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Options for Pursuing Moderator Exclusion for Application to Spent-Fuel Transportation Packages

Technical Report

Options for Pursuing Moderator Exclusion for Application to Spent-Fuel Transportation Packages

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REPORT SUMMARY

This report discusses options for pursuing moderator exclusion, either by itself or in combination with burnup credit, for application to the criticality evaluation of spent nuclear fuel transportation packages. Also, information is provided on how to proceed in developing a request for rulemaking if the industry determines that changes to the existing regulations for streamlining implementation of moderator exclusion are highly desirable.

Background
Evaluating the nuclear criticality safety of transportation packages for spent commercial light water reactor fuel must be performed with a fully flooded cask containment per 10 CFR 71.55(b) and per 10 CFR 71.55(e) under the tests specified in §71.73 ("Hypothetical accident conditions"). Exceptions to the requirement for the fully flooded condition ("moderator exclusion") are provided in 10 CFR 71.55(c) and in Spent Fuel Project Office's Interim Staff Guidance 19 (SFPO ISG-19). 10 CFR 71.55(c) stipulates that the Nuclear Regulatory Commission (NRC) may approve exceptions to the requirements of 10 CFR 71.55(b). Exceptions are allowed if the package incorporates special features that ensure that no single packaging error would permit leakage and if appropriate measures are taken before each shipment to ensure that the containment system of the transportation package does not leak. However, it is worth noting that no evidence has been found to show that NRC has ever granted such an exception for a spent nuclear fuel package under 10 CFR 71.55(c). SFPO ISG-19 provides review guidance for meeting the fissile material package requirements in 10 CFR 71.55(e). When considering high burnup spent fuel, applicants may choose an approach based on "moderator exclusion," which includes physical testing of the water exclusion boundary. However, to date, no applicant has pursued this option under SFPO ISG-19.

Objectives
To examine existing options for pursuing moderator exclusion for application to the criticality evaluation of spent-fuel transportation packages, including one-time shipment of storage-only systems and transport of contemporary dual-purpose and future-design systems; to examine benefits and risks of pursuing rulemaking that would appropriately modify provisions applicable to spent nuclear fuel in 10 CFR 71.55 to enable use of moderator exclusion, not as an exception, but as a full-fledge option; and, alternatively, to examine how moderator exclusion may be possible within the current regulatory framework of §71.55 through an interpretation of these regulations and the development of guidance documents to allow moderator exclusion under certain conditions in which it can be demonstrated that water inleakage is not *credible*.

Approach
The research team reviewed U.S. regulations (10 CFR 71) and regulatory guidance (Standard Review Plan, NUREG-1617; SFPO ISG-8, -11, and -19) as well as International Atomic Energy

Agency regulations (IAEA TS-R-1) governing transport of spent nuclear fuel for their relevance to moderator exclusion. They then evaluated benefits and risks of changing U.S. regulations through rulemaking. The team applied the moderator exclusion options, with burnup credit as a backup or in combination with burnup credit, to canister/welded and metal/bolted systems. Options for the following cases were assessed: (1) one-time shipment of storage-only systems, (2) transportation of already-loaded dual-purpose systems, and (3) transportation of future transportation systems, including advanced design systems.

Results

Moderator exclusion may be applicable singly or in combination with burnup credit. Of the two options, moderator exclusion seems to hold the promise of an easier, less-costly path to success, particularly for advanced and next-generation technology. However, for general application of moderator exclusion, rulemaking may be required to relieve the NRC of having to use the exception approach to certification. Moderator exclusion may be possible within the current regulatory framework through an interpretation of current regulations and development of guidance documents to allow moderator exclusion under certain conditions in which it can be demonstrated that water inleakage is not credible.

For transport of storage-only systems through the use of exemptions or exceptions to 10 CFR 71, cask designers may consider NRC approval of a special arrangement; the exemptions/exceptions will be part of the overall safety evaluation for the package. If moderator exclusion is considered by cask designers as part of their overall arguments regarding the equivalent safety of these already-loaded storage-only systems, existing regulations contained in §71.55(c) can be relied on rather than relying on NRC guidance in SFPO ISG-19, at least for spent-fuel burnups less than 45 GWd/MTU. For advanced dual-purpose systems, moderator exclusion may be used in combination with burnup credit under existing guidance in SFPO ISG-19 or possibly through provisions contained in §71.55(c). If it appears that most advanced dual-purpose systems will require exemptions/exceptions under existing regulations, there may be a near-term need to consider a change in the regulations through rulemaking. Alternatively, a change may be needed in interpreting existing rules and developing guidance documents to allow moderator exclusion under certain conditions in which it can be demonstrated that water inleakage is not credible.

EPRI Perspective

Earlier versions of this report were distributed for comments among stakeholders with a strong interest in streamlining the approach to moderator exclusion in spent-fuel package criticality evaluation. Stakeholders include spent-fuel storage installation and transportation package licensees, storage and transportation systems designers, and industry and government groups. This final report was finalized taking stakeholder comments into account. Information in this report is intended to inform decision-makers about options that may exist with regard to pursuing moderator exclusion for application to spent-fuel transportation packages.

Keywords

Spent nuclear fuel   Transportation   Criticality   Moderator exclusion   Burnup credit

EXECUTIVE SUMMARY

This report discusses the concept of moderator exclusion, either by itself or in combination with burnup credit, for application to the criticality safety evaluation of spent nuclear fuel transportation packages. Information is also provided on how to proceed in the development of a request for rulemaking if the industry determines that changes to the existing regulations for streamlining implementation of moderator exclusion are highly desirable. Additionally, industry and NRC might also consider whether moderator exclusion may be possible within the existing NRC regulations, specifically those in §71.55, through an interpretation of these regulations and development of guidance that would allow moderator exclusion with no changes in the existing regulations.

Background

Evaluation of the nuclear criticality safety of transportation packages for spent commercial light water reactor fuel must be performed with a fully flooded cask containment per 10 CFR 71.55(b), and also per 10 CFR 71.55(e) under the tests specified in §71.73 ("Hypothetical accident conditions"). Exceptions to the requirement for the fully flooded condition, i.e., "moderator exclusion," are provided in 10 CFR 71.55(c) and in Spent Fuel Project Office's Interim Staff Guidance 19 (SFPO ISG-19). 10 CFR 71.55(c) stipulates that the Nuclear Regulatory Commission (NRC) may approve exceptions to the requirements of 10 CFR 71.55(b) if the package incorporates special features that ensure that no single packaging error would permit leakage and if appropriate measures are taken before each shipment to ensure that the containment system of the transportation package does not leak. However, it is worth noting that no evidence has been found to show that the NRC has ever granted such an exception for a spent nuclear fuel package under 10 CFR 71.55(c). SFPO ISG-19 provides review guidance for meeting the fissile material package requirements in 10 CFR 71.55(e). When considering high burnup spent fuel, the applicant may choose an approach based on "moderator exclusion", which includes physical testing of the water exclusion boundary. However, to date, no applicant has pursued this option under SFPO ISG-19.

Report Objectives

- To examine existing options for pursuing moderator exclusion for application to the criticality evaluation of spent fuel transportation packages, including one-time shipment of storage-only systems and transportation of contemporary dual-purpose and future design systems

- To examine the benefits and risks associated with pursuing rulemaking that would appropriately modify the provisions applicable to spent nuclear fuel in 10 CFR 71.55 to enable the use of moderator exclusion, not as an exception, but as a full-fledged option. Or alternatively, to examine how moderator exclusion may be possible within the current regulatory framework of §71.55 through an interpretation of these regulations and the development of guidance documents to allow moderator exclusion under certain conditions in which it can be demonstrated that water inleakage is not credible.

Discussion

Storage-Only Systems

The older storage-only systems can benefit from applying moderator exclusion technologies, but there are other issues with older systems besides criticality safety. System shipping may need to be made under some form of exemption/exception or special arrangement where moderator exclusion might be one element in a comprehensive argument on equivalent transportation safety.

If moderator exclusion is considered by cask designers as part of their overall arguments regarding the equivalent safety of these already-loaded storage only systems, the existing regulations contained in §71.55(c) and in §71.55(e) might be relied upon rather than relying on NRC guidance in SFPO ISG-19, at least for spent-fuel burnups less than 45 GWd/MTU. External package modifications might be required to satisfy specific shipping requirements.

Obtaining NRC approval for transportation will be challenging. Cost, risk and exposure studies should be made between the exemption/exception and the repackaging options.

Dual-Purpose Systems

The first generation of dual-purpose cask systems were licensed for both storage and transport. The original designs of most dual-purpose cask systems have been licensed for both storage and transport without the need for either moderator exclusion or burnup credit. The more advanced dual-purpose systems, with higher capacities that have been designed to store and transport high initial enrichment spent fuel with high burnups (i.e., >45 GWd/MTU) can clearly benefit from moderator exclusion. Moderator exclusion may be used in combination with burnup credit under the existing guidance in SFPO ISG-19 or possibly through the provisions contained in §71.55(c) and §71.55(e). If it appears that most advanced dual-purpose systems will require exceptions under the existing regulations, there may be a near-term need for NRC to consider either (1) an interpretation of §71.55 and the development of guidance documents to allow moderator exclusion under certain conditions in which it can be demonstrated that water inleakage is not credible or (2) rulemaking to change §71.55 to remove the “exception” language.

The application of moderator exclusion to advanced dual-purpose systems has a very good chance of success, especially in light of SFPO ISG-19. Whether through rulemaking or within the current regulatory framework, logic would suggest that moderator exclusion could be implemented with no real reduction in safety.

Next Generation Casks

Next generation transport casks, whether dual-purpose or transport-only systems, will evolve from the advanced designs now being evaluated by the NRC. Next generation systems must be capable of transporting spent fuel with assembly-average burnup up to ~60 GWd/MTU, short cooling times (high decay heat), initial enrichments up to 5.0 w/o, and advanced cladding materials. Cask designers will require advances in technologies and methods in order to design transport systems that can handle the wide range of SNF that will need to be moved to a repository. The ability to use moderator exclusion in criticality safety assessments will be a valuable tool for these next generation systems.

If changes are made to §71.55 to allow moderator exclusion without the need for an exception, there may be additional requirements that will be imposed on package design features or quality assurance requirements. The interpretation of §71.55 to allow moderator exclusion under certain conditions in which it can be demonstrated that water inleakage is not "credible" may provide an alternative to rulemaking. While rulemaking would be time consuming, it would leave no ambiguity in the rules. Interpretation of existing regulations would allow for near-time compliance but must deal with the "optics" of exceptions to the rules even though such exceptions are allowable. Either way the next generation systems will benefit. In fact, in most cases considering the fuel to be shipped, moderator exclusion will be necessary in order to preserve the efficiency of high capacity casks.

Conclusions

As this report demonstrates, moderator exclusion and burnup credit may be applicable singly or in combination. Of the two, moderator exclusion seems to hold the promise of a less-costly path to success, particularly for advanced and next-generation technology. For general application of moderator exclusion, it seems clear that 10 CFR 71 rulemaking may be required to relieve the NRC of having to use the exemption/exception approach to certification. However, one should not dismiss the possibility of allowing moderator exclusion within the existing regulatory framework of §71.55 and amending applicable guidelines to permit moderator exclusion under certain conditions in which it can be demonstrated that water inleakage is not credible.

If moderator exclusion is determined to be of value to the industry to assist in the transport of advanced dual-purpose designs and next-generation designs, then it is suggested that the nuclear industry and NRC begin to discuss how NRC would proceed with either (1) an interpretation of §71.55 and development of guidance to allow moderator exclusion under certain conditions, such as a recognition of design features that result in moderator intrusion being not "credible" under both normal and accident conditions (allowing water inleakage into the containment system but not into the fuel region may be one aspect of this alternative), or (2) rulemaking to change §71.55 to remove the "exception" language. Underlying these discussions should be an acknowledgement that the risks associated with a criticality accident are extremely small and would not result in any significant increase in transport risks. If it is decided that rulemaking should be pursued, rulemaking activities should commence as soon as possible since the rulemaking process and any ensuing administrative issues may be time consuming and complex. Alternatively, if moderator exclusion within the current regulatory framework of §71.55 appears to show promise, industry and NRC should begin dialogue to determine under what conditions moderator exclusion would be allowed and to develop associated guidance documents to facilitate implementation.

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INTRODUCTION

1.1 Background

Spent nuclear fuel (SNF) shipping cask designs have evolved significantly over the past four decades. In the 1960s and 1970s, cask designs supported the chemical separation (i.e., reprocessing) of SNF. The characteristics of SNF that could be transported in these early transport casks included modest initial enrichments and burnups [e.g., 3.5 weight percent (w/o) U-235 and 35 gigawatt-days/metric ton of uranium (GWd/MTU)] and relatively short cooling times. Due to the shorter cooling times, fuel assembly decay heat was substantially higher than that allowed in today's cask designs. For example, the circa-1975 NLI-10/24 rail cask was designed to carry 10 pressurized water reactor (PWR) assemblies or 24 boiling water reactor (BWR) assemblies with a total package decay heat of 70 kilowatts, or 7 kW/PWR assembly. As a consequence, heat load and its companion radiation source strength were the limiting factors in determining cask capacity at that time. In contrast, storage and transport cask designs deployed in the 1990s through today have decay heats of approximately 1 kW/assembly or less, with some newer designs with heat loads as high as 2.0 kW/assembly.

With the demise of commercial reprocessing in the late 1970s and the passage of the Nuclear Waste Policy Act of 1982 (NWPA), the character of the back-end of the fuel cycle changed. Nuclear operating companies began planning for long-term, on-site storage of SNF, leading to the development of on-site dry storage technologies. It became clear to the industry and to the regulators that high-capacity storage and transportation packages would be necessary to safely and efficiently accommodate projected SNF fuel volumes.

The desire to achieve higher capacity storage and transport casks required an increase in the SNF cooling times for both storage and transport. Although decay heat remained a primary parameter limiting cask capacity, criticality safety became a dominant consideration in designing transportation casks. In the early transport package designs that allowed high fuel assembly decay heat, fuel assemblies were contained in a fairly open basket lattice. In fact, most early transport casks were water-filled to promote internal cooling, and, therefore, moderator was present in the cask. Criticality control was accommodated by incorporating specific neutron poisons in the cask interior-supporting-structure (or "basket") and by the use of flux traps to provide efficient neutron capture.

The current dual-purpose storage and transport technologies evolved due to demands for more efficient SNF storage, including the capability to transport SNF in dry storage without the need to repackage. Most dual-purpose cask systems include a sealed, high-capacity steel canister that is stored within a shielded, ventilated concrete structure for onsite storage. For transportation, the canister is remotely removed from the storage structure and transferred into a transportation overpack for shipment offsite. SNF cask/canister capacities have jumped from the 10 PWR or 24 BWR assemblies from the 1970s to as many as 40 PWR or 87 BWR assemblies for dual-purpose cask designs currently deployed today or under U.S. Nuclear Regulatory Commission (NRC) review. These higher capacity casks are more cost-effective to load as fewer casks are needed to store and transport SNF. In addition, higher capacity transport casks result in fewer shipments which reduce accident risk and lower routine radiation exposures. Thus, there are clear incentives to nuclear operating companies to increase storage and transport cask capacities consistent with design and regulatory criteria. However, these higher capacity systems have presented challenges to cask designers with regard to heat transfer, shielding and criticality safety. While thermal and shielding issues can be tempered by increased cooling time, criticality issues do not have a time-based solution.

1.2 Transport Cask Design Limitations

Today's cask and associated canister designs focus on increasing the number of SNF assemblies that can be stored and transported. Cask capacity is typically driven by cask system weight. That is, there is an upper bound for a loaded transfer or shipping cask weight that is set by crane lifting capabilities at the nuclear power plant sites; a 125-ton lifting capacity is a relatively standard maximum. The practical upper bound on total system weight (i.e., ~250,000 pounds) results in a cask or canister inside diameter of approximately 70 inches. This inside diameter will vary depending upon the specific design and whether or not a canister is used. To optimize cask efficiency, the cask designer must address the challenge of "packing density" of the fuel. This challenge involves locating fuel assemblies as close to each other as is feasible, while complying with the structural, thermal, shielding and criticality safety criteria derived from the Federal regulations contained in Title 10, U.S. Code of Federal Regulations Part 72 (10 CFR 72 for storage) and Part 71 (10 CFR 71 for transport). Of these criteria, criticality safety has the greatest effect in limiting the packing density or cask capacity. The reason for this is that the criticality safety evaluation requires a series of regulatory-prescribed assumptions and practices that together result in an extraordinarily conservative design.

1.3 Criticality Safety Evaluation

The regulations and accepted practices currently applied to NRC-certified transport casks are listed below:

1. Subcriticality is assured, that is, the neutron multiplication factor (k_{eff}) < 1.
2. Moderation by water occurs to the most reactive credible extent.
3. Full reflection of the system on all side by water occurs.
4. The system is in its most reactive credible configuration consistent with the chemical and physical form of the material.
5. The allowed k_{eff} is then reduced from 1 to account for such things as modeling and calculational biases and uncertainties.
6. Additionally, the allowable k_{eff} is further reduced by applying an additional arbitrary criticality safety margin of 5%, i.e., $\Delta k_{eff} = -0.05$.
7. The fuel is assumed to be in its most reactive state, which is generally unburned. This is known as the fresh fuel assumption.

8. The fuel is positioned eccentrically within the basket in its most reactive configuration.
9. Credit may be taken for only 75% of theoretical boron content in fixed neutron poisons unless it can be demonstrated to be greater.

Items (1) to (4) above are part of the regulations contained in 10 CFR 71.55. The current NRC application of Item (2) requires that leakage of water into the transport package containment system be assumed under any normal or accident conditions. This could be described as the "moderator intrusion" requirement. Options for applying or modifying NRC regulations contained in §71.55 and in NRC guidance documents to allow the use of moderator exclusion are the topic of this report.

Item (5) is simply an acceptable approach to criticality safety analysis. Item (6) is an additional safety factor applied to the analysis.

Item (7), the fresh fuel assumption, is the practice that is modified when burnup credit is used. The modified item (7) might read, "The fuel is assumed to be in its most reactive arrangement, after credit for the spent fuel's burnup is determined using conservative depletion analysis." This is briefly discussed in the next section.

NRC guidance for criticality safety assessment recommend the application of Items (8) and (9) as assumptions in the criticality safety analysis.

Taken together, these assumptions limit both the number of fuel assemblies and the enrichment of the fuel assemblies that can be loaded and transported in a specific transport cask design.

1.4 Burnup Credit

To support development of advanced technology spent fuel transportation casks, the U.S. Department of Energy (DOE) began to pursue the use of burnup credit in the mid-1980s. The approach that DOE planned to follow in its pursuit of burnup credit was first presented to the NRC at a DOE-sponsored workshop held in February 1988. At the time, the DOE strategy was to seek NRC approval of "full" burnup credit that would cover the range of initial enrichments and burnup values of all spent fuel in the anticipated inventory. Although the full burnup credit approach was not actually intended to take credit for all possible negative reactivity that could be attributed to burnup, it would take credit for the negative reactivity that could be readily justified. That is, it would account for all fissile actinides, most neutron absorbing actinides, and a small number of fission products that accounted for about 80% of the available credit for all fission products. The use of "full" burnup credit would allow for very efficient assembly "packing" and would markedly reduce the need for specific neutron poison components, even for fuel assembly enrichments up to 5 w/o U-235.

During the early 1990s, DOE and NRC began formal discussions on burnup credit methodology. At that time, the NRC did not allow burnup credit but suggested to the DOE that in the near-future it would only consider the use of the depletion/buildup of fissile isotopes in evaluating burnup credit methodologies – referred to as Actinide-Only Burnup Credit. On that basis, the DOE submitted an Actinide-Only Burnup Credit Topical Report in 1995 for NRC review and approval. The final version, Revision 2, was submitted in 1998. The use of actinide-only burnup credit, although allowing some improvement in efficiency for multi-element truck casks, fell far short in terms of permitting efficient packing density at the expected fuel enrichments and exposures for larger rail casks. To facilitate the use of actinide-only burnup credit by applicants, the NRC's Spent Fuel Project Office (SFPO) issued Interim Staff Guidance (ISG) 8, entitled "*Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks*." It is currently up to Revision 2.

Although SFPO ISG-8 is a significant step forward, its application in cask design yields a marginal improvement of the cask efficiencies compared with that which would be achievable if full burnup credit (i.e., allowing credit for fission product buildup plus actinide depletion) was allowed. Actinide-only burnup credit has substantial technical and operational limitations and implementation costs for only marginal gains. As a result, to date no cask designs which utilize actinide-only burnup credit have been advanced. The path to achieving the acceptance of full burnup credit remains an important objective for industry although it is technically challenging and potentially costly as discussed in an earlier EPRI report, "*Fission Product Benchmarking for Burnup Credit Applications*," completed in 2002.[1] The challenges associated with obtaining regulatory acceptance of burnup credit have caused cask designers to examine other potential solutions to achieving the transport of SNF with higher enrichments and burnups in high-capacity casks. One solution approach that is thought to have great potential is called moderator exclusion.

1.5 Moderator Exclusion

In the absence of water moderation, even new fuel assemblies in any array would remain subcritical. Therefore, in the absence of water, the potential for a critical configuration is non-existent, and considerations related to criticality safety are not limiting with regard to transport package capacity.

The concept of moderator exclusion has been recognized by NRC packaging and transportation regulations for decades as later discussed in Section 2. For example, within Subpart 71.55 – *General requirements for fissile material packages*, §71.55(c) allows the NRC to approve exceptions to the water in-leakage assumption of §71.55(b). To be granted an exception, an applicant must demonstrate to the NRC's satisfaction that (1) the package incorporates special design features that ensure that no single packaging error would permit in-leakage of water, and (2) appropriate measures are taken before each shipment to ensure that the containment system does not leak. *It is worth noting that no evidence has been found to show that the NRC has ever granted such an exception for a SNF package.* §71.55(e) addresses criticality under accident conditions. While the regulations discuss water moderation to the most reactive "credible" extent, current regulatory guidance requires an assumption of full moderation without regard to its credibility.

In 2003, in response to the issue of demonstrating subcriticality of high burnup SNF (i.e., >45 GWd/MTU) under accident conditions, the NRC issued SFPO ISG-19 – *Moderator Exclusion under Hypothetical Accident Conditions and Demonstrating Subcriticality of Spent Fuel under the Requirements of 10 CFR 71.55(e).* The issuance of this ISG was initially encouraging for applicants as it appeared that the NRC might be more willing to consider moderator exclusion than in the past.

[1] *Fission Product Benchmarking for Burnup Credit Applications.* Progress Report, EPRI, Interim Report, December 2002. EPRI 1002879.

SFPO ISG-19 was written to address the shipment of SNF with an emphasis on fuels with burnups greater than 45 GWd/MTU. This is based on the fact that sufficient evidence exists to provide reasonable assurance that fuel burned to lesser levels will remain structurally intact under hypothetical accident conditions, and will, therefore, be bounded by the criticality analysis performed to demonstrate compliance with §71.55(b). This is not to suggest that moderator exclusion has no application to fuels of less than 45 GWd/MTU. According to SFPO ISG-19, post-accident fuel geometry is judged to be indefinable at this time for criticality computations because of insufficient knowledge of fuel cladding material properties for SNF with greater than 45 GWd/MTU burnup. SFPO ISG-19 offers a reconfiguration alternative where an applicant can define a *bounding* reconfigured fuel geometry for computational purposes as long as it does not strictly rely on material properties.

Additionally, the NRC offers a moderator exclusion alternative to the reconfigured fuel alternatives in SFPO ISG-19. The moderator exclusion option involves physical testing of representative structures to measure leak tightness. One can conclude that the concept of moderator exclusion has been taken a step further by the release of SFPO ISG-19. At this time, no applicant has elected to apply for moderator exclusion under SFPO ISG-19. From a practical standpoint, casks are designed to accommodate fuel with burnups that are both greater and less than 45 GWd/MTU in the same loading. Thus, if §71.55(b) must still be satisfied, the benefits of employing SFPO ISG-19 may be diminished unless the moderator exclusion alternative offered in ISG-19 can be applied to fuel with any burnup.

1.6 Moderator Exclusion Implementation

In summary, transportation package criticality calculations for commercial light water reactor fuel must be performed with a flooded package per 10 CFR 71.55(b) and historically have been performed with a flooded package to comply with §71.55(e), i.e., with moderator *intrusion*. Moderator exclusion could, in principle, be implemented by one of the following alternatives:

1. Moderator exclusion through successfully satisfying §71.55(c), which allows an exception to §71.55(b). This approach provides an alternative to §71.55(b), but the requirements of §71.55(e) still need to be satisfied. This alternative is presently available under the current regulatory framework.
2. Moderator exclusion through successfully satisfying the acceptance criteria contained in SFPO ISG-19. This approach provides a "moderator exclusion" option for demonstrating compliance with §71.55(e) for spent fuel with burnup greater than 45 GWd/MTU, but the requirements of §71.55(b) still need to be satisfied. This alternative is presently available under current regulatory framework.
3. Moderator exclusion through interpretation of §71.55 and the associated guidance to allow an exception from §71.55(b) via §71.55(c), and to recognize design features or other conditions that make water intrusion not "credible" under accident conditions thus satisfying §71.55(e). This alternative is presently available under the current regulatory framework although it is thought that there may be reluctance to approve exceptions under §71.55(c).

4. Moderator exclusion through successfully satisfying both §71.55(b) and (e), but with fewer regulatory restrictions than presently embodied in §71.55(c) and SFPO ISG-19. For example, §71.55(c) would specify acceptance criteria to which the package would need to conform without reference to being an "exception," and SFPO ISG-19 applied to spent fuel of any burnup. To achieve this, it will be necessary to supplement the design rules and acceptance criteria possibly through rulemaking. This alternative is not presently available under current regulatory framework.
5. Moderator intrusion into the containment system but exclusion from the fuel cavity for canister-based transport packages. This approach recognizes that 10 CFR 71.63 no longer requires a separate inner container for plutonium. This alternative may be implemented under the current regulatory framework if it can be shown that the internal canister makes water intrusion into the fuel cavity not "credible".

Of the alternatives discussed above, two were highlighted during an industry-NRC discussion at the May 2005 NEI Dry Storage Information Forum. These two alternatives include:

- "Rulemaking to modify 71.55(c) to eliminate 'exception' wording
- Re-interpret 71.55(b) to allow leakage into 'containment system' but not into fuel region …"[2]

If moderator exclusion is determined to be of value to the industry to assist in the transport of advanced dual-purpose designs and next-generation designs, then it is suggested that the nuclear industry and NRC begin to discuss how NRC would proceed with (1) an interpretation of §71.55 and development of guidance to allow moderator exclusion under certain conditions, such as a recognition of design features that result in moderator intrusion being not "credible" under both normal and accident conditions (allowing water inleakage into the containment system but not into the fuel region may be one aspect of this alternative), (2) rulemaking to change §71.55 to remove the "exception" language, or both. If it is decided that rulemaking should be pursued, rulemaking activities should commence as soon as possible since the rulemaking process and any ensuing administrative issues may be time consuming and complex. In the alternative, if §71.55 and applicable guidance documents can be interpreted to recognize design features or conditions that make moderator intrusion not credible, industry and NRC should begin dialogue to determine under what conditions moderator exclusion would be allowed.

This report is intended to discuss significant aspects of moderator exclusion alone and moderator exclusion in conjunction with burnup credit, and to make recommendations regarding how NRC might grant moderator exclusion within the existing regulations or how industry and NRC might proceed in developing a petition for rulemaking should industry determine that an amendment to 10 CFR Part 71 is highly desirable.

[2] Hodges, Wayne, NRC SFPO, Technical Challenges Associated with Transportation, Panel Discussion, Nuclear Energy Institute, Dry Storage Information Forum, Key Biscayne, Florida, May 10, 2005.

2

OVERVIEW OF CURRENT REGULATIONS AND GUIDANCE

2.1 International Atomic Energy Agency Transport Safety Standards

The International Atomic Energy Agency (IAEA) is responsible for developing international transport safety standards for radioactive materials in conjunction with its Member States (which include the U.S.). The safety standards are, in turn, adopted by international transport organizations, such as the International Maritime Organization, and are used by many IAEA Member States as the basis for their national transport regulations. The standards apply to all aspects of the transport of radioactive materials, including: design, manufacture, maintenance and repair of packaging; preparation, consignment, loading, carriage, and storage incident to transport; and unloading and receipt at the final destination. The IAEA transport safety standards are reviewed on a two-year cycle and are revised as needed. Within the U.S., two federal agencies – the U.S. Department of Transportation (DOT) and the NRC – jointly regulate the transportation of radioactive materials. U.S. transportation safety regulations for radioactive materials are revised periodically to take into account changes in the IAEA transport safety standards.

IAEA Regulations for the Safe Transport of Radioactive Material, 1996 Edition (Revised), No. TS-R-1, provides the framework for approval of packages to transport radioactive materials. Paragraph 677, Assessment of an individual package in isolation, provides the IAEA requirements for water in-leakage for a package criticality analysis. This paragraph states that:

> *"(I)t shall be assumed that water can leak into or out of all void spaces of the package, including those within the containment system. However, if the design incorporates special features to prevent such leakage of water into or out of certain void spaces, even as a result of error, absence of leakage may be assumed in respect to those void spaces. Special features shall include the following:*
>
> *(a) Multiple high standard water barriers, each of which would remain watertight if the package were subject to the tests prescribed in para. 682(b) [the test under normal conditions of transport and accident conditions], a high degree of quality control in the manufacture, maintenance and repair of packagings and tests to demonstrate the closure of each package before each shipment"*

Paragraph 679 requires that the package remain subcritical under the conditions of paragraph 677 with the package conditions that result in the maximum neutron multiplication factor (k_{eff}) consistent with normal conditions of transport and under the hypothetical accident conditions specified in paragraphs 719 – 724 of TS-R-1.

Thus, it appears that TS-R-1 would allow moderator exclusion if the special features identified have been incorporated into the package design. However, to-date, there do not appear to be any SNF packages that have been approved by national regulatory authorities with a moderator exclusion assumption.

2.2 NRC Regulations for Packaging and Transport of Radioactive Materials

The Atomic Energy Act of 1954 gave NRC the authority to regulate the receipt, possession, use and transfer of radioactive materials. Regarding transportation of radioactive materials, the NRC is responsible for regulating users of radioactive materials and the design, manufacture, use, and maintenance of shipping containers for certain types of radioactive materials, including spent nuclear fuel. NRC's transportation-related regulations can be found in Title 10, U.S. Code of Federal Regulations within Part 71, "Packaging and Transportation of Radioactive Material" (10 CFR 71).

2.2.1 10 CFR Part 71.55

The NRC's regulations for packaging and transport of spent nuclear fuel, as contained in 10 CFR 71, define both normal shipping conditions and hypothetical accident conditions that SNF casks must be able to withstand, along with the acceptance criteria under these conditions. Criticality analyses for fissile material packages, including SNF packages, must analyze criticality under both normal conditions of transport and the hypothetical accident conditions that are specified in the regulations. The specific regulations that address criticality analyses for fissile material packages under normal conditions of transport and under the hypothetical accident conditions can be found in §71.55(b) through (e):

§71.55(b) requires that "Except as provided in paragraph (c) of this section, a package used for the shipment of fissile material must be so designed and constructed and its contents so limited that it would be subcritical if water were to leak into the containment system, or liquid contents were to leak out of the containment system so that, under the following conditions, maximum reactivity of the fissile material would be attained:

1. *The most reactive credible configuration consistent with the chemical and physical form of the material;*
2. *Moderation by water to the most reactive credible extent; and*
3. *Close full reflection of the containment system by water on all sides, or such greater reflection of the containment system as may additionally be provided by the surrounding material of the packaging."*

§71.55(c) provides that "The Commission may approve exceptions to the requirements of paragraph (b) of this section if the package incorporates special design features that ensure that no single packaging error would permit leakage, and if appropriate measures are taken before each shipment to ensure that the containment system does not leak."

§71.55(d) requires that "A package used for the shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in §71.71 ("Normal conditions of transport") –

1. *The contents would be subcritical;*
2. *The geometric form of the package contents would not be substantially altered;*
3. *There would be no leakage of water into the containment system unless, in the evaluation of undamaged packages under §71.59(a)(1), it has been assumed that moderation is present to such an extent as to cause maximum reactivity consistent with the chemical and physical form of the material; and*
4. *There will be no substantial reduction in the effectiveness of the packaging."*

§71.55(e) requires that "A package used for the shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in §71.73 ("Hypothetical accident conditions"), the package would be subcritical. For this determination, it must be assumed that:

1. *The fissile material is in the most reactive credible configuration consistent with the damaged condition of the package and the chemical and physical form of the contents;*
2. *Water moderation occurs to the most reactive credible extent consistent with the damaged condition of the package and the chemical and physical form of the contents; and*
3. *There is full reflection of water on all sides, as close as is consistent with the damaged condition of the package."*

Under current regulations and regulatory guidance, the NRC must approve exceptions to the regulations when a package is to be approved without an assumed inleakage of water into the containment system per §71.55(b) [through §71.55(c)], and without assuming full moderation under accident conditions per §71.55(e) (through SFPO ISG-19). One alternative that has been recommended is to interpret §71.55 to allow moderator exclusion under certain conditions such as recognition of package design features that make water intrusion not credible under both normal and accident conditions (allowing water inleakage into the containment system but not the fuel region may be one aspect of this alternative. Another alternative would be to change §71.55 through rulemaking to remove the "exception" wording. NRC has made it clear to the industry that it will not consider wholesale licensing of cask designs under the "exception" rule of §71.55(c).

2.2.2 Special Arrangements

Special arrangement is a term that comes from IAEA regulations but is not specifically defined in U.S. regulations. Under NRC regulations, the special arrangement is an infrequently used but available portion of 10 CFR 71 regulations. Specifically, §71.12 allows the NRC to grant an exemption from the requirements of Part 71. Further, in §71.41(c) the NRC may approve environment and test conditions that differ from the Normal Conditions of Transport (NCT) and the Hypothetical Accident Conditions (HAC). The applicant must demonstrate to the Commission's satisfaction that *"… controls proposed to be exercised by the shipper are demonstrated to be adequate to provide equivalent safety of the shipment."*

"Special arrangement" in its fullest sense has not been used for SNF shipment in the U.S. Many shippers have thought about the requirements that would produce an equivalent level of safety. Generally, the ideas come down to route selection and travel control. Route selection involves avoidance of obvious obstacles such as bridges, steep drops, populous areas, deep water, roadside hazards such as chemical plants and pipelines, and other potential threats to the package. Travel controls involve escorts, reduced speeds, hours of operation, time of year, communications, emergency response capabilities, and similar things. In addition, consideration of the reduced risk associated with the one-time shipment of a loaded canister could be considered under the provisions of a “special arrangement”. Also, exterior protective features may be added to the package being transported.

2.3 Related NRC Guidance

The NRC also publishes other documents that supplement the regulations to provide additional guidance to licensees and which must be taken into consideration during the design process. With respect to the design and certification of SNF transport packages, guidance documents include a Standard Review Plan (SRP) for Transportation of Spent Nuclear Fuel, Regulatory Guides, NRC technical documents (NUREG and NUREG/CR) publications, as well as Interim Staff Guidance (ISG) documents. NRC’s SRP for SNF transport packages and NRC ISGs that relate to criticality assessment under normal conditions of transport and accident conditions are discussed further below. NRC regulations and guidance documents as well as industry practice in performing criticality analysis all play a role in how an applicant may utilize moderator exclusion, possibly in conjunction with burnup credit methodologies or special package features, in a package safety assessment.

2.3.1 Standard Review Plan for SNF Transport Packages

In January 2000, NRC Staff completed a *Standard Review Plan for Transportation of Spent Nuclear Fuel*, NUREG-1617 (“Transport SRP”). The SRP provides NRC staff with guidance for the review and approval of applications for SNF transport packages under 10CFR71. The stated objectives of the Transport SRP are to *“(1) summarize 10 CFR Part 71 requirements for spent fuel transport package approval, (2) describe the procedures by which NRC staff determines that these requirements have been satisfied, and (3) document the practices used by the staff in reviews of package applications*.”

At the time the SRP was first issued, it was expected that it would be updated periodically. To address technical issues between the periodic updates of the SRP, the NRC issues ISG documents on an as-needed basis. To date, NRC SFPO has issued 20 ISGs related to SNF storage and transport issues. ISGs associated with the criticality assessment in SNF transport packages are summarized in Sections 2.3.2 through 2.3.4.

Section 6 of the Transport SRP contains the guidance for the criticality review of SNF transport packages. The Transport SRP describes the regulatory requirements contained in 10CFR71 that pertain to the criticality review. In addition to the regulatory requirements, the SRP also specifies acceptance criteria for criticality design. For example, the SRP includes guidance that in a criticality evaluation, the sum of the effective neutron multiplication factor (k_{eff}), two standard deviations (95% confidence), and the bias adjustment should not exceed 0.95 to demonstrate subcriticality by calculation. The guidance also states that a bias that reduces the calculated value of k_{eff} should not be applied.[3] This guidance, setting an acceptance criterion

[3] NRC, NUREG-1617, p. 6-4.

for k_{eff} equal to 0.95, is an example of an arbitrary conservatism that has been applied to criticality analysis. SNF criticality evaluations have also traditionally assumed that that the fuel is unirradiated, a "fresh fuel" assumption. The SRP recognizes that a methodology may be used that takes credit for burnup in the criticality evaluations if the methodology has been approved by NRC. The NRC has included more recent guidance in its ISGs regarding burnup credit methodology, as discussed in Section 2.3.2.

2.3.2 SFPO ISG-8, Rev 2, "Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks"

SFPO ISG-8, Revision 2, was issued by NRC in September 2002. SFPO ISG-8 provides guidance for the use of burnup credit for pressurized water reactor (PWR) SNF transport packages and provides more flexibility than was originally provided in the SRP. SFPO ISG-8 expanded the regulatory guidance in the Transport SRP to allow burnup credit based on actinide compositions for UO_2 PWR fuel with assembly average burnups up to 50,000 MWd/MTU, cooling times up to 40 years out of the reactor, and enrichments up to 5 weight percent U-235. The ISG also allows burnup credit for assemblies that contained burnable absorbers during irradiation and it references a source for selecting bounding axial burnup profiles.

As discussed in Section 1, SFPO ISG-8, in its present form, is a reasonable first step, but its application in cask design does not yield the cask efficiencies that would be achievable if full burnup credit were allowed. This is particularly important considering that the SNF transport package Certificate of Compliances (CoC) limit the *maximum* burnup of fuel assemblies for reasons related to shielding and thermal performance. If too high a minimum burnup for SNF is required to comply with the burnup credit analysis due to over-conservatism in the approach, it may not leave a large enough "window" below the maximum permitted burnup to permit loading a large population of spent fuel assemblies. In addition, SFPO ISG-8 recommendations associated with burnup measurement to validate fuel burnup could be resource intensive and could not be achieved for spent fuel that has already been loaded and sealed in SNF casks. The path to achieving the acceptance of full burnup credit remains an important objective for industry but is technically challenging as discussed in an earlier EPRI report, "Fission Product Benchmarking for Burnup Credit Applications," completed in 2002.[4]

2.3.3 SFPO ISG-11, Rev 3, "Cladding Considerations for the Transportation and Storage of Spent Fuel"

In November 2003, NRC staff issued SFPO ISG-11, Revision 3. SFPO ISG-11 notes that NRC staff is *"currently reevaluating the technical basis for the transportation of spent fuel including assemblies with average assembly burnups exceeding 45 GWd/MTU. The staff is reviewing data and technical reports to further understand the mechanical and fracture toughness properties of spent fuel cladding in relation to the transportation of high burnup fuel under 10 CFR 71.55. Therefore, until further guidance is developed, the transportation of high burnup commercial spent fuel will be handled on a case-by-case basis using the criteria given in 10 CFR 71.55, 10 CFR 71.43(f) and 10 CFR 71.51."* NRC Staff's concerns are related to the potential for degradation of high-burnup spent fuel cladding and the subsequent reconfiguration of SNF geometries under the hypothetical accident conditions.

[4] *Fission Product Benchmarking for Burnup Credit Applications*. Progress Report, EPRI, Interim Report, December 2002. EPRI 1002879.

SFPO ISG-11 goes on to state that under hypothetical accident conditions, *"the licensee must assure that there is no significant cladding failure. This is in accordance with the criticality requirements of 10 CFR 71.55 and by the shielding and containment requirements of 10 CFR 71.51."* It should be noted that the statement in SFPO ISG-11 that *"this is in accordance with the requirements of 10 CFR 71.55"* is not consistent with the regulations and is in conflict with NRC Staff guidance in SFPO ISG-19 (discussed below) that requires licensees to perform its criticality assessment assuming a reconfigured fuel geometry. If there is "no significant cladding failure" under the hypothetical accident conditions, then there cannot credibly be a reconfigured fuel geometry for high burnup fuel. In addition, the margin to subcriticality is highly likely to be greater with a reconfigured geometry.

2.3.4 SFPO ISG-19, "Moderator Exclusion Under Hypothetical Accident Conditions and Demonstrating Subcriticality of Spent Fuel Under the Requirements of 10 CFR 71.55(e)"

The NRC issued SFPO ISG-19 in May 2003. The ISG addresses the criticality safety of SNF under the hypothetical accident conditions specified in 10 CFR 71.73. Particular focus in this SFPO ISG is given to the criticality analysis associated with transport of SNF with burnups greater than 45,000 MWD/MTU, which may have brittle cladding that could fracture under the impact loads associated with the hypothetical accident free drop test conditions. SFPO ISG-19 states that the "criticality safety of the reconfigured fuel assembly must be demonstrated."[5]

SFPO ISG-19 outlines two basic approaches that an applicant can use to address the requirements of §71.55(e) that a single damaged package remains subcritical: *"(1) showing that reconfigured fuel is subcritical even with water inleakage, and (2) showing that the package excludes water under hypothetical accident conditions*." Table 2-1 summarizes the approaches in SFPO ISG-19 for demonstrating subcriticality of spent fuel.[6]

As shown in Table 2-1, for applications that show that reconfigured fuel is subcritical assuming water inleakage into the package under the hypothetical accident conditions, the criticality assessment approach recommended in SFPO ISG-19 is to provide "bounding or credible reconfigured fuel geometries". Such an assessment would involve one of two approaches to the criticality analysis: postulating bounding post-accident fuel reconfigurations for criticality; or postulating credible bounding configurations on the basis of structural and material behaviors, and limited reliance on material properties of high burnup fuel cladding and failure criteria. However, the SFPO ISG-19 guidance states that at this time, "there is insufficient material property information for high burnup fuel to allow" a structural evaluation to determine reconfigured fuel geometries based on the material properties of the SNF cladding. Therefore, the "credible reconfiguration" method for the moderator intrusion approach would entail providing the NRC with a significant amount of research data on the structural characteristics of high burnup irradiated fuel cladding and the probabilities of fuel redistribution due to an accident in order to be successful in licensing.

[5] NRC, SFPO ISG-19, "Moderator Exclusion under Hypothetical Accident Conditions and Demonstrating Subcriticality of Spent Fuel under the Requirements of 10 CFR 71.55(e)", p. 1.

[6] NRC, SFPO ISG-19, Table 1, p. 4.

For applications that base approval on an assumption that there will be no water inleakage (referred to as "moderator exclusion" in SFPO ISG-19) under the hypothetical accident conditions, NRC guidance recommends that physical testing of the water exclusion boundary be performed. For transport packages that include welded canisters, the guidance recommends that a canister drop test be performed as part of the impact limiter testing. This would include a scale model of the canister and contents in the 30-foot drop test, leak testing before and after each drop, and demonstration of an acceptable leak rate after the drop to provide assurance of moderator exclusion under the hypothetical accident conditions. It is worthy of note that the same rational that currently permits the use of modern-day analytical tools to be adequate to substantiate a typical transportation cask containment boundary without testing seems equally as applicable to qualifying the canister shell as a water exclusion boundary.

For applications to transport packages with welded canisters and for direct-loaded casks (bolted closures), the guidance recommends that the bolted closure system be tested as part of the impact limiter testing. This would include a scale model of the cask bolted closure system as part of the scale model used to perform the impact limiter testing. Again, the guidance recommends that leak testing be performed before and after each drop, and that an acceptable leak rate be demonstrated to provide assurance of moderator exclusion under the hypothetical accident conditions.

Prior to issuance of SFPO ISG-19, analysis of the fuel condition under hypothetical accident conditions assumed that the fuel and fuel geometry remained unchanged and that there was no reconfiguration of SNF geometry under accident conditions. Therefore, the criticality analysis performed to demonstrate compliance with 10 CFR 71.55(b) would apply and constitute a bounding case for accident conditions governed by 10 CFR 71.55(e). Consideration of potential reconfiguration is a direct result of spent fuel burnups in excess of 45 GWd/MTU, as discussed in SFPO ISG-11.

Table 2-1
Summary of Approaches for Demonstrating Subcriticality of Spent Fuel Under the Requirements of 10 CFR 71.55(e)[7]

(1) Approvals Based on Reconfigured Fuel		
Approach	**Characteristics**	**Objective**
Criticality Assessment of Bounding or Credible Reconfigured Fuel Geometries Based on Criticality Assuming Water Inleakage	1. Postulate bounding fuel configurations for criticality. 2. Evaluate criticality and credibility of bounding configurations based on basic structural and material behavior. 3. Reduced reliance on material properties of high burnup fuel cladding and failure criteria. 4. Criticality analyses of reconfigured fuel from criticality bounding configurations.	With water inleakage, demonstrate subcriticality of defined set of credible or bounding fuel configurations based on criticality.
Criticality Assessment of Reconfigured Fuel Geometries Based on Actual Structural and Material Behavior Assuming Water Inleakage	1. Need material properties of high burnup fuel cladding and failure criteria. 2. Requires nonlinear finite element analysis of fuel assemblies and fuel rods under drop impact conditions. 3. Failure modes and fuel rod failure distributions to be addressed (probabilistic approach to the distribution of material properties among fuel rods). 4. Develop credible fuel reconfiguration geometries. 5. Criticality analyses of reconfigured fuel structural analysis results.	With water inleakage, demonstrate subcriticality of credible fuel configurations based on actual structural and material behavior. This requires extensive data for irradiated hydrided cladding material properties for high burnup fuels. These data are currently not available. Therefore it is judged that this approach is currently not practical.

[7] Ibid.

Table 2-1
Summary of Approaches for Demonstrating Subcriticality of Spent Fuel Under the Requirements of 10 CFR 71.55(e) (Continued)

(2) Approval based on moderator exclusion		
Approach	**Characteristics**	**Objective**
Criticality Assessment of Reconfigured Fuel Assuming Moderator Exclusion	1. Demonstrate water-tight barrier under hypothetical accident conditions. 2. Perform drop test of cask (i) OR inner canister (ii) as described below.	
(i) For Welded Canister-Based Systems: Canister Drop Test as Part of Impact Limiter Testing	1. Include scale model of canister and contents in transport cask impact limiter 30-foot drop tests. 2. Perform relative leak rate testing by testing before and after each drop. 3. Demonstrate leakage rate acceptable to prevent water inleakage.	Physical test of scaled canister to provide added assurance of moderator exclusion under accident conditions.
(ii) For Canister-Based Cask Systems and Direct-Loaded Casks: Bolt Closure Systems Test as Part of Impact Limiter Testing	1. Included transport cask bolt closure systems in scale model of cask in 30-ft drop tests of the impact limiter. 2. Perform relative leak rate testing by testing before and after each drop. 3. Demonstrate leakage rate acceptable to prevent water inleakage.	Physical test of scaled bolt closure system to provide added assurance of moderator exclusion under accident conditions.

3
APPROACHES TO MODERATOR EXCLUSION

3.1 Introduction

It should be recognized that moderator exclusion, if approved by the NRC, might not be universally applicable to all spent fuel transport cask designs. Its use, alone or in combination with some other methodology (such as burnup credit), may be acceptable under one set of circumstances, but not under a different set. This section discusses the different SNF storage/transport scenarios and offers some observations on how moderator exclusion might be applicable.

3.2 Transportation of Previously Loaded Storage-Only Casks

The dry-storage-only cask was the first generation of the out-of-pool SNF storage systems. Storage-only systems were intended to provide on-site dry storage without specific provisions for off-site transportation. The first storage-only systems deployed were metal storage-only casks containing uncanistered or "bare" fuel assemblies at the Surry Nuclear Power Station. Dry storage systems that involved canistered spent fuel, typified by the NUHOMS and VSC-24 systems, provide on-site dry storage in concrete above-ground structures. Due to the unique design features of the metal storage-only casks compared to the canister-based systems, each will require different solutions for possible transport off-site without the need to repackage the SNF.

For both the bare-fuel metal cask systems and the welded canister systems, it was originally assumed that the fuel would be returned to the storage pool, and subsequently would be unloaded and repackaged into a transport-only cask. For the welded canister system, an alternative considered was that the loaded canister would be transferred into a transportation overpack and moved under some "special arrangement," as discussed in Section 2.2. Metal casks could also employ the special arrangements allowed by regulation, but this might be more difficult in that these systems do not have a seal-welded, dual-closure systems. It was believed that storage-only canisters could be transported off-site through a set of arrangements that met the NRC's "equivalent safety" criterion. This philosophy is still valid and moderator exclusion can play a role in the demonstration of equivalent safety. It should be noted that storage-only canisters were certified only under 10 CFR 72 for storage; thus, the canisters and their companion transportation overpacks will have to undergo NRC certification under Part 71. This process involves more than a simple moderator exclusion demonstration. It must run the gamut of structural, thermal, shielding, and criticality safety analyses culminating in a Safety Analysis Report – Packaging (SARP).

Some of the challenges facing the potential to transport storage-only cask systems without the need to unload the packages are detailed in Section 5 of this report. The possible role of moderator exclusion in the one-time transport of storage-only casks systems is also discussed.

3.3 Transportation of Today's Dual-Purpose Casks

Contemporary dual-purpose casks are certified for both on-site storage and off-site transportation. One type of dual-purpose cask system is similar in basic design to those store-only systems in Section 3.2. It consists of seal-welded canisters that contain SNF, with concrete overpacks for on-site storage and a metal transport cask for off-site shipment of the loaded canister. One design, the Hi-Star 100 System, has certified the metal overpack for both storage and transport of SNF in the inner canister. In addition to canister-based dual-purpose casks, there are also cask designs in which spent fuel is loaded directly into a metal cask, without the use of a sealed metal canister.

All of the dual-purpose systems being marketed today have at least one design that has received Part 72 certification for storage and Part 71 certification for transportation. It should be noted that some of the dual-purpose certifications have received amendments to the Part 72 certification for more advanced designs that can store a wider range of fuel, but not all of the Part 71 certificates have been amended to match the storage contents. Some of these amended systems are in the NRC Part 71 review process. Further, some of the systems have been loaded for storage at nuclear power plant sites in anticipation of receiving Part 71 certification for the advanced designs.

Aside from being to be transportable from inception, dual-purpose systems have a number of advantages over the storage-only systems that were loaded in the 1990s. First, all of the dual-purpose systems that are certified have benefited from the experience gained and the lessons learned associated with the loading processes and characterization of fuel designated for dry storage. In addition, QA procedures and records for both the fuel and the storage/transport system hardware have improved markedly compared with those of a decade or more ago. Additionally, enhancements in system fabrication methods and inspection techniques have also been incorporated.

The dual-purpose systems also differ from the storage-only systems from the perspective of the fuel that is being or will be stored. Since the first generation of dry storage systems was deployed, there has been an increase in both fuel initial enrichment and burnup, with enrichments of up to 5 w/o and average fuel discharge burnups of 55-60 GWT/MTU not being uncommon. Nuclear operating companies require that dry storage systems be capable of accommodating fuel with a wide range of enrichments, burnups and cooling times, presenting challenges to cask designers in the area of thermal designs for storage and in criticality safety design for transport. If options such as burnup credit and moderator exclusion cannot be employed as part of the criticality safety analysis for transport of SNF, the result will be a decrease in cask capacities for the transport of SNF. Another result would be that current cask designs that are capable of accommodating most fuel for storage would be limited in the fuel that can be accommodated for shipment off-site. Without burnup credit and/or moderator exclusion, some high capacity systems that were designed as dual-purpose would, in fact, only be licensed for storage, requiring that SNF be unloaded and transferred to a lower-capacity transport cask for shipment offsite.

In fact, there are a few examples of such systems that have already been loaded and are in storage. Section 6 discusses the possible application of moderator exclusion to dual-purpose cask systems.

3.4 Transportation of Next-Generation Casks

The history of the SNF dry storage system designs suggests that the next generation of packages will not substantially differ from the systems of today. The evolution will be in materials and methodologies used for cask design and analysis. An example of the latter is the inclusion of convective cooling internal to the canister during storage to permit shorter cooling times, higher burnup and/or greater fuel assembly capacity. Improved analytical tools for structural, thermal and nuclear processes will contribute to making the systems more efficient. Further improvements in fabrication techniques and economies of scale may reduce the unit cost of storage system hardware as more fuel is placed in storage. All of these improvements are important but are at the margins of the system designs. Vendors are not starting advanced designs with a blank page, but rather are evolving from the current cask designs to the next generation designs while applying lessons learned during the licensing, fabrication and operation of existing designs. Of course, the goal for the next generation of cask designs will be increased fuel assembly capacity with some reasonable range of fuel burnup, enrichment, and cooling time. Of all the cask parameters, capacity is the biggest competitive attribute since it affects cost, operations, and risks, consistent with facility physical limitations.

The availability of moderator exclusion as part of an integral criticality safety methodology would make a significant, positive contribution to the design of the next generation systems. From a risk-informed perspective, burnup credit can be taken into consideration, but the industry need not solely rely on burnup credit to demonstrate criticality safety if moderator exclusion can be reasonably implemented. With moderator exclusion as the licensing basis, a criticality analysis of the flooded condition with full burnup credit may be used to demonstrate subcriticality as a defense-in-depth feature of the package design during licensing. Section 7 discusses the possible application of moderator exclusion to next-generation cask systems.

3.5 Bare Fuel Transport-Only Cask

The initial increments of SNF that will be transported to a geologic repository for permanent disposal will likely come from inventories in reactor pools, not on-site dry storage. Only by moving spent fuel directly from the SNF storage pools will nuclear operating companies be able to avoid the cost of loading additional dry storage systems. This means that early in the SNF disposal process, there will be requirements for “bare” (i.e., uncanistered) fuel transportation casks. At this time, there are only two contemporary dual-purpose rail transport casks certified in the U.S., the TN-68 (for BWR fuel) and the NAC-STC (for BWR or PWR fuel), that accommodate bare fuel for both storage and transport. In addition, the NAC-LWT truck cask (for BWR or PWR fuel) and the GA-4 truck cask (for PWR fuel) are also designs that are available to transport bare fuel. These system designs are currently approved for transportation of a limited range of fuel parameters and limited capacity; thus, the designs have not required either moderator exclusion or burnup credit.

Future bare fuel transport casks will need to accommodate higher enrichments and burnups and shorter cooling times. Because BWR spent fuel pools do not contain soluble boron, the spent fuel pool loading condition with unborated water is the limiting case for criticality analysis and moderator exclusion is of no benefit in licensing for the normal condition but may have greater application to higher exposure fuel under accident conditions. Ultimately, burnup credit for BWR fuel will be required to raise the enrichment limit for transportation in high capacity casks above about 4.2 w/o with a minimum burnup of 45 GWd/MTU. For PWR fuel, casks are loaded in borated-water spent fuel pools where the soluble boron can be credited in the criticality analysis for loading operations which is not applicable to the transportation criticality analysis since fresh water moderation must be assumed. Thus, moderator exclusion and/or burnup credit will be required for efficient, high-capacity bare fuel transport cask designs.

It is likely that future bare fuel cask designs will be derived from current generation transportation overpacks for canister-based systems or from storage-only metal casks. The designers of conventional bare fuel transport-only casks will face a greater challenge if seeking an exception in that §71.55(c) invokes the single error assumption that may be difficult to meet for a conventional bare fuel cask that employs a single bolted closure lid. This could be resolved if bare fuel casks are derived from single or dual-purpose metal cask designs that include redundant sealing of the containment system for transport, since Part 72 requires "redundant sealing of the confinement systems" [§72.236(e)]. . How the single error assumption will be applied to future configurations is unknown at this time. It does appear that NRC regards a QA-qualified *welded* canister to be less prone to leakage than a bolted closure since under 10 CFR 72, bolted storage systems require leakage monitoring between redundant seals whereas welded systems have no such requirement. This suggests that to be eligible for moderator exclusion, separate redundancy and pre-shipment leak testing capability may be required for bolted closure systems. Alternatively, one can envision canistering bare fuel prior to off-site transportation although the expense of this option would motivate seeking a bare-fuel cask design solution. Section 7 discusses the possible application of moderator exclusion to bare fuel transport casks.

4

MODERATOR EXCLUSION THROUGH EXISTING REGULATIONS OR THROUGH RULEMAKING

At present, the NRC's position is not to generically certify entire storage and transport cask designs based on an exception to the moderator intrusion requirements of §71.55(b) pursuant to §71.55(c). Given this fact, there are two alternatives that can be considered to employ moderator exclusion. Preferably, moderator exclusion may be possible within the current regulatory framework of §71.55 through an interpretation of these regulations and the development of supplementary guidance documents to allow the exclusion of moderator under certain conditions such as recognition of package design features that make water intrusion not "credible" under both normal and accident conditions (allowing water inleakage into the containment system but not the fuel region may be one aspect of this alternative). Alternatively, NRC may determine that rulemaking to modify §71.55(c) to remove the "exception" wording is necessary. This section describes the rulemaking process, including the process for a request for rulemaking, and the analysis needed to support such a request. It also describes the benefits and risks associated with proceeding with rulemaking and describes different alternatives regarding the types of regulatory changes that might be considered.

4.1 Moderator Exclusion Through Existing Regulations

Under the current regulations, §71.55 contains what appears to be some contradictions. The requirements of §71.55(b) which apply to the pristine package, stipulate that water inleakage to the containment is a mandatory (non-mechanistically driven) assumption. However, the language of §71.55 (b) also requires consideration of moderation by water to the most reactive "credible" extent. On the other hand, §71.55(d) and §71.55(e) which apply to normal and hypothetical accident conditions respectively have different requirements. §71.55(d) prohibits inleakage of water and §71.55(e) requires the consideration of inleakage to the extent "credible" consistent with the damage to the package due to design basis normal and hypothetical accident conditions as currently defined in §71.71 and §71.73.

Clearly §71.55(b), as currently interpreted, becomes the most restrictive of the three §71.55 requirements for criticality evaluation purposes. However, §71.55(c) provides a means for obtaining an exception from the flooded containment system requirement of §71.55(b), and this exception together with proof of the "incredibility" of water intrusion under hypothetical accident conditions could be used to justify the use of moderator exclusion. The assumption that a cask is "leaktight" under §71.55(d) for normal conditions is always a cask design basis so compliance is not an issue.

Another alternative regarding the requirements of §71.55(b) that has been proposed to the NRC, that would not require the use of the "exception" allowed by §71.55(c), is to interpret the regulations and supplement the guidance documents to allow leakage into the "containment system" but not into the fuel region (if this can be demonstrated as not "credible"). This seemingly only has application to canistered systems.

The benefits of an approach that would allow moderator exclusion within existing regulations include near-term implementation, avoiding the time needed for rulemaking and more efficient use of NRC and industry resources to address other storage and transport issues.

4.2 Rulemaking Process

The process of developing or modifying NRC regulations as defined in the applicable CFR sections is called rulemaking. NRC may issue a Notice of Proposed Rulemaking to make changes to existing regulations or to develop new regulations. Alternatively, any member of the public may submit a petition to the NRC to develop, change, or rescind one of its regulations. NRC regulations contained in 10 CFR 2, Subpart H provide the requirements for rulemaking including the submittal of a petition for rulemaking (§2.802).

Petitions filed under §2.802 must:

1. *"Set forth a general solution to the problem or the substance or text of any proposed regulation or amendment, or specify the regulation which is to be revoked or amended;*

2. *State clearly and concisely the petitioner's grounds for and interest in the action requested;*

3. *Include a statement in support of the petition which shall set forth the specific issues involved, the petitioner's views or arguments with respect to those issues, relevant technical, scientific or other data involved which is reasonably available to the petitioner, and such other pertinent information as the petitioner deems necessary to support the action sought. In support of its petition, petitioner should note any specific cases of which petitioner is aware where the current rule is unduly burdensome, deficient, or needs to be strengthened."*

If the NRC determines that the petition includes all of the information required by §2.802, a docket number will be assigned to the rulemaking petition and the docketed petition will be made available for public comment. Public comment can be requested through a notice of docketing of the petition in the Federal Register or upon publication in the Federal Register of a proposed rule developed in response to the petition.

4.3 Contents of a Petition for Rulemaking

The petition for rulemaking would contain background information and analysis to support the proposed changes. As outlined in §2.802, the petitioner would include a "statement of petitioner's interest," outlining the reason petitioner has an interest in the changes to the regulations and providing evidence that the petitioner is an "interested party" as defined in §2.802. The Nuclear Energy Institute (NEI) would likely file such a petition on behalf of its members in such cases based on input from industry.

The petition would provide the grounds for the proposed action, providing a summary of the reasons for making changes to the existing regulations. This would include an overview of the benefits of the proposed change (such as lower SNF transport risk due to a decrease in the number of casks being loaded and transported).

The petition would provide background information related to criticality safety for SNF transport. This analysis would include a discussion of current regulations and regulatory guidance, including the chronology of changes to NRC ISGs related to transport criticality safety assessments and international transport safety regulations. It would also provide a summary of the conservative assumptions that have been made historically in criticality safety analyses (fresh fuel assumption, k_{eff}<0.95, etc.) as summarized in Section 1.3. The background information would also include an overview of related NRC analysis and discussions regarding criticality safety issues for SNF transport.

The petition would include a statement in support of the petition setting forth the proposed changes to the regulatory language in §71.55 as well as proposed changes to regulatory guidance documents such as the Transport SRP and ISG documents. The petition would also identify whether any additional NRC regulations are affected by the proposed changes.

Petitioners must also provide relevant technical, scientific or other data or information that is needed to support the proposed action. For example, the petitioner might provide a risk assessment that identifies the transport risk associated with the proposed change compared to the transport risk if the change is not made. The transport risk of the proposed change would include the risk of misloaded fuel assemblies, cask handling risks, transport accident risks, and risks associated with normal conditions of transport (worker and public dose). This would be compared to the same types of risks assuming that the proposed change to the regulations was not made.

In addition to the technical information provided with the petition for rulemaking, a petitioner would also provide NRC with supplementary analysis to support NRC's review of the petition. This would include analysis of whether the petition has any environmental effects under the National Environmental Policy Act (NEPA) of 1969; whether the petition contains any new or amended information or requirements that would be subject to the 1980 Paperwork Reduction Act; whether the petition would result in new requirements being imposed on a licensee such that the NRC must perform a backfit evaluation; and whether the petition would have an economic impact on small businesses as defined in the 1980 Regulatory Flexibility Act.

4.4 Benefits of Rulemaking to Change §71.55

There are a number of benefits associated with making changes to §71.55 to allow moderator exclusion for spent-fuel transportation packages. A properly modified §71.55, which clearly states under what conditions moderator may be excluded from the criticality safety analysis, would alleviate the current ambiguity and provide greater predictability to licensees when designing transport packages to meet industry needs. The ability to license higher capacity casks also reduces the risk of transportation events. Modifying §71.55 also removes the need for the NRC to issue regulatory exceptions as currently required in the application of §71.55(c).

With moderator exclusion, transport package designs can be simplified. Flux traps and neutron poisons can be reduced or eliminated from package interior basket configurations allowing increased packages capacities. Increased package capacities will result in the need to load fewer SNF transport packages, thus reducing the risk of accidents during handling operations and transport, reducing worker exposure during handling and transport, and reducing the public exposure due to routine transport. As identified in DOE's Final Environmental Impact Statement (FEIS) for the Yucca Mountain Repository,[8] the greatest risk to the public and workers associated with the transport of SNF is the radiation dose associated with routine transport. If the number of shipments can be reduced, the risk of transporting SNF is also reduced. In addition to lowering the radiological risk associated with transporting SNF, having fewer shipments of SNF will also result in a reduction in the number of traffic accidents associated with the transport, reducing the non-radiological risks as well.

As discussed in SFPO ISG-11, Revision 3, NRC staff is currently reviewing *"data and technical reports to further understand the mechanical and fracture toughness properties of spent fuel cladding in relation to the transportation of high burnup fuel under 10 CFR 71.55."*[9] A modification to the regulations that would allow licensees to show that, under the hypothetical accident conditions, the intrusion of water is not credible in the transport package cavity, would reduce the need for NRC staff and industry to spend additional resources to assess the integrity of high-burnup fuel under highly unlikely, hypothetical accident conditions. If there is no moderator present, no matter the configuration of the fuel, the package and its contents would remain subcritical. In the absence of moderator, the structural integrity of the fuel cladding becomes one of satisfactorily showing that the fuel can be safely retrieved rather than justifying alternative configurations for criticality safety.

In a similar way, if it can be shown that moderator intrusion is not credible in a SNF transport package that meets certain predetermined requirements, it may reduce the need to rely solely on burnup credit for package approval. In doing so it may reduce the need for the NRC and industry to expend significant resources to obtain additional experimental data to support approval of methodologies for fission product burnup credit.[10] The combination of existing burnup credit research (providing the margin) and moderator exclusion methodologies are likely to provide sufficient assurance for criticality safety of SNF packages under normal and hypothetical accident conditions of transport. Risk assessment techniques are likely to play a role in these assessments.

4.5 Risks Associated with Rulemaking to Make Changes to §71.55

While there are benefits associated with making changes to the regulations contained in §71.55, there are also risks associated with rulemaking. Potential risks include those associated with the NEPA analysis for transporting SNF and the possible imposition of additional requirements in the regulations for SNF package testing or analyses.

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[8] DOE, Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High Level Radioactive Waste at Yucca Mountain, Nye County, Nevada, DOE/EIS-0250F, February 2002, Table 6-1, p.6-15.

[9] SFPO ISG-11, p. 1.

[10] Parks, C., J. C. Wagner, I. C. Gauld, and B. L. Broadhead, Oak Ridge National Laboratory, DRAFT, Working Discussion Paper: Experimental Needs to Implement Full Burnup Credit in Transport of Spent Nuclear Fuel, November 5, 2003.

4.5.1 NEPA Evaluation

Since the rulemaking process is an open, public process, NRC must seek public comment on any changes to its regulations. As part of the rulemaking process, NRC must determine whether the proposed changes affect the NEPA analysis associated with the transport of SNF. In December 1977, the NRC issued a "*Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes*", NUREG-0170. The environmental statement was completed to assess the impacts associated with the transport of radioactive materials in the U.S. in order to ensure that NRC regulations continued to meet the goal of limiting radiological impacts to a level that is as low as reasonably achievable. In this study, the NRC evaluated the impact of regulated radioactive materials transport activities on public health and safety in compliance with NEPA. The study concluded that the risk to the public and transport workers was small. Since NUREG-0170 was released in 1977, NRC staff and contractors have performed additional risk assessments regarding SNF transport that have found that the risks assessed are even lower than those identified in NUREG-0170 due to conservatisms in the original assessment and due to better analytical capabilities available today.[11, 12]

It should be noted that the transportation risk assessments noted above, as well as the transport risk assessment performed as part of the FEIS for the proposed Yucca Mountain repository, assumed that casks were moderated with water under accident conditions to the most reactive, credible extent, consistent with current regulations. Since it was assumed that all packages used to transport SNF could not result in a critical configuration under hypothetical accident conditions, there was no need for the risk assessments to even consider criticality risk. Risk assessments examined worker and public radiological risk for routine transport and under accident conditions. The accident conditions evaluated included impact resulting in the dispersal of some of the radioactive contents and loss-of-shielding accidents, but did not assess accident risks involving the potential for criticality. It should be noted that EPRI Technical Report 1002879[13] included a risk perspective related to burnup credit that estimated that consequences of a criticality accident during SNF transport as being "inconsequential from a risk standpoint", with probabilities in the range of 10^{-13} to 10^{-17} per year. Thus, while criticality accidents during transport have not been specifically included in the transport risk assessments that have been performed, the risks associated with a criticality accident are extremely small and would not result in any significant increase in transport risks.

While the assessment of the risk of criticality accidents during transport can be shown to be essentially zero, a change to NRC regulations to allow moderator exclusion in criticality assessments for the hypothetical accident conditions may result in closer public attention to SNF transport risk. While the change to the regulations can be shown to not increase the risk of transport, it could appear that allowing moderator exclusion somehow increases the risk of a criticality accident since these previously were not even assessed. A risk assessment would look at the probability and the consequences of such an accident, but the public might focus on only the consequence.

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[11] NRC, Shipping Container Response to Severe Highway and Railway Accident Conditions", NUREG/CR-4829.

[12] NRC, Reexamination of Spent Fuel Shipment Risk Estimates, NUREG/CR-6672.

[13] *Fission Product Benchmarking for Burnup Credit Applications*. Progress Report, EPRI, Interim Report, December 2002. 1002879.

Rulemaking to change §71.55 to allow moderator exclusion for SNF packages might also result in re-opening the transportation risk assessment for Yucca Mountain or requiring supplemental analyses since criticality risk was not considered in the Yucca Mountain EIS. It should be noted that the State of Nevada did raise the issue of criticality risk associated with potential sabotage scenarios for SNF transport, so it is unlikely that this action would occur without comment from the State of Nevada.[14] Again, while the risks of a criticality accident are essentially zero, there could be a perception that a change to §71.55 would result in increasing the risks associated with transporting SNF simply because criticality risk must now be assessed. Potentially off-setting this is that greater cask capacity as a result of moderator exclusion results in fewer shipments and hence lower accident risk.

4.5.2 Other Considerations

The proposed change to §71.55 should be written such that the new language does not preclude the exceptions allowed under current regulations. This is particularly important for possible moderator exclusion applications for the transport of storage-only casks. Potential applicants should not be precluded from utilizing the current regulations that allow NRC to grant an exception to the regulatory requirement of §71.55(b) requiring moderation under normal conditions if the applicant can show that it meets the requirements of §71.55(c). In addition, any change to the regulations should be written as a new section that applies to SNF packages allowing the existing regulations to stand for other fissile material packages.

The rulemaking will need to address two items pertaining to moderator intrusion. Specifically:

- 10 CFR 71.55(b) requires moderator intrusion into the containment system under normal conditions to occur non-mechanistically for all fissile material packages. No provision exists for the cask designer to prove that a spent fuel cask design will prevent moderator intrusion under normal conditions. Only the provision of §71.55(c) allows for moderator exclusion *on an exception basis.* This normal condition regulation needs to be modified to recognize packages that, by design, have the features necessary to make moderator intrusion under normal conditions non-credible.
- 10 CFR 71.55(e) requires moderator intrusion to be assumed during accident conditions unless, per SFPO ISG-19, the designer can prove by analysis and testing of the design that it is not credible under all accident conditions and does not need to be assumed in the criticality analysis. This part of the rule should be clarified.

Paradoxically, the current rules allow moderator exclusion to be demonstrated by design, analysis, and testing for the conditions where it otherwise *is* a credible event (the hypothetical accident condition), but does not allow the designer to prove moderator exclusion for the condition where it is not a credible event (the normal condition).

It is possible that a change to the regulations to allow moderator exclusion would result in the imposition of additional physical tests to demonstrate sufficiently low leakage into the containment boundary to claim moderator is excluded under credible or bounding accident

[14] State of Nevada, "Petition for Review of DOE's Final Environmental Impact Statement for the Proposed Yucca Mountain High-Level Nuclear Waste Repository" in the U.S. Court of Appeals for the District of Columbia, Case No. 02-1179, June 6, 2002.

conditions. NRC SFPO ISG-19, Table 2, discusses applicants performing physical drop tests of the transport cask and/or inner canister to demonstrate the water-tight barrier (either the cask bolt closure or the welded inner canister) for accident conditions. The same rational that currently permits the use of modern-day analytical tools and the acceptance criteria provided by such industry standards as the ASME B&PV Code to qualify a typical transportation cask containment boundary without testing may be equally as applicable to qualifying the canister shell as a second transportation containment boundary or a water exclusion boundary without testing.

For normal conditions of transport, in addition to appropriate structural analyses, the guidance of SFPO ISG-18[15] should be considered as the basis for moderator exclusion without additional testing. A "wet" criticality analysis would still need to be performed, as it currently is, for the normal cask loading condition in the spent fuel pool where soluble boron can be credited (for PWR fuel) with or without burnup credit. BWR fuel with higher enrichments may require burnup credit to be employed for the normal cask loading condition.

In addition to drop tests, SFPO ISG-19 also includes a leak rate test. It is possible that a change to the regulations would result in the imposition of additional leak testing of the containment boundary (or boundaries) prior to shipment. While such testing is typically required by the CoC for a bolted bare fuel transport cask containment boundary during fabrication and on the containment seal prior to shipment, for a welded canister the testing could only be performed in the fabrication shop on the containment boundary shop welds and at the time of canister fuel loading on the field closure welds. SFPO ISG-18 and the drop testing required by SFPO ISG-19 may provide the justification for no additional canister leak testing prior to shipment.

In at least one current design of a canister-based system, the bolted overpack is considered the containment boundary. This was necessary to meet the dual containment requirement of 10 CFR 71.63(b) for transporting fuel debris (i.e., essentially loose plutonium). This requirement has since been rescinded by rulemaking. Therefore, designers of canister-based transport systems might consider defining the containment boundary as the canister rather than the bolted overpack. This would allow credit to be taken for the leaktight canister and also make the transport containment boundary the same as the storage confinement boundary for these systems. Additional analyses may be needed to implement such an approach in order to show that the internal canister complies with the external pressure requirement in 10 CFR 71.61.

On the downside, it is possible that a change to the regulations to allow moderator exclusion would result in additional quality assurance and quality control requirements for the manufacture and use of package bolt closure systems. Current package safety analyses assume that packaging errors are related to loading/sealing errors, not fabrication errors. If NRC requires that fabrication errors also be considered, this could result in additional quality requirements during fabrication.

Making changes to the regulations contained in §71.55 is expected to be a lengthy and time-consuming process. If NRC does not proceeding with a rulemaking change of its own accord and a petition for rulemaking is needed, it could take several years for the new rules to come into effect, assuming NRC agrees to proceed with the rulemaking.

[15] SFPO ISG-18, "The Design/Qualification of Final Closure Welds on Austenitic Stainless Steel Canisters as Confinement Boundary for Spent Fuel Storage and Containment Boundary for Spent Fuel Transportation.

4.6 Conclusions

While there may be benefits to changing §71.55 through rulemaking to allow moderator exclusion for SNF packages, there are also a number of risks that must be assessed prior to moving forward with such a petition. In particular, the possible NEPA-related risks associated with including criticality risk in transportation risk assessments could open other issues that would require both NRC and DOE to reevaluate the risks of transporting SNF. Any changes to the regulations through rulemaking should not preclude applicants from using existing regulations and guidance in future amendment applications, but instead should be focused on allowing new and existing cask designs to take advantage of the added flexibility to address future SNF inventories. On the other hand, interpretation of existing regulations and associated guidance documents to allow the exclusion of moderator under certain conditions could provide a near-term solution and allow both industry and NRC to utilize resources to resolve other important storage and transport issues.

5
MODERATOR EXCLUSION APPROACHES FOR STORAGE-ONLY SYSTEMS

5.1 System Descriptions

The dry-storage-only systems were the first designs deployed at reactor sites to relieve pool storage capacity limits. Designs of this type were primarily intended to provide on-site dry storage without specific provisions for off-site transportation. These come in a variety of designs. The description and population of deployed storage-only systems is shown in Table 5-1.

Table 5-1
Deployed Storage-Only Systems at U.S. Nuclear Power Plant Sites (as of June 2005)

System	Description	No. Deployed*	Comments
CASTOR V/21	Non-canister, ductile cast iron metal cask	25	21 PWR
CASTOR X	Non-canister, ductile cast iron metal cask	1	33 PWR
NUHOMS-07P	Canister, horizontal concrete	8	7 PWR
MC-10	Non-canister, metal cask	1	24 PWR
NAC I28	Non-canister metal cask	2	28 PWR
NUHOMS 24-P	Canister, horizontal concrete	134	24 PWR
NUHOMS 52-B	Canister, horizontal concrete	27	52 BWR
TN-32	Non-canister, metal cask	57	32 PWR
TN-40	Non-canister, metal cask	19	40 PWR
VSC-24	Canister, vertical concrete	58	24 PWR

*Number of deployed systems is approximate as of November 2005.

5.2 Storage-Only System Design Types

One storage-only system design type that has been loaded at several nuclear power plant sites is the welded canister system using a horizontal or vertical concrete structure for above ground shielding and heat management. A transfer cask moves canisters between the reactor pool and the storage module. For off-site transportation, it was originally thought that the fuel would be returned to the storage pool, decanistered, and loaded into a transport-only cask. Examples of the welded storage-only canister system are the NUHOMS-07P, -24P, and -52B, and the VSC-24. Note that the NUHOMS-07P canister was designed for transport in the IF-300 cask but has not been submitted to the NRC for transport certification.

Another of the storage-only system design types that have been deployed at nuclear power plant sites is the metal storage-only cask that has the appearance of a transportation cask, but that was originally intended only for storage. It was originally thought that the casks would be returned to the storage pool, unloaded, and the SNF reloaded into a transport-only cask. Examples of the metal storage-only cask are the TN-32 and -40, CASTOR V/21 and X/33, Westinghouse MC-10, and NAC I28.

A third storage-only design type was proposed, but only one installation was constructed. This is a concrete storage vault where fuel assemblies are stored in individual chambers within a shielded building. Remote operations are used to move assemblies. Off-site shipping is via a transport-only cask intended for bare fuel. Since there is only one such facility storing commercial spent fuel and it is now owned and operated by DOE, no further consideration will be given to this storage system type in this report.

5.3 Fuel Characteristics

In general, when nuclear operating companies deployed the early storage-only systems, the oldest, intact (not damaged) SNF was loaded into dry storage. At that time, the nuclear operating company's primary motivation was to free pool rack space; long-term fuel management strategy was not the driving consideration. The fuel being loaded into dry storage had substantially lower initial enrichments and burnups than today's fuel. In general a typical PWR enrichment was in the range of 3.0% to 4.0% with an average burnup of 35 to 40 GWd/MTU. A typical BWR enrichment was in the range of 2.5% to 3.0% with an average burnup of 30 to 35 GWd/MTU. The minimum cooling time for either assembly type was typically five years , but in most cases the actual fuel stored in these systems had cooling times at the time of loading in excess of 10 years.

5.4 Special Arrangement

Since neither storage-only system type has a ready solution for shipment of the loaded systems off-site, it is worth considering the "special arrangement" that might have been advocated at the time of system deployment. This alternative off-site shipping plan was considered for both canistered fuel and for metal storage-only casks. Loaded storage-only canisters would be transferred intact into transportation overpacks and moved under some "special arrangement" through exemptions to certain 10 CFR 71 requirements. Similarly, metal casks would be placed on transport vehicles and also moved under a "special arrangement." As described in Section 2.2.2, the term comes from IAEA regulations (i.e., TS-R-1, 824 - 826), but the philosophy is the same under NRC regulations.

As discussed below, several storage-only cask designers have approached the NRC to discuss plans to seek for NRC approval under 10 CFR 71 for transport of previously loaded storage-only systems. While the cask designers hope to qualify many of the loaded storage-only systems for transport off site within the existing 10 CFR 71 regulations without exemptions , the overall approach to qualify all of the canisters for transportation involves:

- Burnup credit with limited fission product credit,
- Deviations from certain regulatory guidance acceptance criteria such as k_{eff}<0.95 criterion,
- Limited use of the §71.55(c) moderator exclusion provision for normal transport conditions, and
- A small number of exemptions from the Part 71 regulations for canisters that cannot meet all of the 10 CFR 71 requirements using risk based methods, e.g., one-time shipment to a known location along a known transportation route under controlled conditions with quantifiable risks.

5.4.1 VSC-24 Transport

BNG FuelSolutions (BFS) has held several discussions with NRC staff regarding approving previously loaded storage-only canisters for transport in a Part 71 certified transport cask. BFS met with NRC staff to discuss BFS' preliminary evaluation of transporting the VSC-24 canister within BFS' FuelSolutions TS125 transport cask. Several meetings have taken place to discuss alternatives for ensuring criticality control during transport of the VSC-24 canister including seeking burnup credit and/or moderator exclusion for shipment of the 24-PWR assembly canisters.

During the first meeting in 2003, NRC recommended that BFS should limit the number of shipments that might require moderator exclusion. BFS' approach presented during the January 2004 meeting was developed to minimize the need for moderator exclusion. BFS' criticality evaluation will be based on the specific characteristics of the fuel assemblies loaded into each VSC-24 canister in order to more accurately predict the reactivity of the fuel within each canister. Using this type of analysis, BFS is expected to be able to qualify more of the canisters for shipment using a burnup credit approach than if it had performed a typical burnup credit calculation using generic fuel characteristics. Fuel parameters will be based on reactor records for core location, burnup, enrichment and discharge date. The preliminary burnup credit analysis presented by BFS considered three different approaches: actinide only burnup credit; actinides plus five major fission products; and actinides plus 15 major fission products.

In order to qualify the greatest number of VSC-24 canisters for shipment, BFS may need to deviate from current NRC staff guidance associated with criticality safety parameters, such as a k_{eff} limit <0.95 (BFS suggests 0.98 as a limit), credit for fission products, and a reliance of reactor records for burnup rather than confirmation of reactor records with a burnup measurement. In some of these cases, probabilistic risk analysis may be required to demonstrate that an accidental criticality event during transport resulting from possible misloads of under-burned fuel assemblies is not credible. Depending upon the approach used (burnup credit and criticality margin), different numbers of the 58 previously loaded VSC-24 canisters may be qualified for transport. Those that do not qualify for transport using the selected burnup credit approach would require an exemption from existing regulations assuming either moderator exclusion or other risk-based methods, or would have to be unloaded and reloaded into a qualified transport cask. Thus, moderator exclusion may play a role in the transport of loaded storage-only canisters, either through exemptions or exceptions in existing regulations or through some future change to the regulations. At the time of this report, BFS is preparing an amendment request for the transport of the VSC-24 canisters in the FuelSolutions TS125 transport cask and plans to submit the application to NRC in the near term. The initial submittal is not expected to address possible exemptions or exceptions associated with moderator exclusion.

5.4.2 TN-40 Transport

Transnuclear, Inc (TN) has also met with NRC to discuss TN's consideration of qualifying the TN-40 metal storage cask for limited transport. Repackaging the spent fuel into dual-purpose casks would result in increased risk associated with additional fuel moves, additional heavy load movements in the plant, additional heavy haul of casks, and increased worker radiation exposure. If the TN-40 casks can be qualified for limited transport, this would alleviate the need to transfer and repackage spent fuel.

TN presented an overview of the technical and licensing issues associated with transport of the TN-40. The TN-40 cask can meet the Part 71 requirements for transport if TN can assume burnup credit in its criticality analysis. TN's preliminary analysis showed that using actinide-only burnup credit, TN can probably qualify most of the TN-40 casks for transport. TN may need to take credit for actinides plus some fission products. This will be determined after a more detailed examination of the cask inventory characteristics. TN did not state that it would need moderator exclusion to qualify the loaded TN-40 casks for transport. However, TN would not have burnup measurement verification for the 19 loaded TN-40 casks to support the burnup credit analysis and would need to rely on reactor records for burnup verification. Thus, an exemption to the regulations under the "special arrangements" provision may not be necessary.

5.5 Consideration of Moderator Exclusion in the Transportation of Storage-Only Systems

The early storage-only systems, whether canister-based or metal casks, have a host of issues associated with their transportation. Excluding the option of repackaging the fuel into transport-only casks, each of these systems may have to be transported under some form of "special arrangement," as described in Section 5.4. The details of the special arrangement will be unique to both the system (or even the canister/cask) being shipped, and, possibly, the locations of the shipping and receiving points. Moderator exclusion should be considered as one of many technical approaches that may have to be employed to make a comprehensive special arrangement or exemption/exception application.

Some of the PWR fuel was put into storage before the acceptance of "boron credit" during package loading in the spent fuel pool, so its criticality safety evaluation for storage considered the fully flooded condition and the internal structure in the canister or cask contained the necessary neutron poisons for criticality control during loading. However, there are a number of systems that were loaded with "boron credit," for which the criticality evaluation for a fully flooded cask during loading in the spent fuel storage pool took credit for the boron in the pool. Criticality safety under transportation conditions has not been evaluated for these "boron credit" cask systems.

As a general observation, the storage-only canistered systems with redundant closures shipped within a next-generation NRC certified overpack have a reasonable chance for approved transportation under existing 10 CFR 71 requirements (i.e., no special arrangements), or possibly with minimal restricted conditions (special arrangements). The double and even triple redundancy of containment, even if the containment provided by the welded canister does not have a contemporary pedigree, combined with some form of remote inspection of the canister exterior and some additional structural analysis or testing, may be sufficient to allow moderator exclusion. However, these canisters, including the internal basket structures, may have challenges in the structural area when analyzed for the hypothetical accident conditions specified in 10 CFR 71.73 (e.g., the nine meter drop). Structural testing of these storage-only canister-based systems may be required to demonstrate survivability under accident conditions.

Metal storage-only casks may have a more difficult time with moderator exclusion. The lack of redundancy in some closures may make the "single error" assumption in §71.55(c) difficult to overcome. However, there are external hardware additions that may be considered to make these casks more amenable to the application of moderator exclusion. For casks with acceptable materials of construction, it may be possible to weld the bolted lid or to create an additional new lid that encompasses the old lid. The entire cask could be enclosed in a containment-only overpack for that matter. Again, each system is unique. Such systems are likely to be candidates for the exemption or exception needed for a special arrangement shipment.

Regardless of the system, moderator exclusion can play an important role in making storage-only systems transportable.

5.6 Regulatory Considerations

5.6.1 Moderator Exclusion

The NRC approval of a special arrangement for the transportation of a storage-only SNF system may or may not require one or more exemptions or exceptions to 10 CFR 71. Any exemption/exception requests would be part of the overall safety evaluation for the package. If moderator exclusion is included in the argument for shipment equivalent safety, then the exception allowed by §71.55(c) could be invoked for moderator exclusion during normal conditions and the SFPO ISG-19 guidance could be used for moderator exclusion in the accident condition. If the NRC agrees that the normal condition moderator exclusion is truly an "exception" rather than a broad-based generic approval of moderator exclusion for an entire package design, no exemption will be required. That is why it is imperative for the cask designers to minimize the number of casks for which moderator exclusion is requested under §71.55(c) through the use of other licensing and analysis techniques, such as burnup credit and a one-time increase in the k_{eff}<0.95 reactivity acceptance criterion.

5.6.2 Containment System

The containment system for bare fuel metal casks is the bolted cask itself. These bolted closures typically include two concentric mechanical seals to provide a redundant barrier against leakage of radioactive material. These containment systems are licensed with a non-zero, but acceptably low leakage rate that is based on the containment analysis in the licensing basis. That is, the design basis leakage rate for the containment maintains the radionuclide release rate within the limits specified in 10 CFR 71. Moderator exclusion for these systems is more challenging, but not impossible, as discussed in Section 5.7.

The containment system for a welded canister-based system has typically been considered the bolted outer package, primarily for reasons discussed previously (i.e., the ability to conduct pre-shipment leakage testing). Certain systems licensed to transport loose plutonium, defined as "fuel debris" in the CoCs, credited the inner canister as the separate inner container (at the time) by 10 CFR 71.63(b). Changes in the regulations and the NRC review guidance have created new opportunities to re-define the containment boundary for canister-based transportation packages. First, the separate inner container requirement of 10 CFR 71.63(b) was rescinded through rulemaking, effective in October 2004. Second, SFPO ISG-18 permits leakage to be considered non-credible from appropriately designed, inspected, and tested austenitic steel fuel canisters. That is, no leakage of radioactive material from the containment boundary is required to be assumed. With these two regulatory changes, designers of canister-based spent fuel transportation cask systems interested in moderator exclusion should consider changing the definition of the containment boundary from the bolted outer closure to the welded inner closure. Clearly such a change has many detailed considerations such as ASME Code application, deep submergence issues, QA criteria, material selections, and manufacturing processes. But this should not deter anyone from investigating this paradigm shift.

Finally, if the traditional transportation cask containment system (i.e., the inner cavity) is retained, but canistered fuel is shipped, then the possibility exists for demonstrating that criticality safety criteria can be satisfied with a flooded containment but a dry canister. That is, moderator is present exterior to the canister but not within the canister.

5.7 Factors Supporting the Approval of Direct Off-Site Shipment

There are several factors regarding SNF stored in storage-only canisters that might assist in NRC approval for direct shipment of these packages without the need to open the packages, remove spent fuel, and transfer the spent fuel to a transportation cask.

First, the characteristics of the SNF assemblies stored in each storage-only system are reasonably well known. While burnup measurements were not universally performed for this SNF (i.e., some measurements were done in a few plants), the reliance on reactor records for burnups, enrichments, and fuel condition should provide sufficient information to support a safety case. Many of the fuel assemblies loaded have relatively low enrichments and burnups; therefore, burnup credit and/or moderator exclusion may not be necessary for the criticality safety assessment of each and every previously loaded storage-only system.

In addition, the QA records for the storage-only canisters or casks should be of sufficient quality to support a request to transport SNF in these packages. For transport of a sealed canister in a qualified Part 71 transport cask, the transportation casks are generally designed to be leak tight under the hypothetical accident conditions. Canistered systems include two welded lids that provide another level of safety for the safety assessment. While cask designers might not rely on the canister for containment since it is not possible to perform leak tests prior to shipment, containment would be provided by the transport overpack. It may be necessary to develop transport overpack designs to transport loaded storage-only canisters; thus, there is an opportunity to add features that would provide an additional level of safety to the canister and its contents during transport. Depending upon fuel characteristics in already loaded storage-only canisters, cask designers may also be able to rely on actinide-only burnup credit, burnup credit including fission products, and/or moderator exclusion, as well as provide additional risk insights for the transport of this material.

5.8 Factors that may Prohibit the Approval of Direct Off-Site Shipment

One must also consider a range of factors that may make it difficult for the NRC to approve the direct shipment of already loaded storage-only systems without the need to open the packages, remove spent fuel, and transfer the spent fuel to a transportation cask.

Much of the material that was loaded into the early dry storage systems was older SNF that might not have been characterized to the same level of rigor as today. While the characteristics of the SNF assemblies stored in each storage-only system are reasonably well known, the burnup verification was not as rigorous as it is today. For example, in cases where direct measurements were made, at the time there may not have been benchmarks of burnup measurements to reactor records. Thus, in cases where burnup credit is needed to transport the loaded package, the current application of burnup credit described in SFPO ISG-8, Rev. 2, may not be possible due to certain requirements that can only be met prior to loading into a canister (e.g., burnup measurement).

While quality assurance has always been a key element in the fabrication of spent fuel storage systems, there have been significant changes to the quality assurance requirements for spent fuel storage systems over the past decade. The application of contemporary QA requirements to older packages may show a record keeping deficit even though the records reflect full compliance with requirements at the time of construction.

For canistered systems, the performance of the canister internals that provide fuel support and spacing, and contain neutron poison, may not have been analyzed under the §71.73 hypothetical accident event sequences. It may be a challenge to demonstrate retroactively that the canister internals meet NRC acceptance criteria. The transportation overpack essentially must limit accident loads to canister-acceptable values.

5.9 Conclusions

The older storage-only systems can benefit from applying moderator exclusion technologies, but there are other issues with older systems besides criticality safety. System shipping may need to be made under some form of exemption/exception or special arrangement where moderator exclusion is one element in a comprehensive argument on equivalent transportation safety. Obtaining NRC approval for transportation will be challenging. Cost, risk and exposure studies should be made between the exemption/exception and the repackaging options.

6
MODERATOR EXCLUSION APPROACHES FOR DUAL-PURPOSE SYSTEMS

6.1 System Descriptions

Dual-purpose storage and transport systems evolved from the initial storage-only designs. Dual-purpose systems have been licensed under Part 72 for storage as well as Part 71 for transport. Both canister-based systems and metal casks have been licensed as dual-purpose systems. The majority of canister-based dual-purpose systems use a horizontal or vertical concrete overpack for onsite storage and a metal transport cask for shipment of the loaded canister off site. One system has licensed a metal cask for both storage and transport as well as including a concrete storage overpack. The description and licensing status of dual-purpose systems currently deployed or under NRC review is shown in Table 6-1.

Table 6-1
Dual-Purpose Systems Deployed at U.S. Nuclear Power Plant Sites or Under NRC Review

System	Description	Comments
FuelSolutions W-74	Canister, concrete storage overpack, metal transport cask	74 BWR (Big Rock Point), Part 71 certified
HI-STAR 100	Canister, metal storage/transport overpack	24 PWR, Part 71 certified 68 BWR, Part 71 certified 32 PWR, not yet Part 71 certified
HI-STORM 100	Canister, concrete storage overpack, HI-STAR transport cask	24 PWR, Part 71 certified 68 BWR, Part 71 certified 32 PWR, not yet Part 71 certified
NAC MPC/STC	Canister, concrete storage overpack, metal transport cask	24/26/36 PWR, Part 71 certified
NAC-STC	Metal dual purpose cask	26 PWR, Part 71 certified
NAC UMS	Canister, concrete storage overpack, metal transport cask	24 PWR, Part 71 certified 56 BWR, Part 71 certified
NAC MAGNASTOR	Canister, concrete storage overpack, metal transport cask	37 PWR/87 BWR, under Part 72 review, Part 71 not yet submitted
NUHOMS	Canister, concrete storage overpack. metal transport cask	07P, not yet Part 71 Certified 24PT, Part 71 certified 61BT, Part 71 certified 32PT, not yet Part 71 certified 32PTH/24PTH, not yet Part 71 certified
TN-68	Metal dual-purpose cask	68B, Part 71 certified

6.2 System Design Types

One dual-purpose design type is the welded canister system using a horizontal or vertical concrete structure for above ground shielding and heat management. Another dual dual-purpose canister-based design type is a welded canister contained in a vertical metal cask for storage and transport. A transfer cask moves canisters between the reactor spent fuel pool and the storage module. For off-site transportation, the welded canister is transferred from the storage overpack to the transport cask for shipment offsite. In the case of systems that use a metal cask for storage and transport, the loaded system can be transferred offsite without additional canister handling. Examples of the welded dual-purpose canister system are the HI-STORM 100, HI-STAR 100, NAC MPC, NAC UMS, NAC MAGNASTOR, FuelSolutions W74 and NUHOMS systems.

The second dual-purpose design type is a metal cask in which bare fuel assemblies are loaded into an integral fuel basket, much like a bare-fuel transportation cask. These metal dual-purpose casks have been licensed for both storage and transport. An example of the metal dual-purpose cask are the TN-68 and the NAC-STC.

6.3 Spent Fuel Characteristics for Dual-Purpose Systems

The initial dual-purpose cask designs were licensed to store spent fuel with somewhat limited fuel parameters – burnups up to 45 GWd/MTU, enrichments of approximately 4.5 w/o, and decay heat of 24 kW/package, with cooling times for either assembly type of five years or longer. Initial dual-purpose cask capacities were 24 PWR assemblies and up to 68 BWR assemblies. At these enrichments and number of assemblies in an array, and with poisoned baskets for certain designs, criticality was generally not an issue for storage or transport. All of the initial dual-purpose cask designs have received certification for both storage and transport.

The majority of these initial dual-purpose cask designs have since been amended one or more times to add more advanced canister designs that have higher assembly capacities or that allow the storage of spent fuel with higher initial enrichment and burnup, as well as higher decay heats. The amended designs also employed zoned loading of spent fuel that allows preferential loading of SNF with high decay heat in certain zones and SNF with lower-decay heat in other zones. Some of these amended designs have been approved for both storage and transport, while others (particularly advanced 32 PWR canister designs with high canister decay heat) have been approved for storage, but do not yet have NRC approval for transport of these canisters. These advanced designs will require either burnup credit or moderator exclusion as part of the criticality safety methodology for transport.

6.4 Dual-Purpose Cask System Limitations

As discussed in Section 2.3.3, NRC issued SFPO ISG-11, Revision 3, in November 2003. SFPO ISG-11 states that NRC staff is *"currently reevaluating the technical basis for the transportation of spent fuel including assemblies with average assembly burnups exceeding 45 GWd/MTU. The staff is reviewing data and technical reports to further understand the mechanical and fracture toughness properties of spent fuel cladding in relation to the transportation of high burnup fuel under 10 CFR 71.55. Therefore, until further guidance is developed, the transportation of high* *burnup commercial spent fuel will be handled on a case-by-case basis using the criteria given in 10 CFR 71.55, 10 CFR 71.43(f) and 10 CFR 71.51."* NRC Staff's concern is related to the potential for cladding degradation for high-burnup spent fuel and the subsequent reconfiguration of SNF geometries under the hypothetical accident conditions. Since many of the advanced dual-purpose canister designs have been certified under Part 72 for storage of SNF with burnups exceeding 45 GWd/MTU, the possibility of SNF cladding failure under the hypothetical accident conditions is an issue that must be addressed by licensees seeking Part 71 certification for these advanced dual-purpose canister designs.

Transport of advanced PWR spent fuel canisters, such as high capacity 32-assembly PWR canisters that have been approved for storage of SNF will require more complex criticality analyses due to the high initial enrichments of the loaded SNF and the increased assembly capacities. To meet the criticality safety requirements associated with the hypothetical accident conditions for transport, one approach taken by cask designers has been to include burnup credit methodology in the cask criticality safety assessment. As discussed in Section 2.3, NRC has approved actinide-only burnup credit methodologies in SFPO ISG-8. However, SFPO ISG-8 requirements for burnup measurement to validate fuel burnup could be resource intensive, and could not be achieved for spent fuel that has already been loaded and sealed in these high-capacity canisters. Moderator exclusion may be necessary for these packages in order to address the issues associated with reconfigured fuel geometries under the hypothetical accident conditions.

6.5 Consideration of Moderator Exclusion in the Transportation of Dual-Purpose Systems

As discussed in Section 2.3.4, SFPO ISG-19 addresses the criticality safety of SNF under the hypothetical accident conditions specified in §71.73. Particularly, the ISG addresses the criticality analysis associated with transport of SNF with burnups greater than 45 GWd/MTU. Due to the limitations about cladding properties under impact loading conditions typical of transportation accidents, applicants are presently unable to define the post-accident geometry of the high burnup fuel for use in a criticality evaluation. SFPO ISG-19 offers a reconfiguration alternative where an applicant can define a bounding reconfigured fuel geometry for computational purposes as long as it does not rely on material properties. It also offers a moderator exclusion alternative to the reconfigured fuel alternatives. The moderator exclusion option under SFPO ISG-19 involves physical testing of representative structures to measure leak tightness.

Transport of high capacity 32-assembly PWR canisters will require more complex criticality analyses due to the high initial enrichments of the loaded SNF and the increased assembly capacities. Due to the complexity and potential costs involved in NRC approval of burnup credit, including credit for fission products, cask licensees may consider the use of moderator exclusion in combination with burnup credit. Since most of these high-capacity canisters are also designed to store and transport SNF with burnups in excess of 45 GWd/MTU, the moderator exclusion option discussed in SFPO ISG-19 may provide licensees with an alternative to seeking full burnup credit as well as providing a means to satisfy the evaluation when addressing the possibility of reconfigured fuel geometries under the hypothetical accident conditions. Broad-based application of moderator exclusion under normal condition on other than an exception basis will require an exemption from existing regulations or rulemaking.

6.6 Regulatory Considerations

As more dual-purpose system designs evolve to allow the storage and transport of high burnup, high initial enrichment spent fuel, criticality safety considerations for transport of this fuel will begin to drive the system designs. In order to achieve high dual-purpose system capacities, cask designers will need to employ advanced criticality assessment methodologies. As noted in EPRI Report 1002879[16], SFPO ISG-8, Revision 2, which allows actinide-only burnup credit "can be regarded as providing as much burnup credit flexibility as can be currently expected for UO_2 fuel irradiated in PWRs only, with no burnup credit for fission products." Thus, cask designers seeking NRC approval for methodologies that take credit for fission products will likely encounter long reviews and substantial requests for additional information. Licensees can consider moderator exclusion as part of its criticality safety assessment. However, while NRC staff has indicated that it does not want to regulate by "exception", moderator exclusion may be possible within the current regulatory framework of §71.55 through an interpretation of these regulations and the development of supplementary guidance documents to allow the exclusion of moderator under certain conditions such as recognition of package design features that make water intrusion not "credible" under both normal and accident conditions (allowing water inleakage into the containment system but not the fuel region may be one aspect of this alternative). Another alternative would be for NRC to proceed directly to a rulemaking to modify §71.55(c) to remove the "exception" language. As discussed in Section 4.5, such a rulemaking would likely take several years until the new regulations are implemented and there are additional risks such as those associated with the NEPA evaluation that would be done in conjunction with the rulemaking.

6.7 Factors Supporting the Approval of Direct Off-Site Shipment

All of the original dual-purpose systems that were licensed in the late 1990s through early 2001 are already certified for both storage and transport off site. However, for several of the advanced designs that were submitted as amendments to the original certificates of compliance, NRC has not yet approved off-site transport of the loaded dual-purpose canister in a qualified transport cask. There are several factors regarding SNF stored in dual-purpose systems that provide support for direct shipment of these packages without the need to open the packages, remove spent fuel, and transfer the spent fuel to a transportation cask.

The characteristics of the SNF assemblies stored in each storage-only system are reasonably well known. For advanced, high capacity canister designs that have been loaded with spent fuel that may require burnup credit, while burnup measurements were not performed for this SNF, the reliance on reactor records for burnups, enrichments, and fuel condition should provide sufficient information to support a safety case. In general, greater attention to characteristics of the individual spent fuel assemblies is required to support zone loading of advanced canisters compared with earlier canister designs.

[16] *Fission Product Benchmarking for Burnup Credit Applications*. Progress Report, EPRI, Interim Report, December 2002. EPRI 1002879. p. vi.

The QA records for the fabrication and closure of dual-purpose canisters are of very high quality since these systems were fabricated and loaded after significant changes were made in the dry storage QA process.

Canisters are welded closed and there are two welded closures. Welded austenitic canisters are defined having “no credible leakage” by the NRC provided that the approved QA processes and materials are applied in accordance with SFPO ISG-18. In addition, the transportation overpacks are generally designed to have insignificant leakage under the hypothetical accident conditions. The canister internals that provide fuel support and spacing, heat removal, and criticality control have been analyzed under the accident sequences for all but criticality at the upper design limits.

Burnup credit and moderator exclusion may be available for advanced dual-purpose systems not yet loaded where additional measurements or hardware modifications will provide full compliance with SFPO ISG-8, SFPO ISG-18 and SFPO ISG-19. Again, moderator exclusion on a broad scale (i.e., for a generic package design) will require NRC to grant exceptions as allowed by §71.55(c), an interpretation of §71.55 or rulemaking. Lastly, for many of the dual-purpose canister system designs, only a small number of transportation overpacks have been constructed. This presents an opportunity for the potential addition of features that will provide additional protection and integrity to the canister.

6.8 Factors that may Prohibit the Approval of Direct Off-Site Shipment

While all of the original dual-purpose systems that were licensed in the late 1990s through early 2001 are already certified for both storage and transport off site, several of the advanced dual-purpose canister designs have not yet been approved under Part 71. While there have not been a significant number of these advanced canister designs loaded, there is a possibility that the high-capacity canisters that have been loaded for storage will not be approved for transport off site.

High capacity dual-purpose canisters that have already been loaded for storage may need to rely on burnup credit as part of the criticality safety analysis. Since the burnup of these SNF canisters was not measured to verify reactor records, the use of burnup credit as described in SFPO ISG-8, Rev 2, may not be possible. As discussed previously, actinide only burnup credit appears to be insufficient to provide sufficient criticality margin for high capacity systems and gaining approval for burnup credit that includes credit for fission products may be time consuming and costly.

6.9 Conclusions

The first generation of dual-purpose cask systems has been licensed for both storage and transport. The more advanced dual-purpose systems, with high capacities, that have been designed to store and transport high initial enrichment spent fuel with high burnups may be able to benefit from moderator exclusion, possibly in conjunction with advanced burnup credit methodologies.

To the extent that the industry continues to move toward these advanced, high capacity systems for storage of spent fuel, and given NRC’s reluctance to license by exception, this will require either an interpretation of §71.55 and the associated guidance documents to allow moderator exclusion under certain conditions such as recognition of package design features that make water intrusion not “credible” or rulemaking to remove the “exception” wording in §71.55(c).

7

MODERATOR EXCLUSION APPROACHES FOR NEXT GENERATION CASKS AND BARE FUEL TRANSPORT CASKS

The next generation of transport casks will not substantially differ from the systems of today. There will likely be a continued evolution in materials and methodology for future cask designs. For example, in the advanced dual-purpose system designs discussed in the previous section, cask designers will likely rely on heat transfer methods that consider convective cooling internal to the canister during storage to permit shorter cooling times, higher burnup, and/or greater fuel assembly capacity. Similar advances in methods, with data to support methodologies will be used in the design of next generation casks. It is important to note that, based on the one canister convection model approved to date (HI-STORM 100), canister convection cooling is of limited benefit in transportation because the internal convection design features are predicated on the cask being in the vertical (storage) orientation. Casks are transported in the horizontal orientation, where, due to the internal basket structure design, the convective fluid flow paths are not available.

While all of the recent cask systems that have been developed are dual-purpose systems, it must be recognized that the cask fleet that is likely to be used by the DOE to transport spent fuel from nuclear power plant sites will initially consist of “bare” fuel transport casks. At this time, there are only two modern casks, the TN-68 and the NAC-STC, that accommodate bare fuel for both storage and transport. In addition, the IF-300 rail casks and the NAC LWT and the GA-4 legal weight truck casks are also available designs for transporting bare fuel. These designs are limited in fuel parameters and capacity, and have not required either moderator exclusion or burnup credit, so more advanced or modified designs, including overweight truck casks, will be needed to accommodate a mid-range number of fuel assemblies, higher enrichments and burnups and shorter cooling times.

7.1 Spent Fuel Characteristics for Next Generation Casks

Next generation casks, whether dual-purpose or transport-only systems, will need to be capable of transporting spent fuel with assembly-average burnup up to ~60 GWd/MTU, short cooling times and high decay heat, initial enrichments up to 5.0 w/o, and advanced cladding materials. In addition, specialty casks that can transport damaged fuel, consolidated fuel, and fuel with various fuel assembly inserts will need to be developed to support the transport of spent fuel for disposal in a federal repository.

7.2 Next Generation Cask System Limitations

Next generation casks, whether dual-purpose or transport-only systems, will likely evolve from the current generation of transportation overpacks for canistered fuel or metal casks.

Next generation dual-purpose systems will face similar issues as the advanced dual-purpose systems currently under NRC review. These issues include the potential for cladding degradation and the effect on criticality safety under accident conditions; supporting data and analysis of the material properties and behavior of advanced cladding during storage and transport; advanced heat transfer methodologies including supporting experimental data for high capacity casks and transport of SNF with high decay heat levels.

Next generation transport-only casks will have issues similar to other non-canistered systems discussed in Section 3. For example, §71.55(c) invokes the single error assumption and conventional bare fuel casks typically have a single bolted closure lid. If next generation bare fuel casks evolve from current storage/transport designs, cask designers can build in redundant sealing for the confinement boundary. The bolted closure of a typical transport-only cask, suggests that for the cask to be eligible for moderator exclusion, separate redundancy and pre-shipping testing capability may be required.

7.3 Consideration of Moderator Exclusion in Next Generation Transport Casks

Since next generation casks will need to be capable of transporting spent fuel with assembly-average burnup up to ~60 GWd/MTU, short cooling times and high decay heat, initial enrichments up to 5.0 w/o, and advanced cladding materials, the consideration of moderator exclusion in cask criticality safety assessments will provide significant benefits for the design of the next generation systems.

If NRC either interprets §71.55 and associated guidance documents to allow moderator exclusion under certain conditions or proceeds with rulemaking to amend Part 71 to remove the "exception" language, designers of next generation casks will be able to develop the data needed to support an application that assumes moderator exclusion. For example, there will be opportunities for cask designers to develop QA records that will support the use of moderator exclusion, if additional measures are required in the new regulations. Next generation transport cask designs can accommodate the "single error" assumption and can limit or prevent water ingress through special features.

Next generation transport casks, whether dual-purpose or transport-only systems, will evolve from the advanced designs now being evaluated by the NRC. Next generation systems will need to be capable of transporting spent fuel with assembly-average burnup up to ~60 GWd/MTU, short cooling times and high decay heat, initial enrichments up to 5.0 w/o, and advanced cladding materials. Cask designers will require advances in technologies and methods in order to design transport systems that can handle the wide range of SNF that will need to be transported to a repository. The ability to provide design validation to established regulatory criteria for moderator exclusion in criticality safety assessments will be a valuable tool for these next generation systems. If changes are made to the Part 71.55 to allow moderator exclusion without the need for an exception, there may be additional requirements that will be imposed on package design features or quality assurance requirements.

8 CONCLUSIONS

Transport of spent fuel off-site without the need to repackage fuel that has been loaded into storage-only systems and certain dual-purpose systems is an issue that will have to be addressed by the owners and designers of these systems. When faced with possible restrictions or prohibitions on the transport of loaded storage systems due to issues associated with the criticality safety of these systems under 10 CFR 71, there are two tools available that may provide additional margin in the criticality safety risk assessment: moderator exclusion and burnup credit.

In addition to transport of already loaded storage-only and dual-purpose casks, there will be many challenges associated with approval of next generation transport casks. Next generation systems will need to be capable of transporting spent fuel with assembly average burnups up to ~60 GWd/MTU, short cooling times and high decay heat, initial enrichments up to 5.0 w/o, and advanced cladding materials. Cask designers will require some advances in technologies and methods in order to design transport systems that can handle the wide range of SNF that will need to be transported to a repository. The ability to assume moderator exclusion in criticality safety assessments will be a valuable tool for these next generation systems.

As this report demonstrates, burnup credit and moderator exclusion may be applicable singly or in combination. Of the two, moderator exclusion seems to hold the promise of a potentially less-costly path to success, particularly for advanced and next-generation technology. Preferably, moderator exclusion may be possible within the current regulatory framework of §71.55 through an interpretation of these regulations and the development of supplementary guidance documents to allow the exclusion of moderator under certain conditions such as recognition of package design features that make water intrusion not “credible” under both normal and accident conditions (allowing water inleakage into the containment system but not the fuel region may be one aspect of this alternative). Alternatively, NRC may proceed with rulemaking to amend §71.55(c) to relieve the NRC of having to use the exception approach to certification.

Cask designers may consider NRC approval of a special arrangement for the transportation of a storage-only SNF system through the use of existing exceptions or exemptions to 10 CFR 71. These exemptions/exceptions would be part of the overall safety evaluation for the package. If moderator exclusion is considered by cask designers as part of their overall arguments regarding the equivalent safety of these already-loaded storage only systems, the existing regulations contained in §71.55(c) can be relied upon for a small number of cases and NRC guidance in SFPO ISG-19 can be used for accident conditions.

For advanced dual-purpose systems, moderator exclusion may be used under the existing guidance in SFPO ISG-19 for accident conditions. Rulemaking may be required to license moderator exclusion on a broad basis for normal conditions.

If moderator exclusion is determined to be of value to the industry to assist in the transport of advanced dual-purpose designs and next-generation designs, then it is suggested that the nuclear industry and NRC begin to discuss how NRC would proceed with either (1) allowing moderator exclusion within the current regulatory framework of §71.55 through an interpretation of these regulations and the development of supplemental guidance to allow moderator exclusion under certain conditions or (2) rulemaking to change §71.55 to remove the “exception” language. Underlying these discussions should be an acknowledgement that the risks associated with a criticality accident are extremely small and would not result in any significant increase in transport risks. If it is decided that rulemaking should be pursued, rulemaking activities should commence as soon as possible since the rulemaking process and any ensuing administrative issues may be time consuming and complex. Alternatively, if moderator exclusion might be allowed within the existing regulatory frame of §71.55 with supplemental guidance under certain conditions, industry and NRC should begin dialogue to determine under what conditions moderator exclusion would be allowed.

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