first vertical plate.

- After the nine test canisters were drop tested, each canister was pressurized to 50 psig (345 kPa) with air. In every case the test pressure remained constant—no loss in pressure—for the 1 hour monitoring period. This showed that the pressure boundary had been maintained for all representative test canisters after the drop tests.

- Four of the test canisters, including the highest strained canister (05) and three other significantly strained test canisters representing a variety of impact angles, were helium leak tested after the drop and pressure testing. Test Canisters 01, 04, 05, and 09 were helium leak tested and found to have a maximum leak rate of less than 1×10^{-7} std. cc/sec. This was considered leaktight according to ANSI N14.5[10] criteria and additional proof that the containment was maintained.

In addition, after the pressure and leak tests were completed on the nine canisters, one nondestructive examination was performed. Test Canister 18-15-06-02, which was the canister that was dropped with its center-of-gravity over the impacting corner, was inspected using liquid penetrant methods to check for cracks. The location that was checked was on the outside surface of the lower end skirt where considerable deformation had occurred during the drop event (see Figure A-9). The calculated strains were 107% at the outside surface, 21% at the mid-plane, and 94% at the inside surface (giving an average through wall strain of about 60%) as reported in Table A-4. The liquid penetrant test results showed no indications of cracks on the lower skirt outside surface.

Figure A-9. Location of liquid penetrant test on representative Test Canister 02.

A-1.2.6 Summary of Standardized Canister Drop Testing Results To Date

The 1998 and 1999 testing performed by the NSNFP on the DOE standardized canister project has shown that significant plastic straining of 304L and 316L stainless steels can occur during an accidental drop event without losing containment.

With regard to high plastic strains on the pressure/containment boundary, the 5-in. (127-mm) test cans were calculated to experience a peak strain of 86% (on the surface) and an average through wall strain of about 50%, where the 18-in. (457-mm) representative test canister (Canister 07) experienced a 57% peak and a 34% average. In addition, an 18-in. (457-mm) representative test canister (Canister 02) skirt was shown to not experience outer surface material cracking under a 107% peak surface strain (with an average through wall strain of about 60%).

The 1998 and 1999 tests were performed with all specimens at ambient temperatures (13–29°C [55–85°F]). The DOE standard canisters may be anywhere from 10 to 149°C (50 to 300°F) when handled alone (outside of another container). Because the DOE standardized canisters are made of 316L stainless steel, their ductility at 149°C (300°F) will be greater than at 10°C (50°F). This means that a canister is expected to deform more for an accidental drop at 149°C (300°F) than it would if the canister was at 10°C (50°F). However, the canister material could strain more at 149°C (300°F) before rupturing than it could at 10°C (50°F). Therefore, the strain acceptance criterion is valid for an accidental canister drop at 10°C (50°F) and is valid (though conservative) for a canister drop at 149°C (300°F). Canister temperatures may be much higher or lower only when within another container (e.g., storage container, transportation cask, or waste package).

The 1999 drop testing of representative DOE standardized test canisters was performed to 30-ft (9.14-m) drop heights, exceeding those defined for the repository (23-ft [7-m] vertical drop, and a 2-ft [0.61-m] drop in any orientation). Therefore, these drop heights result in higher material strains than repository drops would yield.

The 1999 drop testing effort confirmed the ability of an 18-in. (457-mm)-diameter DOE standardized canister, in an essentially flaw-free condition, to maintain a leaktight containment after an accidental drop event. In 2002 and 2003, the NSNFP evaluated the ability of a standard canister with a flaw to maintain containment after an accidental drop event at the repository. The maximum anticipated acceptable flaw size was determined using analytical fracture mechanics methods and past metallurgical test results. This maximum anticipated acceptable flaw size was calculated using worst-case stress values from repository-defined drop events for an 18-in. DOE standardized canister. The result was an elliptical-shaped flaw of significant size, approximately 2 in. (50.8 mm) long and 0.20 in. (5.1 mm) deep (see Reference 4). This size of flaw or crack, which is more than halfway through the nominal 0.375-in. (9.53-mm) canister wall thickness, can be easily detected using a variety of nondestructive examination techniques.

A-1.2.7 Current Analysis Efforts on 24-Inch Standardized Canisters and Multi-Canister Overpacks

In 2003, the NSNFP analyzed the 24-in. canister being developed by Foster-Wheeler Environmental for the Spent Nuclear Fuel Dry Storage Project, the 24-in.-diameter DOE standardized canister, and the MCO for drop events similar to those performed in 1999 with the 18-in. standardized canister. The MCO, because it was not developed to meet the 10 CFR Part 71.73c criteria like the DOE standardized canister, was analyzed for drop events identified in the WASRD. Specifically, this evaluated the MCO for a vertical drop of 23 ft and a 2-ft worst-orientation drop. All the analyses performed to date indicate that the peak strains will be within the acceptable limits as validated during the 1999 testing. Three NSNFP reports[11–13] have been issued to document the results of these evaluations. Demonstration of the drop survivability of the 24-in. canisters and the MCOs occurred during 2004. Four drop tests were conducted to compare the analytical models and predicted canister performance against the actual drop test results. To date, no breaching of the canisters occurred during the drop tests, and deformations are within the envelope predicted by the analysis.

A-1.3 Preclosure Safety Strategy Summary

Based on the cost and safety advantages and the technical feasibility presented in the work cited above, a strategy that relied on engineering solutions (i.e., canister and canister handling equipment) to prevent any radiological release was selected as the preferred strategy. During the summer of 2002, specific activities and associated responsibilities for implementation of this strategy were identified during a series of workshops with the Yucca Mountain Project personnel and are documented in a workshop agreement.[14]

NSNFP activities supporting the conclusion that DOE standard canisters will withstand the identified drop scenarios are summarized in Reference 15. This report concludes that, as designed, the canister will not breach if dropped within its design basis. It also concludes that, based on inspection detection thresholds and the minimum critical flaw size, undetected material and weld fabrication flaws will not jeopardize canister integrity. The potential affects of age-related degradation that could occur prior to transporting the canisters to the repository for acceptance are also discussed, and a probabilistic estimation of a failure due to age-related degradation is provided.

The internal hazards analysis[16] prepared by the Yucca Mountain Project personnel identified potential drop and collision hazards during preclosure receipt and handling operations at the Monitoring Geologic Repository (MGR). The likelihood of a release associated with each potential event was also evaluated. All identified event sequences were determined to be Beyond Category 2.[17] Each document is presently being updated to support the license application and to identify controls that will implement the strategy for ensuring that any radiological release from DOE standard canisters during repository preclosure operations will be Beyond Category 2.

A-2. ADDITIONAL MARGIN—BEYOND CATEGORY 2 CONSEQUENCE CALCULATIONS

In addition to providing a robust canister to ensure that any radiological releases from DOE SNF is Beyond Category 2, consequence analyses have been performed in order to provide risk insights associated with handling DOE SNF at the proposed repository. Using conservatively estimated properties, these analyses calculated the doses that could affect an individual located on the MGR site boundary due to a hypothetical release associated with breach of a DOE standard canister, even though no credible (i.e., nonmechanistic). These calculations and their bases are summarized below.

A-2.1 Calculation Inputs and DOE SNF Grouping for Dose Consequence Analyses

Dose calculations require a source term to represent the release and MGR-specific transport and mitigation parameters. For each canister, the source term is the product of the material at risk, the damage ratio, the canister leak path factor, the airborne release fraction, and the respirable fraction.

The material at risk is the total quantity of radiological material that is available to be acted upon by the event. For canister drop events, the material at risk is the total radiological inventory of the canister. DOE SNF radiological inventories have been conservatively estimated. A discussion of these estimates is provided in Appendix D of this report.

The damage ratio is the fraction of the material at risk that is actually affected by the event. The damage to the canister and its contents is most likely to be localized. And the canister, cladding, and other structural material will absorb energy and thus buffer the SNF from the full impact of the event. Nonetheless, the consequence calculations conservatively assume that the entire material at risk is affected, and thus a damage ratio of one is used.

The canister leak path factor ($LPF_{canister}$) is the fraction of the radionuclides that escape the canister. For the screening dose analyses, no credit was taken for the canister (i.e., $LPF_{canister}$ was conservatively assumed to be 1.0). For the Beyond Category 2 calculations, a nonmechanistic breach is assumed. In this case, the $LPF_{canister}$ is assumed to be 0.1. This is also considered conservative. Data cited in DOE-HDBK-3010-94[18] support the use of 0.05 as a conservative value of the $LPF_{canister}$ for a failed canister containing a fine powder. Further, the primary mechanisms for rupture involve only cracking.

Airborne Release Fractions (ARFs) and Respirable Fractions (RFs) may be unique to each waste form and were determined by binning DOE SNFs into groups that shared properties that could be used to conservatively estimate ARFs and RFs. This process is discussed below.

DOE-HDBK-3010-94, "Airborne Release Fractions/Rates and Respirable Fractions for NonReactor Nuclear Facilities," (see Reference 18) provides ARFs and RFs for various material types and forms under a variety of accident conditions. Because the identified hazards during the repository preclosure period are associated with drops or other impacts (i.e., collision/crushing forces), DOE SNFs were grouped into three categories with distinctive behaviors relative to these types of events. Each of these three categories was further subdivided to account for the condition of the fuel prior to the event (i.e., intact or not intact). A more comprehensive discussion of the basis for the grouping is provided in Reference 19.

Fuels were binned, and ARFs and RFs were conservatively assigned to the fuels in each bin. For example, metallic and other fuels that may be potentially chemically reactive were assumed to oxidize completely. Fuels with cladding condition poor or unknown were considered to be 100% powder. A very conservative correlation was used for intact fuels expected to fragment upon impact. Conservative parameters, such as a 10-m drop height and maximum density, were used in the correlation. In addition, no credit was taken for the additional confinement provided by fuel cladding or by packaging that may occur prior to placement in the canister. The design basis event fuel groupings, ARFs and RFs, and preliminary radionuclide inventory estimates were provided to the Yucca Mountain Project with Reference 19.

A-2.2 Screening Dose Analysis

The preliminary source term information provided with Reference 19 was used to develop preliminary dose calculations in the DOE SNF Screening Analysis.[20] Using extremely conservative assumptions (below), this calculation concluded that ~90% of DOE SNFs would not cause doses in excess of 10 CFR 63 Category 2 thresholds.

These conservative assumptions were used in screening the dose analysis.

- No credit for canister retention ($LPF_{canister}$ = 1)
- 100% of the material at risk is impacted by the event (damage ratio = 1)
- No credit for energy absorption by the canister, cladding, or other structural materials
- No credit for confinement within cladding
- ARFs and RFs determined very conservatively:
  - All fuels with degraded cladding were assumed to be 100% particulate
  - 100% oxidation assumed for all fuels that could react with potential for pyrophoric reaction
  - A bounding drop height of 10 m was used to conservatively estimate the pulverization factor used to determine the ARFs and RFs for nonmetal, intact fuels.
- No credit for deposition or filtration within the facility or along the path to the site boundary ($LPF_{facility}$=1)
- Dose was calculated for a presumed 5 km site boundary (boundary is currently at 11 km).

The scoping analyses provided an indication of the bounding consequences associated with a nonmechanistic breach of a DOE standard canister. The analyses showed that, even when using very conservative assumptions, the majority of DOE SNF would not exceed Category 2 dose limits.

A-2.3 Beyond Category 2 Consequence Analyses

Because event sequences resulting in canister breach are Beyond Category 2, dose calculations are not necessary. Nonetheless, to gain additional risk insights, specific fuels were selected (from those that showed the potential to exceed limits) for more detailed analyses. After a review of the preclosure safety strategy and the results of the screening dose analysis,[21,22] the N-reactor fuel and the Shippingport Light Water Breeder Reactor (LWBR) fuels were selected for further analysis. These fuels were selected because they were among the potential bad actors identified by the screening dose analysis and because they represent a significant quantity of fuel. Sufficient fuel-specific information is available to support a more accurate calculation.

Although still conservative, fuel-specific inputs were obtained for the selected fuels in order to more accurately represent the expected conditions and performance of the fuel, the canister, and the repository systems that mitigate the release. These included:

- Accounting for the actual fraction and size distribution of preexisting particulate rather than using a bounding assumption of 100% powder
- Accounting for a level of confinement provided by the canister by using an $LPF_{canister}$ of 0.1, including an LPF to acknowledge the presence of a HEPA filtration system
- Acknowledging that slow oxidation is much more likely than ignition of the bulk uranium metal for N-reactor fuel
- Accounting for atmospheric dispersion over the distance of 11 km rather than 5 km to the site boundary.

Calculations were performed both with and without HEPA filtration and for both the 50% and 99.5% max sector acute atmospheric diversion factor (χ/Q).[23] Results are summarized in Tables A-5 and A-6.

As shown by these results, doses from DOE SNF are expected to be well below regulatory limits.

The screening dose analysis calculated doses for all DOE SNF using preliminary source term information and extremely conservative assumptions. For the selected DOE SNFs, the Beyond Design Basis Event dose calculations used the same preliminary source term information but with more realistic assumptions related to mitigation provided by the canister and the repository systems. A dose consequence analyses is currently being prepared that will calculate the dose for all DOE SNFs using current source term information and the more realistic assumptions (presently being prepared by the RW management and operations contractor).

Table A-5. Beyond design basis event dose results without HEPA filtration.

Fuel	Fuel Type	Fuel Condition	TEDE (rem)	Highest of CDEs+DDE (rem)	SDE (rem)	LDE (rem)
10 CFR 63.111 Limit			5	50	50	15
Shippingport LWBR Scrap (SNF ID #377)	Nonmetal	Not intact				
50% max sector acute χ/Q			1.58E-2	1.29E-1	2.57E-5	1.58E-2
99.5% max sector acute χ/Q			6.96E-2	5.69E-1	1.13E-4	6.97E-2
N-reactor (SNF ID #147)	Other	Not intact				
50% max sector acute χ/Q			2.64E-3	2.21E-2	7.89E-5	2.72E-3
99.5% max sector acute χ/Q			1.16E-2	9.75E-2	3.48E-4	1.20E-2

Table A-6. Beyond design basis event dose results with HEPA filtration.

Fuel	Fuel Type	Fuel Condition	TEDE (rem)	Highest of CDEs+DDE (rem)	SDE (rem)	LDE (rem)
10 CFR 63.111 Limit			5	50	50	15
Shippingport LWBR Scrap (SNF ID #377)	Nonmetal	Not intact				
50% max sector acute χ/Q			7.59E-5	1.11E-4	2.65E-5	1.02E-4
99.5% max sector acute χ/Q			3.35E-4	4.90E-4	1.13E-4	4.48E-4
N-reactor (SNF ID #147)	Other	Not intact				
50% max sector acute χ/Q			1.45E-3	1.53E-3	7.84E-5	1.53E-3
99.5% max sector acute χ/Q			6.40E-3	6.76E-3	3.46E-4	6.75E-3

A-3. REFERENCES

1. U.S. Department of Energy Environmental Management, "FY 1999 Drop Testing Report for the Standardized 18-inch DOE SNF Canister," EDF-NSNF-007, Revision 2, 2002.

2. U.S. Department of Energy, *DOE Spent Nuclear Fuel Canister Handling Trade-Off Evaluation*, BA-0000000-01717-2200-00013, August 1999.

3. U.S. Department of Energy, Office of Civilian Radioactive Waste Management, *Canister, Transportation, and Monitored Geologic Repository Systems, Structures, and Components Performance Allocation Study*; TDR-CRW-SE-000004, December 2000.

4. U.S. Department of Energy Environmental Management, *Conditional Probability Estimate for a Standardized DOE SNF Canister Drop Given a Breach*, DOE/SNF/REP-065, December 2000.

5. U.S. Department of Energy, "Canister Transfer System Event Sequence Calculation," CAL-CTS-SE-000001, August 15, 2001.

6. U.S. Department of Energy Office of Civilian Radioactive Waste Management, *Civilian Radioactive Waste Management System Waste Acceptance System Requirements Document*, DOE/RW-0351, Revision 4, March 2002.

7. American Society for Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section III, Dir. 3, 1997.

8. A. G. Ware Letter to S. C. Gladson, INEEL, "ASME Code Analysis of Accidental Drop of 18-Inch Standardized DOE SNF Canister," AGW-01-00, June 1, 2000.

9. 10 CFR Part 71.73c, *Packaging and Transportation of Radioactive Material Subpart F—Package and Special Form Tests*, 1999.

10. American National Standards Institute, *American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment,* ANSI N14.5-1987.

11. R. K. Blandford, "Structural Response Evaluation of the 24-in. Diameter DOE Standardized Spent Nuclear Fuel Canister," EDF-NSNF-026, Revision 0, August 2003.

12. S. D. Snow, "Analytical Evaluation of the Idaho Spent Fuel Project Canister Accidental Drop Events," EDF-NSNF-027, Revision 0, September 2003.

13. S. D. Snow, "Analytical Evaluation of the MCO for Repository-Defined and Other Related Drop Events," EDF-NSNF-029, Revision 0, September 2003.

14. Preclosure Safety Case Implementation Workshop Agreement, signed by Phil Wheatley and Steve Cereghino, June 11, 2002.

15. U.S. Department of Energy Office of Civilian Radioactive Waste Management, *U.S. Department of Energy Spent Nuclear Fuel Canister Survivability*, 000-PSA-WHS0-00100-000 Revision 00, June 2004.

16. Bechtel SAIC Company, *Internal Hazards Analysis for License Application*, Revision 00A, BSC 2003, ACC# ENG.20040623.0032, June 23, 2004.

17. Bechtel SAIC Company, *Categorization of Event Sequences for License Application*, Revision 00A, ACC# ENG.20040722.0001, July 22, 2004.

18. U.S. Department of Energy, *Airborne Release Fractions/Rates and Respirable Fractions for NonReactor Nuclear Facilities*, DOE-HDBK-3010-94, 1994.

19. P. D. Wheatley (INEEL) to John Clouet, "National Spent Nuclear Fuel Program (NSNFP) Data Support For Design Basis Event (DBE) Analysis," PDW-02-99, February 4, 1999.

20. U.S. Department of Energy Office of Civilian Radioactive Waste Management, DOE SNF Screening Dose Analysis, BBA000000-01717-0210-00047, Revision 00, May 18, 1999 (MOL19990602.0176).

21. J. S. Clouet to P. D. Wheatley, "Beyond Design Basis Event Meeting Summary," April 14, 2000.

22. Jofu Mishima to Richard Morissette, cover letter transmitting letter report entitled "Considerations in Evaluating Source Terms for Beyond Design Basis Events for the Waste Handling Building Monitored Geologic Repository," June 30, 2000.

23. U.S. Department of Energy, "DOE SNF BDBE Dose Calculations," CAL-SPS-SE-000006, Revision 00, May 23, 2001.

Appendix B

Postclosure/TSPA Activities

Appendix B

Postclosure/TSPA Activities

Like preclosure, the performance objectives of 10 CFR 63.113(b) specifically relate to the protection of the offsite public. Like the commercial spent nuclear fuel (SNF), the U.S. Department of Energy (DOE) SNF postclosure licensing strategy relies on the combinations of engineered and natural barriers to meet performance objectives of 10 CFR 63.113(b). The barriers prevent or substantially reduce the movement of water and prevent or substantially reduce the release of radionuclides from the waste for the 10,000-year regulatory period. Table B-1 identifies the systems, structures, and components (SSCs) and barriers for this strategy.

There are currently no waste acceptance criteria for DOE SNF, and the strategy does not result in additional characterization requirements. It is expected that the base case bounding analyses will demonstrate that the performance objectives are not exceeded. Thus there will still be no characterization requirements for DOE SNF. Therefore, the consequences are directly related to the radionuclide inventories. No credit is taken for cladding or fuel matrix in the DOE SNF release model, so there are no characterization requirements for these parameters.

The performance objectives established by 10 CFR 63.113(b) are summarized in Table B-2.

In consideration of the above performance objectives, the National Spent Nuclear Fuel Program (NSNFP) postclosure/total system performance assessment (TSPA) technical activities try to answer three basic questions.

1. Will the addition of DOE SNF into the repository change any of the repository's features, events, and processes (FEPs)?

2. Will the affected FEPs have a greater than 1 chance in 10,000 of occurring in 10,000 years, or will excluding the affected FEPs significantly change the radiological exposure or radionuclide release?

Table B-1 SSCs and barriers for postclosure waste isolation.

SSC and Barriers	Category	Description
Upper natural barriers	Important to waste isolation	The topography and surficial soils and the unsaturated zone above the repository prevent or substantially reduce the movement of water to the drift.
Waste package	Important to waste isolation	The waste package prevents or reduces the water contact with the waste form and confines the waste. Provides dry inert environment until waste packages breach to delay onset of waste form degradation.
Other engineered barriers: drip shield, waste form, cladding, emplacement plug	Important to waste isolation	Even after waste package failure, the other engineered barriers will continue to prevent or reduce the release of radionuclide from the waste. (Note: For DOE SNF, waste form and cladding credit are not currently taken in the postclosure analysis.)
Lower natural barriers	Important to waste isolation	The lower natural barriers delay radionuclide movement to the groundwater aquifer and receptor location by water residence time, matrix diffusion, and sorption. Prevent or reduce the release rate of radionuclides to the accessible environment.

Table B-2. Postclosure performance requirements.

Performance Areas	Performance Objectives
Individual protection—releases from the undisturbed Yucca Mountain disposal system	Less than 0.15 mSv (15 mrem). With a reasonable expectation that for 10,000 years following disposal, the reasonably maximally exposed individual receives less than the above annual dose.
Individual protection—releases from the human intrusion into the Yucca Mountain disposal system	Less than 0.15 mSv (15 mrem). With a reasonable expectation that, at or before 10,000 years, the reasonably maximally exposed individual receives less than the above annual dose as a result of human intrusion.
Groundwater protection—limits on radionuclides in consuming the representative volume of water	Combined radium-226 and radium-228 <5 picocuries per liter (includes natural background). Gross alpha activity (including radium-226 but excluding radon and uranium) <15 picocuries per liter (includes natural background). Combined beta and photon emitting radionuclides <0.04 mSv (4 mrem) per year to the whole body or any organ, based on drinking 2 liters of water per day from the representative volume (does not include natural background). A reasonable expectation that, for 10,000 years of undisturbed performance after disposal, releases of radionuclides from waste in the Yucca Mountain disposal system into the accessible environment will not cause the level of radioactivity in the representative volume of groundwater to exceed the limits above.

3. Will the repository after the inclusion of FEPs affected by DOE SNF in the TSPA model still comply with postclosure public health and environmental standards?

All the postclosure DOE SNF activities were performed by the NSNFP because the deliverables from the activity will help answer the above three questions. These activities were performed over the last several years and under the NSNFP or Office of Civilian Radioactive Waste Management (RW) personnel, which is consistent with the regulatory requirements:

- Determine what are the impacts to the repository's FEPs when DOE SNF are included in the repository
- Screen FEPs according to RW technical criteria and Nuclear Regulatory Commission (NRC) regulations
- Retain and evaluate only those FEPs with probabilities $>10^{-8}$ per year or significantly change radiological exposure or radionuclide releases; include the retained FEPs into nominal TSPA scenario or disruptive scenario model representation
- Provide radionuclide inventory for evaluation during the nominal and disruptive events scenarios; consider uncertainty of estimated inventory in the analysis

- Provide DOE SNF degradation behaviors for radionuclide release evaluation during the nominal and disruptive events scenarios; conduct sensitivity analyses of DOE SNF groups to provide insights of representing DOE SNF as a surrogate

- Conduct analyses on quantity and chemistry of water in contacting DOE SNF and high-level waste to provide insights to the impact of water quantity and material interactions within the codisposal waste package

- Provide information on other radionuclide release mechanisms applicable to DOE SNF

- Evaluate disruptive events and compared releases from DOE SNF

- Demonstration of compliance with the postclosure public health and environmental standards when repository includes the DOE SNF

  - With postclosure individual, human intrusion

  - With separate groundwater protection standards.

B-1. GROUPING OF DOE SNF FOR POSTCLOSURE PERFORMANCE ASSESSMENTS

Showing compliance with the postclosure standards for the DOE SNF is complex because of the varieties of DOE SNF in the DOE Office of Environmental Management (EM) inventory. The report *Source Term Estimates for DOE Spent Nuclear Fuels*[1] shows that there are approximately 2,400 MTHM DOE SNF (excluding naval fuel) identified for direct disposition in the proposed repository at Yucca Mountain. This large number of DOE SNF poses great challenges in showing compliance with the regulatory standards. Taking on this challenge, the NSNFP held a meeting in 1998 with participation from the NSNFP, DOE sites, naval program, RW, and other national laboratories to consider ways to represent the DOE SNF in the repository.

By using the information from the Spent Fuel Database at the time, the team recommended that the DOE SNF be grouped to support specific purposes—specifically for preclosure safety analysis ("design basis events" was the term used during early repository consideration), postclosure TSPA analysis, and criticality. The results of the meeting with the justifications were published in the NSNFP report *DOE SNF Grouping in Support of Criticality, DBE and TSPA-LA*.[2]

For the purpose of postclosure TSPA analysis, the report suggested that DOE SNF could be represented by 10 groups (excluding the naval SNF). The grouping was primarily based on understanding of the fuel's long-term behavior, and a typical fuel for the group is selected based on the quantity of fuel that is in the DOE SNF inventory. The DOE SNF TSPA groups and the typical fuels for each group are indicated in Table B-3. Results from initial TSPA calculations and the NSNFP release rate program showed that a single surrogate was appropriate to represent DOE SNF.

The DOE SNF will be represented in the TSPA-license application model as a single surrogate fuel. The surrogate model uses an instantaneous release where all radionuclides in the fuels will be available for release when the waste package is breached. No credit is taken for DOE SNF cladding or for DOE standard canisters. The confidence of such representation is based on analyses of the individual 10 fuel groups.

Table B-3. DOE SNF groups used in the TSPA-license application.

Fuel Group/Fuel Matrix	Typical Fuel in the Group	Comment
Naval fuel	Naval Fuel [151][a]	Info by NNPP
Pu/U alloy	FERMI Core 1 and 2 (Standard fuel subassembly) [456]	
Pu/U carbide	FFTF-TFA-FC-1 [325]	
Mixed oxide fuel	FFTF-DFA/TDFA [71]	
U/Th-carbide	Fort St. Vrain Reactor [86]	
U/Th oxide	Shippingport LWBR Reflect. IV [371]	
Uranium metal	N-reactor fuel [991]	
Uranium oxide	TMI-2 core debris [229]	
Aluminum-based fuel (UAlx, U_3Si_2, uranium oxide in aluminum)	FRR pin cluster (Canada) [660]	
Miscellaneous	Miscellaneous RSWF fuel [366]	
U-ZrHx	TRIGA STD [235]	

a. The number in [ ] is the fuel identification (SNF ID#) used in the DOE/SNF/REP-078 (see Reference 1).

DFA driver fuel assembly; FRR foreign research reactor; RSWF Radioactive Scrap Waste Facility; FERMI Enrico Fermi Reactor; LWBR light water breeder reactor; TDFA test driver fuel assembly; FFTF Fast Flux Test Facility; NNPP Naval Nuclear Propulsion Program; TMI-2 Three Mile Island Unit 2

B-2. FEATURES, EVENTS, AND PROCESSES CONSIDERATIONS

Through FEPs screening and analyses, RW has identified and retained for postclosure a total of nine barriers that either: (1) prevent or substantially reduce the rate of movement of water or radionuclides from the Yucca Mountain repository to the accessible environment or (2) prevent the release or substantially reduce the release rate of radionuclides from the waste. These nine barriers consist of four natural barriers and five engineered barriers. Figure B-1 is a pictorial depiction of the nine barriers discussed in RW's *Yucca Mountain Preliminary Site Suitability Evaluation*.[3] As part of the safety classification of SSCs and barriers to support the repository design, the importance of these and other systems are further refined. The system, subsystem, and barrier important to waste isolation are indicated in Table B-4.

When DOE SNF was first considered for direct disposition, the NSNFP coordinated with the appropriate RW management and operations contractor personnel to determine the impact of the addition of DOE SNF into the proposed repository.[4] As part of the DOE SNF FEPs screening, two analysis model documents were completed and published by the NSNFP that evaluated the impact of the addition of DOE SNF into the repository. The documents are:

- NSNF/EP-3.05/001—*Feature, Event, and Process Identification to Support Disposal of Department of Energy Spent Nuclear Fuel at the Yucca Mountain Repository*, Revision 0

- NSNF/EP-3.05/004—*Total System Performance Assessment Disposition of Misc. Features, Events, and Processes*, Revision 0.

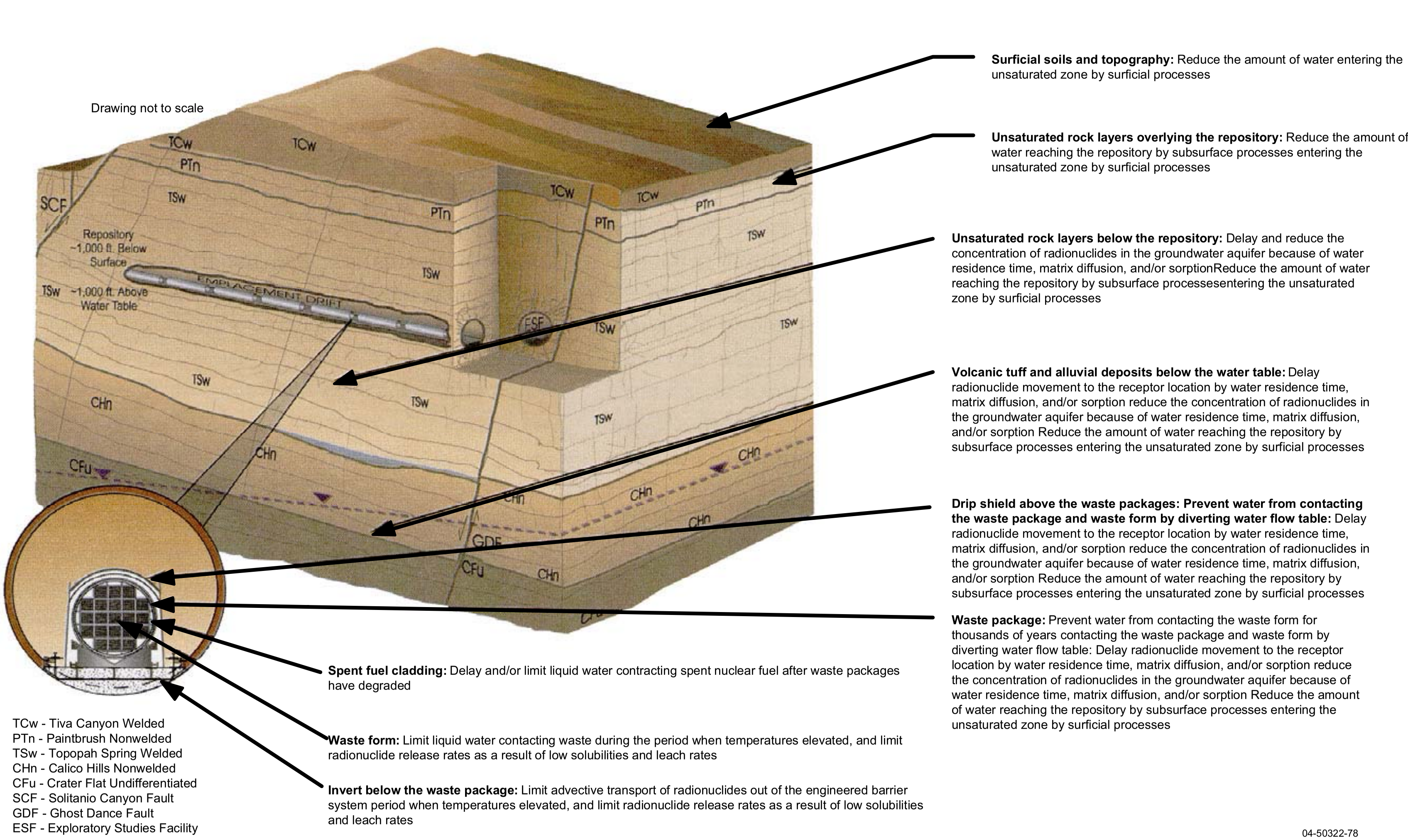

Figure B-1. Conceptual illustration of natural and engineered barriers and their functions contributing to the isolation of waste (see Reference 3).

Table B-4. Yucca Mountain repository engineered and natural barriers classified as important to waste isolation.

System, Subsystem, or Barrier[a]	Important to Waste Isolation[a]	Basis for Classification[a]	Comment
Upper Natural Barriers System			
Topography and surficial soils unsaturated zone to the repository horizon	Yes	Prevents or substantially reduces the movement of water.	
Lower Natural Barriers System			
Unsaturated zone below the repository horizon and saturated zone below and downgradient from the repository	Yes	Delays radionuclide movement to the groundwater aquifer because of water residence time, matrix diffusion, and/or sorption. Prevents or reduces the release rate of radionuclides to the accessible environment	
Engineered Barriers System			
Drip shield	Yes	Prevents or reduces the waterflow that could contact the waste package and waste form by diverting waterflow around the waste package; prevents rockfall damage to the waste package that could affect waste package performance.	
Waste package (Outer corrosion barrier)	Yes	Prevents or reduces the water contact with the waste form and confines the waste. Provides dry inert environment until waste package breach to delay onset of waste form degradation.	
Waste package (internals)	Yes	Provides internal materials whose corrosion products sorb radionuclides from the degraded waste form in order to reduce radionuclide release from the breached waste package. Prevents in-package criticality for postclosure period.	
Waste form	Yes	Limits radionuclide release rates because of low solubilities or low diffusion through degraded engineered barrier.	DOE SNFs do not take waste form credit
Spent fuel cladding	Yes	Prevents the contact of water with the waste form for most SNF.	DOE SNFs do not take cladding credit

Table B-4. (continued).

System, Subsystem, or Barrier[a]	Important to Waste Isolation[a]	Basis for Classification[a]	Comment
Drift invert	Yes	Limits radionuclide release rates through the granular invert material by limiting accumulation of water in the invert regime.	
DOE SNF Disposable Canister			
DOE SNF (Internal)	Yes	DOE SNF canister internal basket limits the quantity of fissile materials could be loaded into the canisters. Neutron absorber, as needed, in the basket prevents in-package criticality for postclosure period.	Needs to be included in the safety classification of SSCs and barriers documentation.
Emplacement Drift System			
Emplacement drift excavated opening	Yes	The emplacement drift opening provides thermal hydrological properties and size and layout properties consistent with TSPA modeling.	
Emplacement pallet	Yes	Provides structural support of the waste package and minimizes potential degradation mechanisms.	
a. See Reference 3.			

The DOE SNF FEPs screening has identified two potential features, which are unique to the DOE SNF, that may impact the repository performance. They are (1) potential of a pyrophoric event—a self sustaining thermal excursion—in the multi-canister overpack loaded with uranium metal (N-reactor) fuels and (2) potential of a criticality beyond the 10,000 years regulatory period from the highly enriched DOE SNF. The following evaluations (NSNF/EP-3.05/002, and NSNF/EP-3.05/003) have dismissed both the events as having a very low probability and consequence relative to the performance of the repository.

- NSNF/EP-3.05/002—*Screening Argument for Pyrophoricity*, Revision 0
- NSNF/EP-3.05/003—*Criticality Risk for Beyond 10,000 Years*, Revision 0.[a]

A detailed analysis of a chemical reaction in a multi-canister overpack was completed as part of the TSPA activity in FY 2001.[5] The analysis has concluded that even if the two adjacent commercial SNF waste packages besides the DOE SNF codisposal package were affected, such a chemical reaction would not increase the release of radionuclides as compared to the nominal scenario. For the potential of a

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a. This evaluation was completed by the NSNFP. However, RW has completed a calculation called *Configuration Generator Model*[6] in which RW reached the conclusion that, based on the current design, criticality in the regulatory period for all SNF (i.e., including DOE SNF) is less than 10^{-4} over 10,000 years. Thus criticality could be screened out of the TSPA-license application model. See Appendix C for further discussion.

criticality, analyses show that fissile loading in conjunction with a neutron absorber, waste package performance, and the amount of water enters the waste package during the regulatory period as such that the probability of a criticality situation will be very low (below the 10^{-8} per year threshold).[6] For disruptive scenarios, the criticality potential and consequences are also very low. More detailed discussions of DOE SNF criticality are covered in Appendix C. Figure B-2 summarizes the DOE SNF FEPs screening activities that were included in the license application.

B-3. DOE SNF MODEL ABSTRACTION

Table B-4 indicated that the waste form is a barrier important to waste isolation, but for the DOE SNF, no credit is taken for the waste form or the cladding. In support of the DOE SNF in the license application, RW has conducted a detailed degraded behavior analysis for each DOE SNF group proposed in the grouping report. The analysis recommended how DOE SNF should be represented in light of the TSPA-license application model.[7] The analysis model report recommended that “Upper-limit models should be used (for all the DSNFs [DOE SNFs]) in cases where their usage in TSPA analyses results in acceptable boundary dose, and when other less conservative models are not needed.” The upper-limit model is an instantaneous release model applied to the DOE SNF in the TSPA-license application safety case. This recommendation was based on the fact that the conservative and best estimate models are not currently fully validated.

However, the analysis model report allows the analyst to choose the conservative uranium metal model for the license application safety case, because the conservative uranium metal model would be bounding for normal TSPA time steps.

Thus the DOE SNFs were represented in the TSPA-license application with an instantaneous release model. The confidence of such representation was built on a number of NSNFP activities that provided a good understanding of the DOE SNF characteristics and behaviors under the repository conditions. The results of these activities indicated that DOE SNF could be bounded using an instantaneous release model. The activities and relationships to the TSPA-license application are summarized in Figure B-3.

Several concerns relative to DOE SNF fuel behaviors in the repository included:

- How the fuel (matrix) will degrade over long periods in the repository environment
- How the fuel cladding will protect or not protect the fuel meats over long periods in the repository environment
- As the fuel degrades, will the corrosion products contain a large number of colloids such that they may aid the transport of significantly more radionuclides of concern out to the accessible environments and thus increase the dose to the public?

Using the above areas of concern and the DOE SNF groups identified in support of postclosure analysis, three activities were performed to provide the confidences that EM understands the behaviors of DOE SNF. The first was a literature search that may help in understanding how the various DOE SNF types may behave under conditions like the repository. As expected, fuel types that have been manufactured and used in significant quantity over the past 20 to 30 years are the fuel types that have the most information. However, because the interests are in the operational performance of the fuel, some conditions are not exactly like what is expected in the repository. However, a lot of good and useful

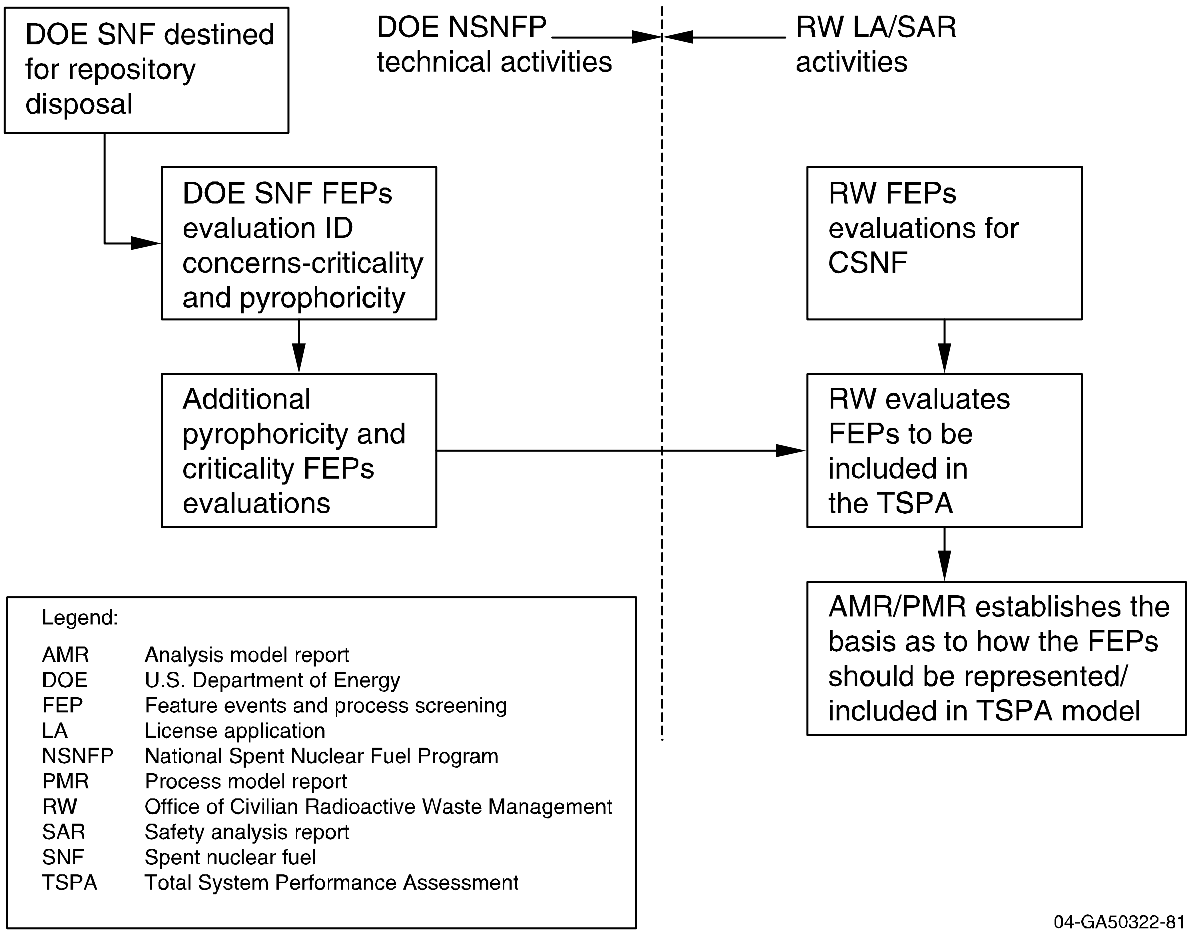

Figure B-2. DOE SNF FEPs screening activities that were considered in the license application.

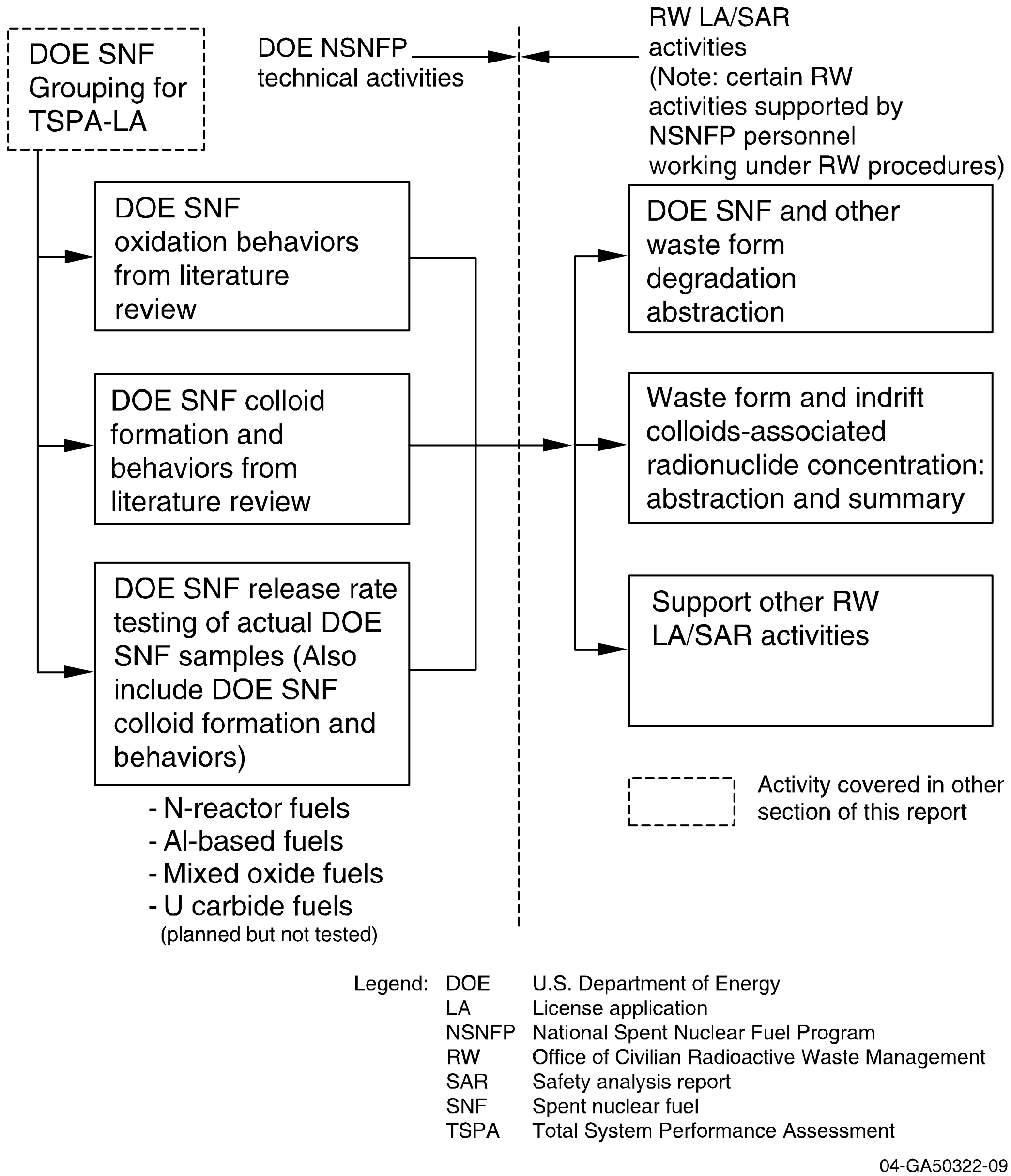

Figure B-3. DOE SNF model abstraction activities as related to repository license application.

information was collected and published in two separate reports covering the DOE metallic and nonmetallic fuels that are in the EM inventory. These reports are:

- Department of Energy Environmental Management *Review of Oxidation Rates of DOE Spent Nuclear Fuel Part 1:Metallic Fuel*.[8]

- Department of Energy Environmental Management, *Review of Oxidation Rates of DOE Spent Nuclear Fuel Part II: Nonmetallic Fuel, Part 2*.[9]

Similarly, a literature search was performed on the potential of colloid formation as the various DOE SNFs degraded in the repository conditions. Because the United States is the only country in the world that has selected an unsaturated repository, some of the information collected is not directly applicable to a repository with an oxidizing environment. Nonetheless, the indication is that the quantity of colloids from DOE SNF degradation should be no more than commercial SNF. Thus, DOE SNF colloids should not significantly increase the transport of radionuclides under the repository conditions. The results of the literature search were published in a report titled, *Colloid Formation and the Potential Effects on Radionuclide Transport in a Geologic Repository for Spent Nuclear Fuel*.[10]

To provide further confidences that irradiated DOE SNF will behave similarly to the unirradiated material tests reported in the literature review above, release rate testing was planned on a selected number of DOE SNFs—specifically, the N-reactor fuels, Al-based fuels; mixed oxide fuels; and U carbide fuels. The fuels were selected based on the potential impact they may have on the repository performance. (See the *Selection of Fuel Types for Release Rate Testing*, DOE/SNF/REP-045.)[11] Results from the release rate testing program were compared to the results from the literature search and were published in the report titled, *Review of DOE Spent Nuclear Fuel Release Rate Test Results*.[12]

As summarized in Figure B-3, the information from all three of these reports has been considered and used in several repository documents. For the repository license application, the DOE SNF degradation behavior covered in References 8 and 9 have been considered and incorporated into the analysis model report *DSNF and Other Waste Form Degradation Abstraction* (see Reference 7). Similarly, the colloid behavior aspects of DOE SNF during degradation were covered in the analysis model report, *Waste Form and Indrift Colloids-Associated Radionuclide Concentration: Abstraction and Summary*.[13] The DOE SNF representation in the TSPA-license application model was based on the recommendations from these two analysis model reports.

B-4. DEMONSTRATION OF DOE SNF COMPLIANCE WITH THE POSTCLOSURE TSPA-LICENSE APPLICATION MODELING

As indicated in the analysis model report *Initial Radionuclide Inventories*, Table 2, "Waste Package Configurations," RW plans to disposition 12 waste package types, totaling ~11,200 waste packages in the upcoming December 2004 license application submittal. The approximate ratio of DOE SNF waste packages to the total number of waste packages is about 1 to 3. This means that for every waste package that may breach and fail in the repository, there is only one chance in three that it contains DOE SNF or high-level waste. Thus, demonstration of DOE SNF compliance cannot be considered for the DOE SNF by itself but rather must be integrated and considered with all the materials placed into the entire repository.

B-4.1 Historic DOE SNF Compliance Activities

Prior to the license application, DOE SNF had been considered and evaluated with the commercial SNF and high-level waste as part of the site recommendation and viability assessment process. To support the efforts, the NSNFP developed two reports titled *DOE Spent Nuclear Fuel Information in Support of TSPA-SR*[14] and *DOE Spent Nuclear Fuel Information in Support of TSPA-VA*[15] as well as developed a source term for use in these analyses (see Reference 1). In both evaluations, the regulatory compliance analyses have shown that the performance of the repository, including DOE SNF waste packages, is below the regulatory limits. In addition, depending on the type of commercial SNF (i.e., commercial SNF with stainless steel or zircaloy cladding), comparison of the nominal DOE SNF radionuclide releases from a codisposal waste package has been shown to be about one order of magnitude below the releases from a commercial SNF waste package.[16,17] Thus the contribution from the DOE SNF to repository performance may be easily bounded by the commercial SNF given the number of

DOE SNF waste packages and the fact that ~21% of the commercial SNFs waste package may contain stainless steel cladding.

B-4.2 DOE SNF TSPA-License Application Compliance Activities

RW's characterization program has acquired additional knowledge of the behavior of the infiltration, water transport, and absorption and desorption characteristics over the last several years. The findings have been represented in the TSPA-license application model. The NSNFP reports, *Source Term Estimates for DOE Spent Nuclear Fuels, Additional DOE Spent Nuclear Fuel Information in Support of TSPA-License Application Analysis* (see References 1 and 18), in conjunction with the DOE SNF model abstraction recommendations (Appendix B-3) form the bases of the DOE SNF representation in the compliance analysis.

The DOE SNF compliance works will not be entirely completed until the license application has been submitted. However, based on the results from the site recommendation and viability assessment, both the NSNFP and RW expect the DOE SNF performance to remain the same relative to the commercial SNF, because the updated repository behaviors are equally applicable to the commercial SNF as well as the DOE SNF. In addition, preliminary TSPA runs using the ongoing license application model appear to confirm that conclusion. Final license application analyses and license application sensitivity runs for DOE SNF have been planned and will be completed during the July through September 2004 timeframe. The final compliance TSPA-license application analyses will provide such confirmation.

The DOE SNF will be represented in the TSPA-license application model in the following manner:

- Single surrogate model—instantaneous release model where all radionuclides in the fuels will be available for release when the waste package is breached. No credit is taken for DOE SNF cladding. No credit is taken for standardized DOE SNF canisters (see Reference 7).

- DOE SNF radionuclide inventories are evenly distributed in the total number of DOE standard canisters estimated in REP-078 (see Reference 1). The radionuclide inventory and number of canister uncertainties are incorporated into the radionuclide inventories used in the TSPA-license application.[19]

As part of the *Total System Performance Assessment (TSPA) Model/Analysis for the License Application* document,[20] RW has identified a series of analyses that will provide the validation and confidence for the surrogate representation of the DOE SNF in the TSPA-license application. The series of nominal scenario analyses include comparison of dose history results from individual fuel groups using the realistic degradation models, fuel groups radionuclide inventory, and canister count to the dose history results from the surrogate DOE SNF model. Other selected sensitivity analyses include evaluation of impact from specific fuel group air alteration rates, uncertainties in fuel surface area, free radionuclide inventory, bounding radionuclide inventory, specific fuel group canister counts, and finally the plots of key radionuclides that contribute to total dose from specific DOE SNF. The results of the analyses will be part of the above license application document that will provide supporting validation and confidences that the DOE SNF has been properly represented in the TSPA-license application model.

Besides the nominal scenario, RW's FEP screening has identified disruptive scenarios that will have to be considered in the TSPA-license application model. Specifically, disruptive events such as igneous intrusion and eruption, and seismic scenarios will be part of the TSPA-license application model. A number of deterministic simulations covering these disruptive cases will be evaluated. Specifically, DOE SNF waste packages will be part of this comparison. During such scenarios, the commercial SNF with stainless steel cladding will essentially be like fuels without any cladding protection. Both the NSNFP and RW believe the commercial SNF waste packages will bound the DOE SNF waste packages. The results of the analyses will be part of the above license application document that will provide supporting evidence that DOE SNF has been properly represented in the TSPA-license application model under the disruptive event scenarios.

B-5. DEMONSTRATION OF DOE SNF RISK INSIGHTS BASED ON UNDERSTANDING OF DOE SNF

In the DOE SNF model abstraction discussion, the NSNFP presented the technical activities that were performed to provide the confidence that DOE has a reasonable understanding of how the DOE SNF will behave under repository conditions. Appendix B-3 covers the oxidation and degradation of DOE SNF materials as well as the potential of forming colloids that may increase the transport of radionuclides to the accessible environment. However, when fuels and high-level waste glass are placed into the same waste package (codisposal waste package concept), questions arise as to the potential interactions that may exist between the DOE SNF, the high-level waste materials, and the waste package and its basket structure. Specifically, how does the interaction in conjunction with the variability (chemistry and quantity) of water flowing through the waste package impact the transport and availability of radionuclides of interest (^{235}U, ^{239}Pu, ^{237}Np, ^{129}I, ^{99}Tc) at the accessible environment?

To understand the potential of such interactions, the NSNFP initiated an activity that uses the RW qualified geochemical modeling computer code called EQ3/6 to evaluate such interaction. However, EQ3/6 cannot track specific radionuclide isotopes, therefore the radionuclide species were represented by the element. The NSNFP has completed two reports[21,22] that cover six DOE SNF groups:

1. Mixed oxide fuel (Fast Flux Test Facility fuels) (see Reference 21)
2. Uranium/thorium oxide fuel (Shippingport light water breeder reactor fuel) (see Reference 21)
3. Plutonium/uranium alloy fuel (FERMI fuel) (see Reference 21)
4. Uranium metal fuel (N-reactor fuel) (see Reference 22)
5. Uranium/thorium carbide fuel (Fort Saint Vrain reactor fuel) (see Reference 22)
6. Al-based fuels (melt and dilute waste form[b]) (see Reference 22).

Uranium zirconium hydride fuel (TRIGA fuel), highly enriched uranium oxide fuel (Shippingport pressurized water reactor fuel), and low-enriched uranium oxide fuel (Three Mile Island reactor fuel) evaluations have been completed recently.[23]

A number of scenarios were considered as part of these evaluations. Under the worst condition using a high degradation rate for the high-level waste, low water flow rate, and suppression of certain mineral formation, all elements of interest were flushed out of the waste package after 100,000 years as compared to the base case where only part of the elements were flushed out of the waste package.

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b. At the time of evaluation, the Al-based fuel was to be treated using the melt and dilute process prior to disposal at the repository. EM has since decided to directly dispose of the Al-based fuel. Although the interaction evaluation has not shown any significant increases of doses due to the codisposal concept, additional evaluations may be required as part of the NRC request for additional inquiry.

Complete radionuclide release from a codisposal waste package is highly unlikely. Even if one assumes that it is possible to release all radionuclides of interest from a codisposal waste package, as compared to the nominal case (releases about 50% of the radionuclide of interest at 100,000 years after the waste package breaches), the potential dose increase would be less than twice the nominal case scenarios. This increase is insignificant in comparison with the other uncertainties regarding the performance of the repository as a whole.

In addition, a HIC was considered for the degraded fuels from long-term basin storage or small pieces of DOE SNF from postirradiation examination. The HIC would be fabricated out of a robust material like Alloy 22. The HIC would serve two purposes. First, it would be a container for handling of the degraded and small pieces of DOE SNF at the DOE sites when the fuel is being packaged for repository disposal. And secondly, it would serve as another barrier and provide additional delay of releases at the repository.

An analysis was performed to consider the HIC's performance and how the number of HICs in a canister may affect the repository performance. The calculation titled, "Performance Assessment of Disposal of Selected U.S. Department of Energy Spent Fuel in High Integrity Cans,"[24] presents the finding of the analysis. Compared to the waste package, the HIC provided an additional 40,000 to 60,000 years of delay in terms of radionuclide release protections.

B-6. SENSITIVITY ANALYSIS IN SUPPORT OF UNDERSTANDING DOE SNF PARAMETERS

Appendix B-3 discussed the DOE SNF activities supporting the DOE SNF model abstraction. A number of sensitivity analyses were also conducted as part of the TSPA-viability assessment and TSPA-site recommendation to understand DOE SNF parameters and how they may have an effect on the repository performance. The sensitivity analyses focused on the DOE SNF parameters that will be needed as part of the compliance analysis.

For the TSPA-viability assessment, the *Waste Form and Indrift Colloids-Associated Radionuclide Concentrations* analysis model report (see Reference 14) presents and describes the abstraction of the colloids process model for the waste form and engineered barriers. Section 6.3.1.2 of the analysis model report discusses the colloids from the corrosion of commercial and DOE SNF. Based on the various reports, references, and test results, the analysis model report concluded that "There are no direct colloid source term contributions from commercial and DOE SNF wastes." Thus, DOE SNF contributions are not required to be included in the TSPA-license application colloid model abstraction.

B-7. OTHER DOE SNF POSTCLOSURE ACTIVITIES

If requested by the NRC, additional postclosure analyses will be conducted. At this time, activities are being planned as part of future work scopes to meet the request for additional information.

B-8. REFERENCES

1. U.S. Department of Energy, Environmental Management, *Source Term Estimates for DOE Spent Nuclear Fuels*, DOE/SNF/REP-078, Revision 1, January 2004.

2. U.S. Department of Energy, Environmental Management, *DOE SNF Grouping in Support of Criticality, DBE and TSPA-LA*, DOE/SNF/REP-046, Revision 0, May 2000.

3. U.S. Department of Energy, Office of Civilian Radioactive Waste Management, *Yucca Mountain Preliminary Site Suitability Evaluation*, DOE/RW-0540, Las Vegas, Nevada, July 2001.

4. Letter PDW-03-98 from P. D. Wheatley (Lockheed) to Distribution, "Workshop to Assemble a Comprehensive List of FEPs Applicable to the DOE SNF," November 18, 1998.

5. U.S. Department of Energy, Office of Civilian Radioactive Waste Management, *CRWMS M&O Performance Assessment of a Potential Post-Closure Pyrophoric Event Involving Uranium Metal Spent Fuel*, CAL-WIS-PA-000009, Revision 0, November 2001.

6. Bechtel SAIC Company, *Configuration Generator Model,* CAL-DS0-NU-000002, Revision 00A, Las Vegas, Nevada: Bechtel SAIC Company, 2004.

7. U.S. Department of Energy, Office of Civilian Radioactive Waste Management, *DSNF and Other Waste Form Degradation Abstraction*, ANL-WIS-MD-000004 Revision 2, ICN-01, July 9, 2003.

8. U.S. Department of Energy, Environmental Management, *Review of Oxidation Rates of DOE Spent Nuclear Fuel Part 1:Metallic Fuel*, DOE/SNF/REP-0054, September 2000.

9. U.S. Department of Energy, Environmental Management, *Review of Oxidation Rates of DOE Spent Nuclear Fuel Part II: Nonmetallic Fuel,* DOE/SNF/REP-068, July 2003.

10. U.S. Department of Energy, Environmental Management, *Colloid Formation and the Potential Effects on Radionuclide Transport in a Geologic Repository for Spent Nuclear Fuel,* DOE/SNF/REP-070, January 2002.

11. U.S. Department of Energy, Environmental Management, *Selection of Fuel Types for Release Rate Testing,* DOE/SNF/REP-045, March 1999.

12 U.S. Department of Energy, Environmental Management, *Review of DOE Spent Nuclear Fuel Release Rate Test Results*, DOE/SNF/REP-073.

13. U.S. Department of Energy, Office of Civilian Radioactive Waste Management, *Waste Form and Indrift Colloids-Associated Radionuclide Concentration: Abstraction and Summary,* MDL-EBS-PA-000004, Revision 00, ICN 01, December 2003.

14. U.S. Department of Energy, Environmental Management, *DOE Spent Nuclear Fuel Information in Support of TSPA-SR*, DOE/SNF/REP-047, Revision 2, February 2002.

15. U.S. Department of Energy, Environmental Management, *DOE Spent Nuclear Fuel Information in Support of TSPA-VA,* DOE/SNF/REP-027, Revision 0, September 1998.

16. "Total System Performance Assessment Sensitivity Studies of U.S. Department of Energy Spent Nuclear Fuel," A00000000-01717-5705-00017, Revision 01, September 30, 1997.

17. U.S. Department of Energy, Office of Civilian Radioactive Waste Management, *Performance Assessment of U.S. Department of Energy Spent Fuels in Support of Site Recommendation*, Revision 00, MOL.20010627.0026, May 2001.

18. U.S. Department of Energy, Environmental Management, *Additional DOE Spent Nuclear Fuel Information in Support of TSPA-License Application Analysis,* DOE/SNF/REP-081, Revision 0, January 2004.

19. U.S. Department of Energy, Office of Civilian Radioactive Waste Management, "Initial Radionuclide Inventories," ANL-WIS-MD-000020, Revision 00, November 2003.

20. *Total System Performance Assessment (TSPA) Model/Analysis for the License Application, Volume I*, MDL-WIS-PA-000004, Revision 00C, Bechtel SAIC, Company, July 2004.

21. U.S. Department of Energy, "Geochemical Interactions in a Failed Co-Disposal Package for Selected U.S. Department of Energy Spent Fuels," Revision 0, OCRWM, 2001.

22. U.S. Department of Energy, Office of Civilian Radioactive Waste Management, *Geochemical Interactions in Failed Co-Disposal Waste Packages for N-reactor and Ft. St. Vrain Spent Fuel and the Melt and Dilute Waste Form*, ANL-EBS-PA-000007, Revision 00, July 2002.

23. U.S. Department of Energy, Office of Civilian Radioactive Waste Management, *Geochemical Interactions in Failed Co-Disposal Waste Packages for Shippingport PWR and TRIGA Spent Fuel and in Dual Canister Packages of Three Mile Island and Fort St. Vrain Spent Fuel*, Revision 0, September 2004.

24. U.S. Department of Energy, "Performance Assessment of Disposal of Selected U.S. Department of Energy Spent Nuclear Fuel in High Integrity Cans," CAL-WIS-PA-0000003, Revision 00, December 2000.

Appendix C

Criticality Activities Supporting Pre- and Postclosure

Appendix C

Criticality Activities Supporting Pre- and Postclosure

Development of a strategy to support promotion of criticality safety relies on the guidance provided in *Disposal Criticality Analysis Methodology Topical Report*.[1] This report presents an overall methodology approach starting in Section 3.1. Section 3.2 discusses the imposition of performance criteria to ensure appropriate criticality controls are implemented in the waste package design. Degradation scenarios built from the features, events, and processes (FEP) screening are described in Section 3.3. Such scenarios include not only those potential criticality configurations inside the waste package, but also the near and far field environments.

Section 3.4 of the Methodology Topical Report describes the individual parameters and how they should be treated using the environmental properties of the repository. The neutronic methodology approach for evaluating the criticality potential (calculated k_{eff}) for the various configurations is described in Section 3.5. The remaining sections (3.6 through 3.8) estimate the probability of occurrence of critical configuration classes, their subsequent consequences, and associated risk, respectively.

U.S. Department of Energy (DOE) spent nuclear fuel (SNF) criticality safety in the repository relies on this same criticality analysis methodology used for commercial fuels. The DOE SNF preclosure and postclosure activities are discussed in the following sections.

C-1. DOE SNF PRECLOSURE CRITICALITY

The strategy for preclosure criticality control is to show that DOE SNF when placed in sealed canisters cannot exceed the critical limit even with the addition of moderator. Single canister analyses in each of the nine fuel groups have shown that this can be achieved as long as the fuel/basket combinations remain intact. Table C-1 identifies the systems, structures, and components (SSCs) to implement the preclosure criticality control strategy.

This strategy is implemented and enforced through facility controls on the introduction of moderator to any packaging facilities, and reliance on the integrity of the sealed DOE standard canister under any postulated accident condition. First and foremost, there is an expectation that any fuel loaded into a standard canister will be controlled under stringent drying conditions. Second, analysis of the final surface facility design will be performed to show that criticality is unlikely regardless of the fuel type as long as moderator is excluded from the canister.

Basket designs and proposed fissile loads per canister are specific to one of nine baseline fuel types. Use of these baskets for other fuels will require a demonstration of how these “other” fuels are bounded by the baseline fuel characteristics. The baseline fuels were selected based on an expectation that they would provide (for fissile species, mass, enrichment, and the ability to fill at least one canister) the highest fissile loading per canister and most reactive parameters expected for all conceivable conditions subsequent to preclosure. The basis for fissile loading the other fuels in standard canisters will have to demonstrate a calculated k_{eff} that is less than baseline fuel calculated k_{eff} for each separate canister loading.[2] For any condition associated with loading operations with the fuel, movement of loaded canisters, and credit for canister integrity (nonbreach) under design basis accident conditions, introduction of moderator into a DOE standard canister at any time during preclosure has been eliminated through the FEP screening process.[3] A significant part of such an analysis relies on maintaining geometry for both baskets and the loaded fuels in each individual canister.

Table C-1. Systems, structures, and components and barriers for preclosure criticality control.

SSC/Barrier	Category	Description
Facility controls on moderator	Important to safety	The Canister Transfer System facility will be designed and operated with control on moderator.
DOE standard canister storage racks	Important to safety	The DOE standard canister storage racks will control the geometry of DOE standard canister arrays. For most DOE standard canisters, the geometry controls will be sufficient to ensure there is no unsafe interaction. In the event that geometry is not sufficient, neutron absorbers will be added to the racks.
DOE standard canister baskets	Additional measure	The DOE standard canister basket will provide geometry control for the DOE SNF even if the DOE standard canister is breached. This geometry control will be sufficient to prevent a criticality for DOE SNF even under flooded conditions. Some baskets contain poisons for post closure criticality control. These provide additional preclosure margin.
DOE standard canisters	Additional measure	Because a DOE standard canister breach is a Beyond Category 2 event, the DOE standard canister will prevent introduction of moderator into the canister even if moderator is allowed into the facility.

Accident scenarios occurring during loading and unloading of the DOE standard canisters are considered for self-moderated fuels. One specialized analysis examined the combination of a dropped canister containing self-moderated fuel (TRIGA/UZrHx).[4] As a result of the accident, fuel may be pulverized and redistributed inside the DOE standard canister. Based on a premise for the retention and interspersed nature of the poisoned basket structure, no degree of fuel degradation as a result of the drop would exceed the critical limit for the fuel load in a standard canister. The canister can be oriented in any position ranging from vertical to horizontal. Based on the results of the physical argument and probability analysis presented in Sections 5 and 6 of additional analysis,[5] performed by the Office of Civilian Radioactive Waste Management management and operations contractor, all key parameters contributing to criticality can be screened out. In particular, criticality is shown to be possible only when neutron poison materials are separated from the fuel material. Physical considerations concluded that such separation cannot occur realistically. The probability analysis results show that the overall probability for reaching criticality (considering all key parameters contributing to criticality) is less than the screening criteria of 1.0E-04 over the entire preclosure operational period (assumed to be 100 years). Therefore, it was concluded that there is no criticality concern for dropping a self-moderated DOE standard canister, such as TRIGA, during the handling operation of the DOE standard canister in the surface facilities.

From a probabilistic basis, there would only be seven canisters out of an estimated 3,000 DOE standard canisters with both the enrichment and fissile mass that are of concern for this scenario. Subsequent analyses are tasked to identify the most reactive package condition at least for a single, loaded DOE standard canister. This effort necessarily includes analyses following introduction of moderation into an otherwise intact DOE standard canister load. Such a moderated condition is usually associated with eventual package breach expected to occur beyond postclosure. A flooded canister is screened out in the preclosure FEP analysis that identifies nonbreach conditions for any waste package (see Reference 3).

Eventually there will be the specialized issue of multiple, loaded DOE standard canisters in arrays, both in surface storage facilities and within a transport cask. While both cases encompass preclosure conditions, such analyses are outside the scope of this report. The conditions and scenarios for DOE standard canister transport to and handling at the repository have yet to be identified and incorporated in appropriate designs.

The SSCs and barriers important to safety will demonstrate that a criticality event is a Beyond Category 2. Beyond Category 2 analysis will include additional measures (prevention and mitigation) and are expected to demonstrate that the consequences of a criticality are well below the performance objectives.

C-1.1 DOE SNF Postclosure Criticality

The performance objectives of 10 CFR 63.113(b) specifically relate to the protection of the offsite public during the 10,000-year regulatory period and do not contain any special provisions for criticality control. While there are no specific requirements for criticality prevention and the consequences are expected to be well below the performance objectives,[6,7] DOE is committed to ensuring that a criticality is a low probability event during the postclosure regulatory period and beyond.

Based on the low probability of early waste package failures, drip shield failure, moderator availability, and of significant moderator entering the waste package, criticality has been screened out. The *Supplemental Science and Performance Analyses* states, “Criticality during the regulatory period was screened out when early failures were screened out. Even in the unlikely event of early waste package failures the conditions required for criticality are not likely. The failure mode postulated for early failures (e.g., cracks in the closure weld) is not sufficient for criticality to occur. … Furthermore, nuclear criticality was considered but determined not to have a significant impact on repository performance.”[8]“Criticality evaluations for various waste forms will be conducted prior to license application to confirm that the repository system will meet the criticality probability criterion of less than 1×10^{-4} per year for the entire repository for the regulatory period.” (See Reference 8.) Table C-2 identifies the SSCs and barriers on which the postclosure criticality control strategy is based.

Beyond the regulatory period, analyses will include additional measures (i.e., additional preventive and mitigative measures) and will be performed in accordance with the Criticality Methodology Topical Report.[9] Detailed discussions of the represented DOE SNF criticality analyses are presented in follow on sections of this overall document. These analyses were simplified through the use of a representative fuel for each SNF group.[10–18] The analysts confirm the suitability of the information for the representative fuels. These analyses demonstrate that a criticality is highly improbable when additional measures, such as the neutron absorbers, are considered.

Because DOE SNF information is not used to demonstrate that criticality is below the probability criterion, there will be no additional characterization of DOE SNF. In addition, the analyses for criticalities beyond the regulatory period (i.e., the analyses performed per the Criticality Analysis Methodology Topical Report) do not result in additional characterization requirements.

Postclosure conditions for loaded DOE standard canisters were based on a horizontal orientation within the sealed waste package. Beyond the orientation, the nonbreached condition of the waste package for anything other than a disruptive event provided assurance of moderator exclusion from both the waste package and the DOE standard canisters. From Section 3.2, “The DOE SNF postclosure licensing strategy relies on the combinations of engineered and natural barriers to meet the performance objectives of 10 CFR 63.113 (b).”

Table C-2. Systems, structures, and components and barriers for postclosure criticality control.

SSC/B	Category	Description
Waste package	Important to waste isolation	Waste package failure is a low probability event .
Drip shield	Important to waste isolation	Drip shield failure is a low probability event.
Fixed and/or loose absorbers in the DOE standard canister (as required)	Important to waste isolation	The neutron absorbers ensure that a critical configuration is not formed even after waste package and drip shield failure, and SNF degradation.
Engineered barriers: waste form, cladding, emplacement plug	Important to waste isolation	Even after waste package and drip shield failure, the engineered barrier system will still retard radionuclide transport. (Note: For DOE SNF, waste form and cladding credit are not currently taken in the postclosure analysis.)
Natural barriers	Important to waste isolation	The natural barriers above the waste package will reduce the likelihood that sufficient water is available to initiate a criticality. In addition, even after waste package and drip shield failure, the natural barriers system will still retard radionuclide transport.

The Repository Safety Strategy determined that "waste packages alone are predicted to prevent any release of radionuclides for more than 10,000 years." The *Supplemental Science and Performance Analyses* examined waste package life expectancy and concluded: "The upper-bound profile … indicates that not considering early waste package failures, the earliest first breach time for a waste package is approximately 120,000 years, much later than the 15,000 years for the previous baseline model and the 10,000 years of the TSPA-SR base case." (See Reference 8.)

In light of the low probability of moderator introduction to waste packages, any criticality analyses that evaluate moderated criticality do so with the goal of identifying mitigative conditions to ensure the calculated k_{eff} will remain below the allowable critical limit for that package. Water intrusion in a postclosure waste package environment introduces moderation that can promote increased reactivity in any waste package containing sufficient quantity of fissile material to go critical in engineered reactor systems. Along with this water intrusion, there is an inevitable degradation process that can contribute to failure of the basket and fuels, and movement of fissile material within the canister.

A significant portion of any such analysis relies on the ability to take credit for continued horizontal orientation of the waste package. Such orientation is a necessary condition that precludes axial reconcentration of fissile material. While radial redistribution of fissile material might occur, that aspect can be addressed with the combination of basket durability against degradation and when necessary, the incorporation of neutron absorbers.

Adoption of a given basket design enforces a limitation of the total fissile loading in a given canister. When that approach alone does not provide a sufficient margin to ensure criticality safety is maintained below the critical limit under degraded conditions, the addition of neutron absorbers, or poisons, to the DOE standard canister becomes necessary. Some baseline fuel analyses revealed no need for poisoning regardless of conditions inside the waste package or DOE standard canister. Other analyses used a combination of fixed poisons in alloyed basket material and, where necessary, the proposed addition of poisons in bead material. The neutron absorber of choice has evolved to gadolinium. Retention of the gadolinium in the bead material relies on the relative insolubility of the gadolinium upon degradation of the bead material. For the alloy material that incorporates the gadolinium in the metal matrix used in basket construction, the relative durability of the C-4 alloy against corrosion promotes gadolinium retention. All degradation analyses allow for some loss of gadolinium at predicted solubilities, yet retention and distribution of the remaining gadolinium in the degraded waste package allows the calculated k_{eff} to remain below the critical limit. The critical limits assigned to each fuel type are shown in Table C-3.

Nonpoisoned canisters are generally associated with low fissile loads. N-reactor fuels were analyzed critically safe because of the extremely low enrichments (<1.15 % 'smeared' ^{235}U). While the Shippingport Pressurized Water Reactor fuels have a relatively high fissile loading, the fuel assembly construction proved impervious to degradation when subjected to EQ3/6 analysis. The Fort St. Vrain fuel exhibited volume limitations that preclude formation of fissile atom-densities needed to support a criticality when stacked in a DOE standard canister. The Three Mile Island Unit 2 debris relied on a relatively low enrichment (<3.0%) to remain below the critical limit within the bounds of a DOE standard canister.

The use of poisoning for the remaining fuel types was determined ultimately by the most reactive, degraded condition predicted inside the DOE standard canister. Degradation scenarios include introduction of moderator into a breached canister as a necessary initial condition. Subsequent considerations include analysis based on mobilization and differential separation of neutron absorbers from the fissile material inside the DOE standard canister. All these analyses dealt with the combination of various degrees of degradation within both the DOE standard canister and the waste package and its associated degradation products.

Table C-3. Proposed fissile loads and poison requirements.

Fuel Matrix	Fuel Type	Fissile Mass (kg)	Critical Limit ($k_{eff} + 2\sigma$)	Poison Mass (kg)
UAl_x	ATR [15]	21.7	0.93	7.21
U metal	N-reactor [991]	54.5	0.93	None
MOX	FFTF [71]	42.3	0.92	9.29
UZr-UMo	Fermi [456]	115.3	0.93	9.04
$UZrH_x$	TRIGA-FLIP [239]	15.2	0.93	8.90
HEU oxide	Shippingport PWR [196]	19.5	0.93	None
U/Th oxide	Shippingport LWBR [380]	16.6	0.92	5.03
U/Th carbide	Fort St. Vrain [86]	7.4	0.93	None
LEU	TMI-2 core debris [229]	13.7	0.97	None

ATR = Advanced Test Reactor
Fermi = Enrico Fermi Reactor
FFTR = Fast Flux Test Facility
FLIP = Fuel Life Improvement Program
HEU = Highly Enriched Uranium
LEU = Low Enriched Uranium
LWBR = Light Water Breeder Reactor
PWR = Pressurized Water Reactor
TMI-2 = Three Mile Island Unit 2
TRIGA = Training, Research, and Isotope (General Atomic)

Transport of fissile material outside the DOE standard canister falls into the category of either near-field or far-field events. Such events require the mobilization of fissile material as a dilute concentration of fissile material and movement to a location that is capable of promoting a reconcentration or accumulation of fissile material in a more favorable geometry with a fissile atom-density and mass capable of producing a criticality. FEP screening arguments (see Reference 3) have eliminated these events on a probabilistic basis.

At this time, all criticality events have been screened out except for seismic events (see Reference 3). Such an event may yet be screened out for DOE standard canisters, depending on the details associated with the seismic event. The potential for a criticality based on seismic event sequences is still being analyzed. Their impact on ensuring criticality safety for DOE SNF might yet be precluded because the scenario ends up as a subset of one or more degradation scenarios already analyzed in degraded case analyses.

C-1.2 Demonstration of DOE SNF Criticality Risk Based on Understanding of DOE SNF

Enrichments of the DOE fuels can range from depleted uranium concentrations (<0.72% ^{235}U) to enriched values >93%. The higher enrichments are generally associated with smaller fuel pieces that can lead to higher fissile loadings in the DOE standard canisters.

Criticality safety for DOE fuels is promoted first by limiting the fissile loading in canisters. The fissile load limits analyzed within each proposed canister fissile load equates to both a linear loading for each canister and a resultant fissile atom-density. Because of this approach, fuel canister packaging with DOE fuels has resulted in fissile loads that remain below the critical limit for any intact fuel condition, whether flooded or not, and regardless of orientation.

The horizontal orientation expected for all postclosure DOE standard canisters promotes the inability to axially concentrate fissile material. This initial fissile loading in each canister may be able to redistribute axially outward, such as sand in a pile will distribute outward with time. Such outward redistribution results in an increased surface to volume ratio for the system; this situation generally increases the neutron leakage, thereby promoting a less reactive system.

Radial redistribution of fissile material within a canister, while not considered a condition with a high probability, cannot be dismissed. There are two conditions that might promote such radial movement of fissile material. Either condition to be considered in any criticality analysis is fuel specific. In the case of intact fuels, such as Fast Flux Test Facility, a breach of the SNF canister and introduction of clay material inside a basket compartment could promote expansion of the fuel pins outward as the fuel assembly degrades. The pins could move radially outward at the ends, while moderator might be concentrated in the fueled zone of the assembly for the most reactive case. In the case of the Three Mile Island canisters that contain rubble from the damaged core, spacing control of the fuel pellets was improbable. Multiple movements of the loaded canister between its vertical orientation during loading and horizontal orientation during transport complicate any criticality analyses. In this case, the goal is to identify the most reactive configuration based on balancing random pellet placement and void fraction within the SNF canister. The ability to “float pellets in space” occurs because of the unknown nature of the “other” debris in the canister that might create a support lattice, but for which no credit is taken for moderator exclusion.

Neutron absorbers are only required in some canister or basket combinations to remain below the critical limit, where degraded conditions can contribute to a more reactive arrangement of fissile material within the confines of the DOE standard canister. Gadolinium installed in a DOE standard canister provides very little neutron interaction without the water to thermalize the neutrons. However, the very presence of water leading to degradation of the canister contents dictates the need for a neutron absorber.

Criticality analyses indicated five of the nine baseline fuels require some form of neutron absorber installation inside the DOE standard canister. For the nonpoisoned canisters, fuel characteristics inside the canister even for degraded conditions (lower enrichments, volume limited, durability of the fuel) do not result in a calculated k_{eff} that exceeds the critical limit for that fissile isotope.

In the final analysis, FEP screening analysis has shown that water intrusion into the waste package is a beyond design basis event. Steps taken to ensure criticality safety beyond the postclosure period (>10,000 years) are mitigative, such as installation of neutron absorbers. Proof that intact conditions exist within any standard canister for all times after the postclosure period ends is contingent on a probability analysis. Similarly, behavior of fissile materials after postclosure due to eventual water-induced breach or disruptive events (seismic or volcanism) should only be evaluated on a case-by-case basis for a generic waste package.

C-2. REFERENCES

1. U.S. Department of Energy Office of Civilian Radioactive Waste, *Disposal Criticality Analysis Methodology Topical Report,* YMP/TR-004Q, Revision 2, November 2003.

2. U.S. Department of Energy Environmental Management, *Qualifying 'Other' DOE Fuels for Criticality Safety in Repository Packaging*, REP/SNF/REP-090, Revision 0, June 2004.

3. Bechtel SAIC Company, *Screening Analysis of Criticality Features, Events, and Processes for License Application*," ANL-EBS-NU-000008, Revision 2, September 2004.

4. U.S. Department of Energy, "Criticality Safety Evaluation for a 'Dropped' SNF Canister Containing Self-Moderated Fuels," EDF-023, April 2003.

5. U.S. Department of Energy Office of Civilian Radioactive Waste Management, "Criticality Screening Analysis on Dropping Self-Moderated DOE TRIGA Spent Nuclear Fuel, Preclosure Safety Analysis," ANL-WHS-RL-000001 Revision 00A, December 2003.

6. U.S. Department of Energy Office of Civilian Radioactive Waste, "Consequence of Analyses of Transient Criticality Internal to a Codisposal Waste Package Containing Intact FFTF SNF Pins," ANL-EDC-MU-000002, OCRWM, April 2002.

7. U.S. Department of Energy Environmental Management, *Nuclear Dynamics Consequence Analysis (NDCA) for the Disposal of Spent Nuclear Fuel in an Underground Geologic Repository*, DOE/SNF/REP-033, Revision 0, October 1998.

8. Bechtel SAIC Company prepared for U.S. Department of Energy, *FY 01 Supplemental Science and Performance Analyses*, Volume I, Revision 0, MOL.20010801.0404, July 2001.

9. U.S. Department of Energy Office of Civilian Radioactive Waste, *Disposal Criticality Analysis Methodology Topical Report*, YMP-TR-004Q, Revision 01, September 2000.

10. U.S. Department of Energy Environmental Management, *FFTF (MOX) Fuel Characteristics for Disposal Criticality Analysis*, DOE/SNF/REP-032, Revision 1, June 2002.

11. U.S. Department of Energy Environmental Management, *Fermi (U-Mo) Fuel Characteristics for Disposal Criticality Analysis*, DOE/SNF/REP-035, Revision 0, February 1999.

12. U.S. Department of Energy Environmental Management, *Shippingport PWR (HEU Oxide) Fuel Characteristics for Disposal Criticality Analysis*, DOE/SNF/REP-040, Revision 0, April 1999.

13. U.S. Department of Energy Environmental Management, *TRIGA (UZrH) Fuel Characteristics for Disposal Criticality Analysis*, DOE/SNF/REP-048, Revision 0, June 1999.

14. U.S. Department of Energy Environmental Management, *Shippingport LWBR (Th/U Oxide) Fuel Characteristics for Disposal Criticality Analysis*, DOE/SNF/REP-051, Revision 0, September 1999.

15. U.S. Department of Energy Environmental Management, *N Reactor (U-Metal) Fuel Characteristics for Disposal Criticality Analysis*, DOE/SNF/REP-056, Revision 0, February 2000.

16. U.S. Department of Energy Environmental Management, *Fort Saint Vrain HTGR (Th/U Carbide) Fuel Characteristics for Disposal Criticality Analysis*, DOE/SNF/REP-060, Revision 0, January 2001.

17. U.S. Department of Energy Environmental Management, *TMI Fuel Characteristics for Disposal Criticality Analysis*, DOE/SNF/REP-084, September 2003.

18. U.S. Department of Energy Environmental Management, *FFTF (MOX) Fuel Characteristics for Disposal Criticality Analysis*, DOE/SNF/REP-032, Revision 0, May 2002.

Appendix D

DOE Spent Fuel Database, Source Term Report And Chemical Reactivity Analysis Activities

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DOE Spent Fuel Database, Source Term Report and Chemical Reactivity Analysis Activities

D-1. DOE SPENT FUEL DATABASE

The National Spent Nuclear Fuel Program (NSNFP) maintains the U.S. Department of Energy's (DOE's) Spent Fuel Database (SFD). The SFD was developed to aid in the management of DOE's national and international inventory of spent nuclear fuel (SNF). The database provides a single source of DOE SNF data to make management decisions for handling, storage, transport, and disposal of SNF. The SFD contains quantitative and characteristic SNF information for all DOE-owned or managed SNF and provides the ability to search on a wide range of parameters. The database is updated periodically, and a reference version was generated to support the license application. The SFD serves as a source for best available existing information for all SNF under the purview of DOE. The SFD is a collection of information for all DOE-owned SNF. The SFD includes data fields that address the following categories:

- **Identifying Information**: Fuel name, number and description of fuel units, Record of Decision storage site, current location, and intended disposition.

- **Physical Characteristics**: Fuel cladding, compound, matrix, geometry, configuration, weight, volume, materials, and condition.

- **Burnup and Operating History**: Timelines of thermal or electric power production history; minimum, average, and maximum burnup, fissile material consumed, and decay heat.

- **Heavy Metal and Isotopic Mass Loading**: Beginning-of-life and end-of-life masses of uranium, plutonium, and thorium; and inventories for 145 radionuclides (estimated as described in D-2).

- **Unusual Fuel Conditions:** Identification of modifications to the original fuel configuration and unique events that may have led to fuel damage during reactor operations, experimental examination, or subsequent handling.

The SFD architecture is more fully described in References 1 and 2. In addition to DOE research and materials production reactors, records include fuels from selected foreign research reactors and non-DOE-owned domestic research reactors (e.g., university reactors), and all commercial fuels in the DOE custody. The SFD is used by several organizations including the DOE sites, Headquarters, and Yucca Mountain Project personnel who need information about SNF. Fuel information sources include fuel fabrication records, data supplied by the irradiating reactor site, and other technical documents. Information used to create and maintain records in the SFD is obtained from SNF storage sites. The sites responsible for the SNF periodically provide updated fuel information as appropriate. The data are checked against Nuclear Materials Safeguards and Security records and Material Control and Accountability records.

SFD Version 5.0.1, released December 11, 2003, was placed under configuration control and designated as the version to support Yucca Mountain Project information needs.[3] The SFD is controlled in accordance with NSNFP Procedure 19.02, "Management of the Spent Fuel Database."

D-2. SOURCE TERM REPORT

DOE is responsible for storage and final disposition of nuclear fuels that span several decades of nuclear research and defense-related material production. To support nuclear nonproliferation objectives, DOE has also taken custody of many foreign research reactor fuels. Therefore, DOE SNFs come from a wide range of reactor types (such as light and heavy water moderated reactors, graphite-moderated reactors, breeder reactors) with various fuel compounds, cladding materials, and enrichments. Many of these reactors, now decommissioned, had unique design features, such as core configuration, fuel element and assembly geometry, reflector and coolant materials, operational characteristics, and neutron spatial and spectral properties. These fuels have been safely handled and stored for many years at DOE storage facilities using existing information.

As an alternative to reliance on existing information, a methodology was developed to generate a conservative source term estimate for DOE SNF. DOE SNF radionuclide source terms were generated to support licensing analyses for the repository. Source term calculations provide estimates of radionuclide inventories that are used in the calculation of decay heat for thermal analyses of casks and storage canisters, photon emission spectra for shielding calculations, and radionuclide doses associated with preclosure and postclosure repository safety analyses.

D-2.1 Source Term Methodology

The source term methodology was developed by a team of experts representing each DOE storage site (i.e., Hanford, Savannah River Site, and the INEEL). The methodology is based on calculational techniques that have been successfully applied at the storage sites[4] supplemented by the application of similarity principles to bin fuels into groups that can employ precalculated ORIGEN outputs to model the generation of activation products and transuranics at a range of decay times.[5] These precalculated results are then used as a template that can be scaled to account for differences between fuel mass and burnup of the template, and the fuel being estimated. The template is selected based on matching the reactor moderator, the fuel cladding, the fuel compound, and the fuel enrichment with those of the fuel being estimated. The reasons for choosing these four parameters are twofold. First, sensitivity studies show that these four play a key role in establishing the neutron energy spectrum within the core, which strongly influences activation and transmutation. Secondly, these four parameters are known for most SNFs. When not known, conservative assumptions can often be made.

By modeling various combinations of reactor moderator, fuel enrichment, fuel compound, and fuel cladding; templates have been developed to reasonably model a broad range of DOE SNF. These templates provide inventories for 145 radionuclides at 10 different decay periods, ranging from 5 to 100 years following irradiation. To estimate an SNF source term, an appropriate template is selected to model the production of activation products and transuranics by matching the four selected parameters. Conservative assumptions were applied where needed. Precalculated radionuclide inventories are extracted from the selected template at the desired decay period and then scaled to account for differences in fuel mass and specific burnup.

Where burnup information was not available, conservative assumptions were used. Consequently, the methodology includes an algorithm for estimating burnup, using available information that, in some cases, may consist of no more than the end-of-life heavy metal mass.

The source term methodology provides radionuclide inventories and the associated source term, decay heat, and photon emission rates to be estimated for virtually any SNF for decay dates up to 100 years following reactor shutdown. The methodology relies on precalculated ORIGEN results to represent other similar DOE SNF.

A spreadsheet application was developed to facilitate application of the methodology and has been employed to estimate radionuclide inventories for several hundred types of DOE SNF. The results (along with a summary of the methodology, inputs, assumptions, and calculations used in the estimates) are available in Reference 6. The source term report was provided to the Yucca Mountain Project personnel to support design and licensing needs relative to receipt, handling, and disposal of DOE SNF. Results cited or referenced in the preclosure safety and total system performance assessment sections of this report are based on radionuclide inventories produced by the source term methodology. The described methodology for estimating DOE SNF radionuclide inventories has been incorporated directly into Version 5.0.1 of the SFD.[7]

D-3. CHEMICAL REACTIVITY ANALYSIS

Other preclosure safety considerations included the possibility of a fire, resulting from a pyrophoric reaction involving uranium metal SNF. Such a reaction could be initiated if oxygen interacts with uranium hydride on the SNF surface. The repository staff conducted simplified analyses to assess the potential impacts of this event sequence. Analysts conservatively assumed ignition and complete combustion of a multi-canister overpack (MCO) full of N-reactor SNF in the repository surface facilities. No credit was taken for HEPA filtration or other natural and engineered barriers to mitigate the consequences. These analyses indicated that the potential energy and radiological materials released from this event were unacceptable. Therefore, chemical reactivity analyses were undertaken to develop a more accurate assessment of the potential for pyrophoric reaction associated with receipt and handling of uranium metal SNFs.

The following other work was performed in support of these analyses:

1. Laboratory studies were performed at Argonne National Laboratory-West[8] in Idaho and the Y-12[9] facilities in Tennessee to better define the kinetics and thermodynamics of the uranium dehydriding reaction.

2. An existing computer code was modified and used to model different chemical and physical parameters describing the uranium hydride reactions and resulting heat transfer to the bulk uranium metal in SNF.[10] As discussed in Reference 11, GOTH_SNF is a derivative of the thermal-hydraulic code GOTHIC, which was previously accepted by the Nuclear Regulatory Commission.

The studies postulated the occurrence of nonmechanistic failures generating holes at the bottom and top of a vertical SNF package. These holes allowed oxygen in the atmosphere to reach the uranium hydride and begin the reaction. The question to be answered was how large must the holes be to allow the reaction temperatures to exceed a specified limit (i.e., ignition temperature of the bulk fuel metal). The results of the studies (see Reference 11) indicate that the reaction potential of the SNF package is a function of the assumed breach size in the canister wall. Studies were conducted to determine the effects of unequal hole sizes (i.e., one larger than the other, either the high hole or the low hole). No analysis was done to determine the impact response of an MCO. Further work is required to evaluate the size of canister breaches resulting from an impact to the canister.

These studies concluded that the controlling mechanism for the reaction is the gas flow through the smaller hole of a two-hole set, regardless of whether it is the inlet or the outlet hole. Hole sizes up to 0.75 in. in diameter maintained a controlled excursion with peak temperatures of less than the defined limit of 1,200°F for uncontrolled reaction and combustion. The 1,200°F-limit was based on the demonstrated range of the kinetic data available to support the computer analysis. Above this temperature, the computer code was not verified to yield correct results, and it is known qualitatively that other unmodeled reactions will begin to affect the results. Thus, an arbitrary definition of combustion in this zone was made.

Studies also modeled the effect of an open top MCO, which allowed unrestricted gas flow through the system. The open top MCO analysis showed a very rapid oxidation reaction with extremely high temperatures of reaction. However, a flow mechanism through the SNF configuration is required for this event to occur. Such a flow was assumed for the analysis.

These computer modeling studies were independently reviewed[12–14] by personnel from the Oak Ridge National Laboratory. A second set of computer simulations was run to address modeling concerns raised by this review. These runs, using the modified technical assumptions requested by the Oak Ridge reviewers, corroborated the original analysis (see Reference 11, Appendix C), and no changes to the original conclusions were necessary.

As the licensing strategy for DOE SNFs matured, reliance on preventing canister breach came to the forefront of the strategy. In addition to preventing any radiological release, preventing canister breach also precludes the introduction of oxygen and the attending possibility of a pyrophoric reaction. Although not directly used to make the safety case for licensing, the results and conclusions of the chemical reactivity analyses provide additional risk insights to help ensure safe repository disposition of DOE SNF.

D-4. REFERENCES

1. Steven Henline (INEEL) letter to Allan Bringhurst (INEEL), "Spent Fuel Database Software Design Description," SPH-01-99, August 24, 1999.

2. Spent Fuel Database Software Requirements Specifications. Signed by Robert K. Smith (September 3, 2003) David Peterson (September 4, 2003) and Allan Bringhurst (September 22, 2003).

3. P. D. Wheatley (Bechtel BWXT Idaho) letter to Steve Clark (BSC), "Transmittal of DOE/SNF/REP-078 Rev 1 Source Term Estimates for DOE Spent Nuclear Fuels," January 16, 2004.

4. U.S. Department of Energy, Environmental Management, *Methodologies for Calculating DOE Spent Nuclear Fuel Source Terms*, DOE/SNF/REP-055, January 2000.

5. U.S. Department of Energy, Environmental Management, *Guide for Estimating DOE Spent Nuclear Fuel Source Terms*, DOE/SNF/REP-059, July 2000.

6. U.S. Department of Energy, Environmental Management, *Source Term Estimates for DOE SNFs (Volumes I–III)*, DOE/SNF/REP-078, Rev. 1, January 2004.

7. INEEL, National Spent Nuclear Fuel Program, "Check of SFD v5.0.1 Radionuclide Calculations," EDF-035, December 2003.

8. *Oxidation of ZPPR Fuel Corrosion Products: National Spent Fuel Program FY 1999 Final Report*, ANL-99/21, October 1999.

9. "Reaction of Oxygen with Uranium Hydride," Y/DZ-3501, December 2003.

10. U.S. Department of Energy, Environmental Management, *MCO Work Book GOTH_SNF Input Data*, DOE/SNF/REP-071, Rev. 2, January 2004.

11. U.S. Department of Energy, Environmental Management, *GOTH_SNF Chemical Reactivity Final Analysis*, DOE/SNF/REP-077, Rev. 1, March 2003.

12. U.S. Department of Energy, Environmental Management, *Review of Flow Models in GOTH_SNF for Spent Fuel MCO Calculations*, DOE/SNF/REP-087, September 2003.

13. U.S. Department of Energy, Environmental Management, *Review of Heat Transfer Models in GOTH_SNF for Spent Fuel MCO Calculations*, DOE/SNF/REP-088, September 2003.

14. U.S. Department of Energy, Environmental Management, *Review of Uranium Hydriding and Dehydriding Rate Models in GOTH_SNF for Spent Fuel MCO Calculations*, DOE/SNF/REP-089, September 2003.