ELECTRIC POWER RESEARCH INSTITUTE

Transportation of Commercial Spent Nuclear Fuel

Regulatory Issues Resolution

2010 TECHNICAL REPORT

Transportation of Commercial Spent Nuclear Fuel

Regulatory Issues Resolution

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ABSTRACT

The U.S. industry's limited efforts at licensing transportation packages characterized as "high-capacity," or containing "high-burnup" (>45 GWd/MTU) commercial spent nuclear fuel (CSNF), or both, have not been successful considering existing spent-fuel inventories that will have to be eventually transported. A holistic framework is proposed for resolving several CSNF transportation issues. The framework considers transportation risks, spent-fuel and cask-design features, and defense-in-depth in context of present regulations as well as in context of future potential revisions of regulations that would reflect a risk-informed, technically state-of-the-art approach. Within the boundary limits of cases analyzed, the EPRI-sponsored work shows that there are no credible combinations of accident events, accident locations, and fuel misloading or reconfiguration that would result in a critical configuration during the transportation of spent nuclear fuel. The non-mechanistic criticality evaluation performed in the as-loaded or as-designed configuration can be considered the bounding case for all conditions of transportation because this hypothetical reactivity case bounds all those normal and hypothetical accident cases that can *credibly* exist for spent-fuel transportation packages. Criticality during hypothetical transportation accidents should be a regulatory non-issue, given that misallocation of regulatory requirements can lead to greater overall risks, specifically by increasing the number of shipments when overly restricting spent-fuel transportation payloads.

Keywords
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1 INTRODUCTION

1.1 Purpose

This report examines the technical and regulatory issues surrounding the transportation of commercial spent nuclear fuel (CSNF). The industry's limited efforts at licensing transportation packages characterized as "high-capacity," or containing "high-burnup" (>45 GWd/MTU) CSNF, or both, have not been successful from the perspective of the fraction of existing spent fuel inventories that can actually be transported. A major crux of this problem is how, and to what extent, credit for the reduced reactivity of the fuel due to burnup ("burnup credit") can be taken in the criticality analyses used in transportation licensing. In this report, a holistic framework is proposed for developing generic acceptance criteria for transporting commercial spent nuclear fuel (CSNF), which considers transportation risks, spent fuel and cask-design features, and defense-in-depth in the context of present and potentially future revisions of regulations that would reflect a risk-informed, technically state-of-the-art approach.

1.2 From Storage-only to Dual-purpose Systems

1.2.1 Mid-1980s through Mid-1990s

Commercial nuclear power plant operators in the United States (hereafter referred to as "the utilities") began dry storage of spent fuel at Independent Spent Fuel Storage Installations (ISFSIs) at the plant sites in 1986 to provide additional storage capacity beyond that available in the plants' spent fuel pools. The federal government of the United States in 1982 approved the Nuclear Waste Policy Act (NWPA) directing the Department of Energy (DOE) to begin removing spent nuclear fuel from commercial nuclear power plant sites by January 31, 1998. Because DOE did not meet this obligation, the utilities have needed more and longer duration storage in ISFSIs.

The first wave of ISFSI spent fuel storage effort utilized so-called "storage-only" casks. These were casks certified by the U.S. Nuclear Regulatory Commission (NRC) for storage of the spent fuel under the provisions of 10 CFR Part 72, with the expectation that one day the fuel would be re-packaged into casks certified for transportation under 10 CFR Part 71 for shipment to the federal repository. These storage cask designs were "bare fuel" casks, or non-canister based, bolted lid systems. While some bare fuel cask designs are also certified for transportation, most are not. The bolted-lid system facilitates re-opening of casks in the spent fuel pool for re-packaging of their contents into a transport package.

1.2.2 Mid-1990s through Today

The cask designers eventually saw a market for canister-based, dual-purpose spent fuel cask designs. In this concept, a fully welded spent fuel canister is designed and licensed under 10 CFR 72 for onsite storage at the ISFSI when placed inside a ventilated vertical cask or ventilated horizontal module. That same inner canister is also separately licensed for transportation under 10 CFR 71 when placed in a robust, sealed outer packaging, such as a bolted lid steel overpack. This “once-and-done” concept provided the utilities with the convenience of loading the canister once in the spent fuel pool with no need for re-packaging of the individual fuel assemblies for transportation. Transfer of the canister from the storage cask or module into the transportation overpack can be performed without moving the canister back into the spent fuel pool. For shutdown plants, this was particularly attractive because it allowed them to decommission and dismantle their power plant, retaining only the ISFSI. The balance of the plant property could be remediated in order to be dedicated for whatever land use would be desired.

The first dual-purpose, canister-based systems were licensed for moderate capacity (i.e., 21 PWR assemblies and 52 BWR assemblies). The utilities embraced this dual-purpose spent fuel management system concept and a large majority of the fuel in ISFSI dry storage today resides in dual-purpose canisters. A smaller, but still significant amount of fuel is still being loaded into (i) storage-only, canister-based systems, and (ii) transportable bare fuel casks.

The first dual-purpose systems were licensed effectively concurrently, such that the design and contents of the casks/canisters were certified for storage and transportation. That is, there was no question left unanswered as to the transportability of a particular cask system design or its approved contents. Over time, the design capacity (i.e., number of fuel assemblies) and contents’ initial enrichment increased in response to utilities’ demand for loading fewer casks in a more cost-effective manner and with higher enriched, longer burned fuel, consistent with evolving reactor operation regimes. This market demand led the cask designers to focus almost exclusively on certifying the higher capacity systems for storage, with transportation certification lagging behind. The transportability of these higher capacity casks and canisters has now become a concern for the utilities and industry as a whole.

1.3 Regulatory Overview

As already mentioned, the packaging and transportation of radioactive material in the United States is regulated by the NRC under Part 71 of Title 10 in the Code of Federal Regulations (10 CFR 71). The NRC grants a 10 CFR 71 certificate of compliance (CoC) for radioactive material transportation packages upon successful review and approval of the CoC application. The Part 71 regulations include generic requirements for all radioactive material packages as well as unique requirements for specific types of packages. Only those requirements applicable to the criticality control functions and analyses of packages used to transport fissile material in the form of commercial spent nuclear fuel (CSNF) are of interest in this study. Appendix A provides a more detailed, annotated description of applicable regulations and associated regulatory guidance.

The 10 CFR Part 71 regulations are largely deterministic. That is, they provide specific design features, test requirements, and package performance criteria. In the areas where the regulations do not specifically call out the design or performance requirement, guidance has been published by the NRC. The guidance comes in two forms. The first is in the form of guidance tailored for the cask designer and analyst, such as Regulatory Guides and NUREG documents. The second is review guidance used by the NRC staff to scope and structure their reviews of submitted applications for package design certification, i.e., the standard review plan (SRP) [1]. In several areas, and in particular in the area of criticality control and analysis, the state-of-the-art of the analysis work has been evolving rapidly. Thus, the NRC has chosen to make interim changes to the SRP using Interim Staff Guidance (ISG) documents, given that revisions to the SRP are infrequent.

The regulatory guidance, comprised of the SRP and ISGs, includes numerous conservatisms. Whereas the regulations are not specific as to the acceptance criteria for the package criticality evaluation [10 CFR 71.55(d) and 10 CFR 71.55(e) require the package to be "subcritical" under normal and accident conditions, respectively], the SRP establishes a specific reactivity acceptance criterion of k_{eff} <0.95 with a 95% confidence factor. This effectively establishes a five percent minimum safety margin irrespective of the specific contents or design features of the package or the likelihood of the package ever having the fuel cavity flooded with pure water. Other conservatisms include limited burnup credit; limits on neutron absorber credit, such as 75% or 90% depending on material; maximum moderator density, or 1 g/cm^3; flooding of the pellet-cladding gap; etc.

There are several key unresolved criticality safety issues that affect the analyses performed by transportation cask certificate applicants and the reviews of these analyses by the NRC staff. They include incomplete, inconsistent, or overly conservative regulatory guidance in the areas of transportability of high-burnup CSNF[1], burnup credit, fuel assembly burnup measurement, and moderator exclusion.

In the absence of practical and predictable regulatory acceptance criteria for addressing these issues, regulatory reviews of CSNF transportation package CoC amendment submittals in general, and burnup credit applications in particular, require long periods of time (years rather than months), and result in extremely limited approved contents for those applications that are approved. This leaves the U.S. nuclear industry without the confidence that their high burnup, higher enrichment spent fuel, currently being placed into storage in dual-purpose canisters and casks will be able to be transported off site. The problem intensifies on a regular basis as more high-capacity dual-purpose casks and canisters are loaded and placed into spent fuel storage facilities at the reactor sites.

[1] High-burnup CSNF is spent fuel with discharge burnup greater than 45 GWd/MTU

2
REGULATORY ISSUES

2.1 Transportability of High Burnup Commercial Spent Nuclear Fuel

Most of the debate (and R&D) has centered about cladding integrity/performance under transportation accident conditions, as being the key to satisfying the regulatory requirement contained in Part 71.55(e) stipulating that the transportation package contents have to remain subcritical under hypothetical accident conditions.

High burnup (HBU) commercial spent nuclear fuel (CSNF) is understood to mean fuel burned in a reactor to greater than 45 GWd/MTU. For burnup less than or equal to 45 GWd/MTU, the NRC has concluded that hypothetical transportation accidents do not result in significant damage to, or reconfiguration of, the spent fuel. This is based on the fact that sufficient evidence[2] 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) (i.e., for "as-loaded" or "as-designed" configuration). For burnup greater than 45 GWd/MTU, the NRC has concluded that they could no longer assume that no significant fuel damage and reconfiguration would result[3]. Therefore, changes in the packaging under hypothetical accident conditions that could cause the nuclear reactivity to increase need to be addressed.

Clearly, assuming all other relevant parameters being similar, a higher discharge burnup results in higher radiation fields (shielding is more challenging) and in larger source terms (maintaining waste package integrity becomes even more important). But as far as nuclear reactivity is concerned, the higher the burnup, the better it is for criticality safety![4]

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[2] See, for example, Appendix III "Spent-fuel Response to Transport Environments" in T. L. Sanders et al., *A Method for Determining the Spent-Fuel Contribution to Transport Cask Containment requirements*, SAND90-2406 (November 1992)

[3] "It is judged that, at this time, there is insufficient material property information for high burnup fuel to allow this type of evaluation." Excerpt from Interim Staff Guidance 19, *Moderator Exclusion under Hypothetical Accident Conditions and Demonstrating Subcriticality of Spent Fuel under the Requirements of 10 CFR 71.55(e),* NRC Spent Fuel Project Office (May 2003)

[4] Commercial nuclear fuel is limited to an initial ^{235}U enrichment of 5% or less. Therefore, a large fraction of its nuclear reactivity will have been consumed at a burnup greater than 45 GWd/MTU.

2.2 Burnup Credit Methodology

Burnup is a physical reality. How much credit can be claimed for this physical reality is the main issue. It can cover the range from none ("fresh fuel" assumption) to best-estimate "full" (actinides + fission products) burnup credit. The negative reactivity effect of burned fuel is used throughout the nuclear industry in areas such as wet storage rack design and reactor core re-load analyses. The regulatory environment has a long history with the use of burnup credit in these areas. The level of maturity with the preparation and NRC review of burnup credit analyses has allowed the methodology to evolve to include full fission product credit.[5]

Burnup credit has been sought for the transportation of CSNF for over two decades. NRC's review guidance for spent fuel storage and transportation casks practically prohibited any use of burnup credit until 1999. As a result, cask designers have historically assumed unirradiated fresh fuel in every storage location in the cask in the criticality analyses. As time passed, cask designers increased capacity and fuel enrichment limits for their spent fuel cask product lines to respond to the market in the late 1990s. The first issuance of the ISG dealing with burnup credit was issued in 1999. Its latest revision, issued in 2002 [5], endorsed actinide-only burnup credit. Another revision, Revision 3, is expected in 2011. Experimental data necessary for validation of the isotopic compositions and the nuclear cross sections of fission products have not been deemed adequate thus far, and approval of full burnup credit for transportation applications has been subsequently delayed. High-capacity PWR casks and dual-purpose canisters have been loaded for storage since 2000 and, except for the Holtec MPC-32 [6] that can accommodate a limited range of fuel enrichment/burnup, none have been licensed for transportation of high-burnup fuel.

2.3 Fuel Assembly Burnup Measurement

ISG-8 requires in-pool measurement of the burnup of fuel assemblies chosen for emplacement in a transportation or storage cask licensed with burnup credit to confirm the reactor burnup record for the assembly. However, in most instances, fuel assembly burnup information is already well characterized and quality records corroborated by *in-core measurements* already exist.

2.4 Moderator Exclusion

Fissile material transportation packages, including CSNF transportation casks, by regulatory fiat, must be assumed to be flooded with unborated water and analyzed to be subcritical. This condition is not considered a normal condition of transportation or a result of a hypothetical accident. It is a non-mechanistic requirement that leaves the cask designers with no way to design any CSNF transport cask to exclude moderator and permit the criticality analysis to be

[5] Over the past couple of years, NRC guidance in the area of spent fuel pool criticality has been changing and as a result, significant uncertainty and unpredictability has been introduced into the licensing process, specifically with regard to the staff reliance on the "Kopp memo" [2]. The NRC has characterized the situation as "Experiencing short term licensing uncertainty in order to achieve long term predictability" [3], and has developed a draft interim staff guidance on spent fuel pool criticality [4].

performed accordingly. The NRC staff has made it clear that exceptions to the moderator intrusion requirement, as permitted by the regulations, will not be granted on a package *design* basis. Further, the NRC Commission has disapproved an NRC staff recommendation to revise the regulations to allow some flexibility in the moderator intrusion requirement.

The NRC issued ISG-19 [7] permitting cask designers the ability to credit design features for providing the moderator exclusion function during hypothetical accident conditions provided testing was performed. However, this guidance is of limited use for cask designers given the moderator intrusion requirement of §71.55(b). The position in ISG-19 is inconsistent to the extent that the guidance shows a regulatory willingness to accept moderator exclusion under accident conditions, i.e., under loss-of-control conditions (accidents typically result from loss of control), but not under normal configuration conditions when operational controls are assumed to be respected.

3
ISSUE RESOLUTION ELEMENTS

3.1 Risk Assessment

Risks during transportation of CNSF have been addressed by the NRC in References [8] and [9]. These studies assess the risks of accidents capable of breaching the transportation package and resulting in the release of radioactive material to the environment, but do not quantify the frequency of a criticality event.

The only mechanistic manner in which pure water could unexpectedly infiltrate a spent fuel transportation package would be during an accident condition that occurs near a body of water. A detailed analysis of the probability of a criticality event during railroad transportation of CSNF was performed [10] [11], and the results are summarized hereafter.

3.1.1 Probability of a Critical Event during Transportation

To assess the probability of a critical event during railroad transport of CSNF, the following were considered:

1. Probability that fuel assemblies in the transportation package have sufficient nuclear reactivity to produce criticality
2. Frequency of railroad transportation accident
3. Probability of the transportation package suffering damage sufficient to permit in-leakage of water
4. Probability of becoming submerged in water resulting in internal flooding that would produce the geometry, moderation, and reflection conditions necessary to produce criticality.

Table 3-1 shows that the likelihood of a criticality event during transportation of a 32-PWR spent fuel assembly package is equal to ~10^{-16}/shipment, which is well below any credible event probability historically considered in regulatory practice. This result arises from a number of independent factors:

- The extremely low likelihood that a railroad accident will produce the damage and immersion needed to achieve criticality, as determined by the U.S. NRC-sponsored research.
- The very low likelihood of an error in the recorded burnup of fuel assemblies due to flux mapping measurements and use of fuel assembly burnup to predict and verify core performance during active fuel cycles in the core.

- The low likelihood of a misload due to the controls and verification requirements followed when loading fuel assemblies into the spent-fuel cask.
- The ability to access core burnup and Special Nuclear Material accountability data at any time prior to shipment of a spent fuel cask offsite in order to verify compliance to the cask's CoC.

Table 3-1
Summary of the Risk of Criticality during Railroad Transportation [10]

Description	Freight Trains
Train Accidents per Train-Mile (All Accidents, All Speeds, All Track Classes), 2000 - May 2006.	2.7E-06
Probability of Accident of Interest, Given Any Accident (>2% Strain and Immersion) per Modal Study	7.8E-09
Frequency of Accidents of Interest for Criticality/Train-Mile	2.1E-14
Assumed Average Number of Miles per Shipment	2,000
Frequency of Accidents of Interest for Criticality/Shipment	4.2E-11
Likelihood of Shipping a Misloaded Spent Fuel Cask	2.6E-06
Likelihood of an Accident with a Potential for Criticality/Shipment	1.1E-16

A number of operational safeguards and controls would further reduce the risks, such as more closely controlling and monitoring the trains transporting spent nuclear fuel than the generic population of freight trains evaluated in the risk assessment. For example, train speed limits can be established below a threshold speed needed to produce damage. They could be further reduced selectively for those stretches of track that have the close proximity to water deep enough to fully immerse a spent fuel cask. A National Academy of Sciences (NAS) committee also recommended that no other traffic should be allowed in tunnels during transit of spent fuel through the tunnel [20].

3.2 Critical Configuration

The potential for a critical configuration depends on:

- Amount of nuclear reactivity inadvertently introduced as a result of misloading, and
- Reconfiguration of fuel that would result in a higher nuclear reactivity.

3.2.1 Misloads

The likelihood of shipping a misloaded spent fuel cask, equal to $2x10^{-6}$ as shown in Table 3-1, carries the assumption that a *single* misloaded fuel assembly would introduce sufficient reactivity for resulting in a critical event.

EPRI examined the impact of misloading both “under-burned” and fresh fuel into a 24-PWR-assembly cask design[6] containing fixed boron neutron absorbers [12]. “Under-burned” fuel is spent fuel that has not achieved the burnup obtained from the reactor core-follow calculations used to plan reactor operations and reloading schemes. The calculations accounted for actinide depletion and buildup of five neutron-absorbing fission products. The results are shown in Figure 3-1 for spent PWR fuel with a discharge burnup of 45 GWd/MTU from a hypothetical fuel cycle utilizing 5% enriched fuel. The value for k_{eff} of ~0.88 with no misloaded fuel assemblies indicates that there is considerable margin for uncertainty in the calculations when the presence of neutron absorbers actually in the fuel is accounted for.

The two sensitivity cases in Figure 3-1 show the impact of the substitution of up to eight assemblies (i) with a burnup of only 25 GWd/MTU (about 56% of the design burnup), and (ii) consisting of fresh fuel. For both cases, the substitution of these assemblies is made by grouping them together in the middle of the cask, which produces the highest increase in k_{eff}. It can be seen that the substitution of up to eight fuel assemblies with a burnup of 25 GWd/MTU results in a maximum k_{eff} of only ~0.95. Furthermore, it shows that more than one assembly of fresh fuel must be misloaded into the cask to result in a k_{eff} greater than 0.95, while three are needed to produce criticality. As expected, the consequences of a misload with fresh fuel would be more significant. Misloading a single fresh assembly with 3, 4, or 5 wt% ^{235}U enrichment would result in an increase in k_{eff} of ~0.02, 0.04, and 0.06, respectively [13].

Therefore, even assuming the worst possible nuclear reactivity to start with, i.e., $k_{eff} = 0.95$, the assumption that a *single* misloaded fuel assembly would introduce sufficient reactivity ($\Delta k_{eff} \geq 0.05$) necessary for a critical event is true only when the misload involves a fresh 5% enrichment assembly. However, there is a very low likelihood that fresh fuel would be in the spent fuel pool when spent fuel casks are loaded. Since all fuel is handled by one group within a plant and spent fuel pool space is limited, spent fuel cask loading is typically scheduled to be made early in a fuel cycle run to make room for the next refueling operation. The new fuel is received into the new fuel storage area, where it is inspected and stored to just prior to refueling.[7] In addition, there is a distinct difference in the appearance of fresh and once-burned fuel assemblies, as illustrated in Figure 3-2.

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[6] A sensitivity study shows that cask size does not significantly alter the k_{eff} of the package. For the calculations using spent fuel with the same burnup, the criticality calculations for a 32-PWR assembly cask produce essentially the same k_{eff} as a 24-PWR assembly cask. This results from the fact that the calculation groups all the under-burned assemblies in the center of the cask in their most reactive configuration. The misloaded fuel acts like a small core surrounded by a reflector of the more highly burned assemblies directly adjacent to the misloaded fuel. Addition of assemblies beyond the adjacent assemblies increases k_{eff} only to the extent that net neutron leakage out of the misloaded region is reduced. As the immediately adjacent assemblies provide the vast majority of reflection back into the misloaded group, the overall effect on k_{eff} of using a larger cask is insignificant.

[7] Some plants that changed from a three to two cycle shuffle may have to transfer new fuel to the spent fuel pool to make room for the last shipment of new fuel, but the work necessary for processing the new fuel will take priority over any loading of spent fuel casks.

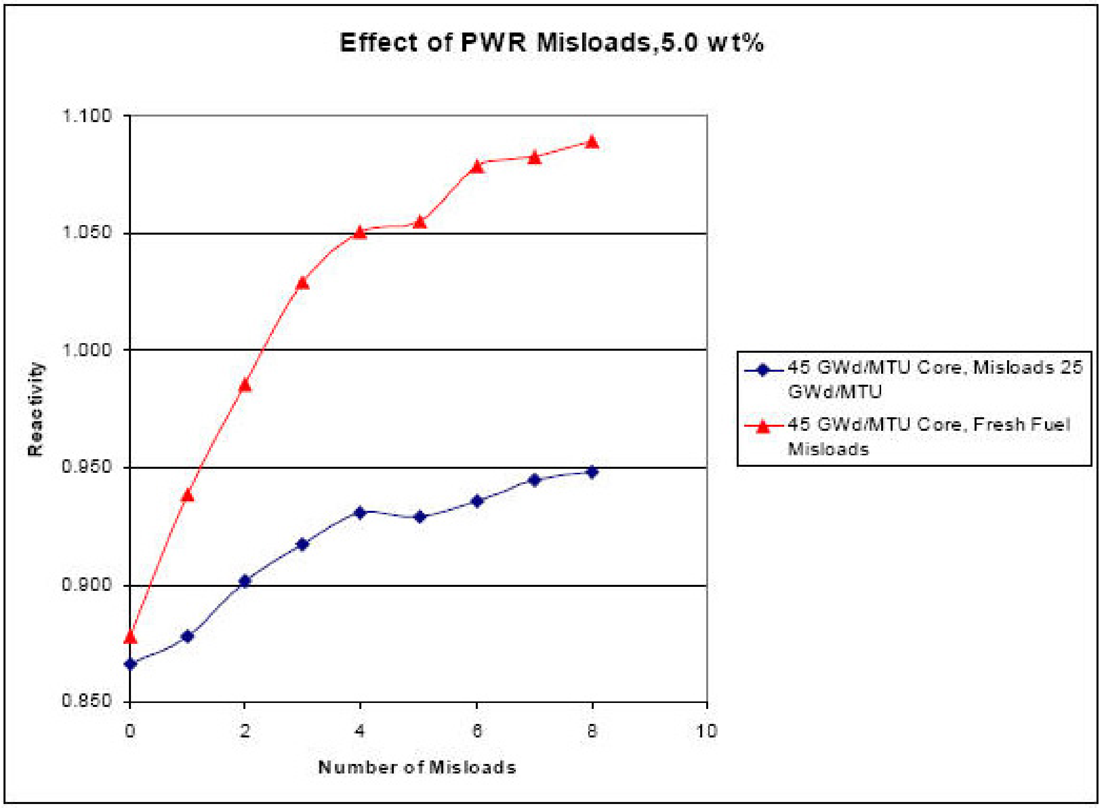

Figure 3-1
Effect of misloaded fuel assemblies on the k_{eff} of a conceptual 24-PWR spent fuel cask [12]

Figure 3-2 shows an arrangement of fuel assemblies during a refueling operation. The fresh fuel assemblies have their original metallic color, while the once-burned assemblies have been darkened by crud deposition and/or corrosion. The risk assessment takes no credit for the ability of members of the refueling team to recognize the differences between a fresh fuel assembly and a once-burned fuel assembly. However, the readily recognizable difference in appearance provides additional assurance that the likelihood of a misload will not involve a fresh fuel assembly.

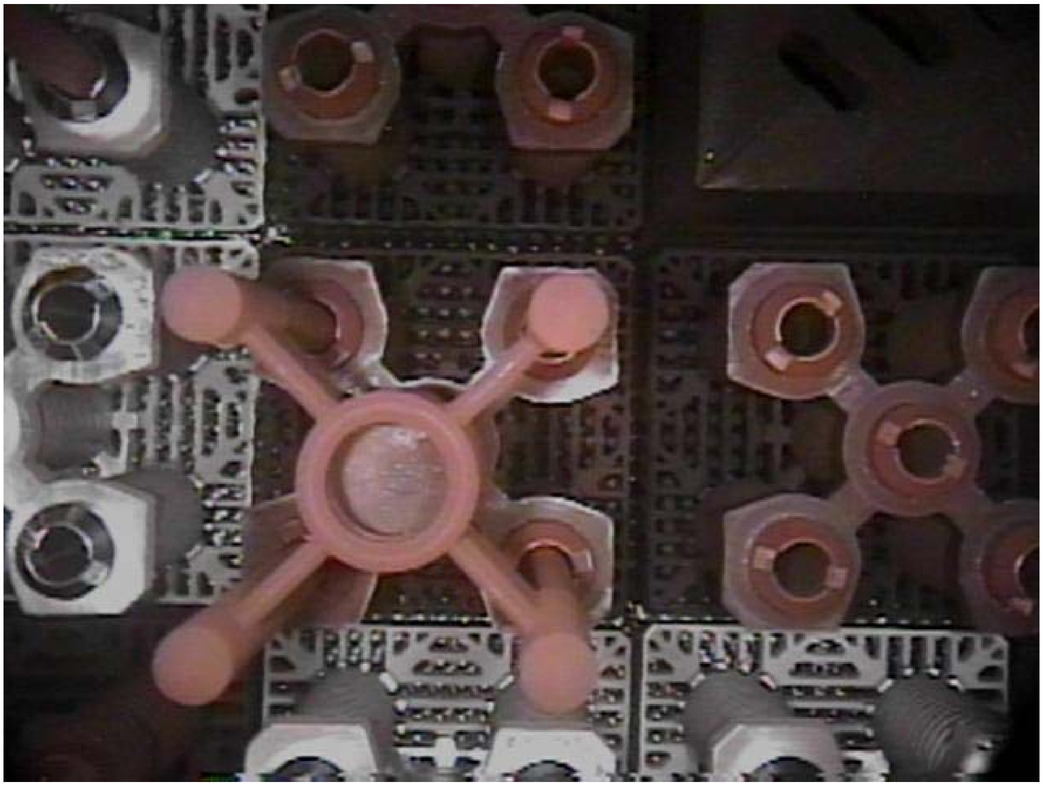

Figure 3-2
New and Once-Burned Fuel in a Reactor Core

3.2.2 High Burnup Fuel Reconfiguration

One of the U.S. NRC spent fuel transportation system requirements is that criticality calculations be performed with the fuel geometry in its most *credible* configuration that would maximize k_{eff}. As long as the cladding remains sufficiently ductile, damage to the spent fuel in a transportation accident would be minor [16]. Today's fuel designs are expected to be burned in excess of 45 GWd/MTU. According to ISG-19 [7]:

> *"Due to effects of irradiation, the cladding of spent fuel, and particularly high burnup fuel (i.e., fuel with a burnup greater than 45,000 MWD/MTU) may become brittle. If excessively brittle, the cladding could fracture under impact loads currently associated with hypothetical accident free drop test conditions. Consequently, criticality safety of the reconfigured fuel assembly must be demonstrated."*

3.2.2.1 Worst-Case Scenarios

The effects of "worst-case" accident scenarios were surveyed in Reference [14]. The survey used scenarios that were postulated to provide theoretical upper limits for reactivity effects of fuel relocation, although they were described as going "beyond credible conditions."

In order to provide credible estimates of the probability and maximum reactivity changes, EPRI delved deeper into the physical conditions that make up the theoretical scenarios and applied physical limits based on current cask design practices [15]. The scenarios involved physical changes either to fuel assembly rod arrays or to collections of fuel pellets with the fuel skeleton removed. These scenarios were deconstructed into a set of scenarios and the physical phenomena required to create the scenario were identified. The boundary between credible (but unlikely) and incredible scenarios is easily discernible with this methodology.

The study showed that the unyielding cask basket structure prevents fuel rod arrays from attaining optimum moderation conditions, thus limiting reactivity increases. The study concluded that the maximum reasonable reactivity increase for unlikely, but perhaps credible, “worst-case” scenarios was either less than the administrative margin of 0.05 for scenarios involving physical changes to fuel assembly rod arrays (Figure 3-3), or a substantial reactivity decrease for scenarios involving physical changes to free pellet arrays.

Figure 3-3 shows the most reactive case corresponding to fuel rods expanding within the basket fuel cell with all rods remaining parallel and equally spaced (without grids), with increasing water moderation. The maximum reactivity increase can be calculated by expanding the fuel basket cells and unrealistically allowing the cask diameter to grow to accommodate the larger basket, as shown in Figure 3-3. The reactivity increases until optimum moderation is reached, and then it decreases as the array becomes over-moderated. Alternatively, if the cask basket is unyielding (i.e., is not capable of expanding), which is a much more reasonable assumption, then fuel rods must be removed from the array to provide the space needed for expansion, and optimum moderation is reached with a smaller array within the fixed basket cell. This case is more pertinent for spent fuel casks, and the maximum reactivity is reduced by half, from ~0.06 to ~0.03.

3.2.2.2 Best-Estimate Case Scenarios

Transportation accidents postulated for spent fuel shipments are bounded by the regulatory hypothetical accident of a 9-meter drop of a rail transport cask onto an unyielding surface. Of the three possible drop orientations of the cask at impact, namely the end drop, the corner drop with slap-down and the side drop, the latter is the most severe because it activates the cladding failure mode with the highest failure potential.

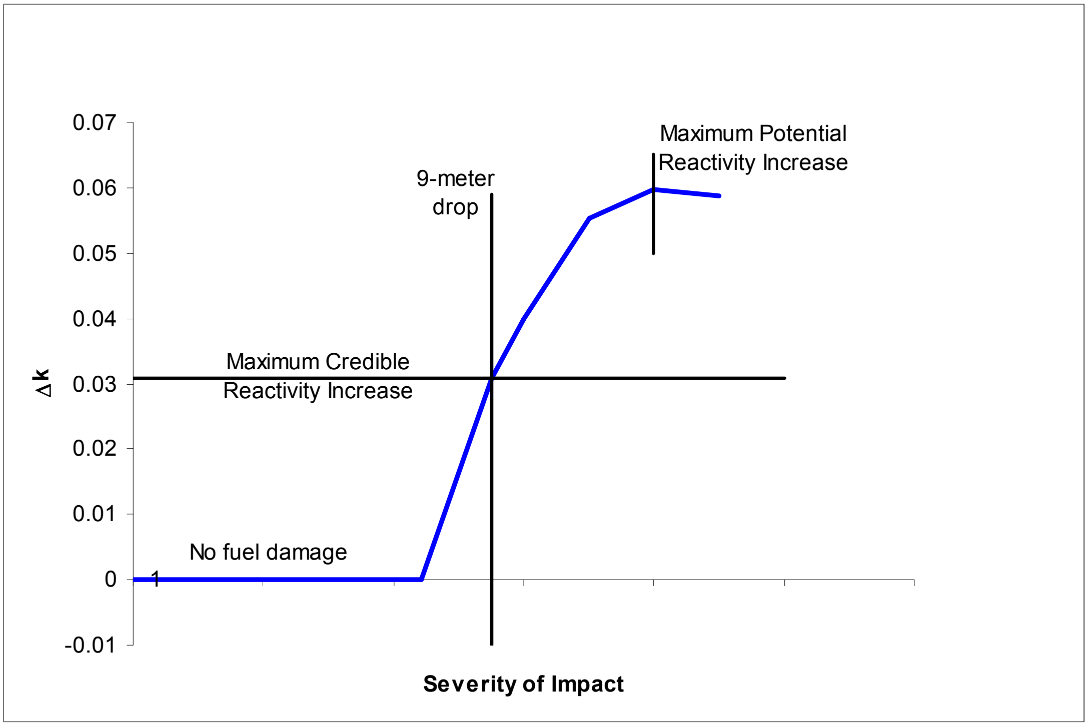

Figure 3-3
Cask System Reactivity versus Cask End Drop Severity

The source-term study conducted by Sandia National Laboratories nearly two decades ago for the spent fuel inventory known at the time, which was in the low-to-medium burnup range (~35 GWd/MTU), showed that the effects of transportation accidents on spent fuel failures, and consequential radioactivity release to the environment, were relatively benign [16]. The results from these studies have provided the justification for ignoring significant damage and potential reconfiguration as a result of accident conditions. However, with today's discharged fuel burnup routinely greater than 45 GWd/MTU, potential hydride reorientation during interim dry storage and its effects on cladding properties have become one of the primary cladding performance concerns for spent fuel transportation.

Laboratory tests of un-irradiated and irradiated cladding specimens subjected to heat treatments promoting hydride dissolution followed by re-precipitation in the radial direction have shown that relatively moderate concentrations of radial hydrides can significantly degrade cladding ductility, at least at room temperature. The absence of specific data that are relevant to high-burnup spent fuel under dry storage conditions have led to the conjecture, deduced from those tests, that massive cladding failures, possibly resulting in fuel reconfiguration, could be expected during cask drop events. Such conclusions are not borne out by the findings in the EPRI studies [17] [18] [19], as discussed below.

There are three types of physical and material conditions of spent fuel rods at the end of dry storage that could have an effect on cladding failure behavior under transportation accident conditions. These are:

(a) Burnup-dependent conditions, such as the dependence of cladding mechanical properties on irradiation and cladding thickness loss due to corrosion, which affect the magnitude of cladding deformations.

(b) Dry-storage conditions, such as creep-induced fuel-cladding gap and hydride re-orientation, which affect cladding resistance and vulnerability to failure.

(c) Cladding defects, such as hydride lenses and incipient cracks, which behave as precursors for cladding failure initiation.

These are incorporated in the EPRI methodology, and their effects are reflected in the results. The results indicate that type (a) conditions play an indirect role in cladding failure behavior, namely, through their effects on cladding deformations. Type (b) conditions play a very direct role in cladding failure behavior in two ways: firstly, through the effects of radial hydrides on cladding fracture resistance, and, secondly, through the effect of the fuel-cladding gap size on limiting cladding deformations due to fuel pellets participation in resisting the load. This latter effect of gap size plays a similar role in the behavior of type (c) defects, where cladding contact with the fuel pellets prevents the propagation of cracks or surface defects to through-wall failures.

The analysis results indicate that cladding failure is bi-modal: a state of failure initiation at the inside wall of the cladding remaining as part-wall damage with less than 2% probability of occurrence, and a through-wall failure at a probability of $\sim 10^{-5}$. It is important to note in this regard that the through-wall cladding failure probability of $\sim 10^{-5}$ is of the same order of magnitude as calculated in the Sandia study for lower burnup fuel [16]. A summary account of EPRI results is shown in Attachment B.

In summary, the EPRI studies showed that breakage is limited when considering the nine-meter drop scenario and fuel reconfiguration of a magnitude discussed in worst-case scenarios is not credible.

3.3 Moderator Exclusion and Burnup Credit

Risk information supports the concept of defense-in-depth when considering that moderator exclusion or burnup credit could be used singly or in combination in the design of transportation package:

5. For a “burnup-credit” package design, it is highly unlikely that fuel would be exposed to water during transportation because of the low frequency of severe enough accidents in the proximity of a water body [10] [11].
6. For a “moderator-exclusion” package design, it is highly unlikely that even if water flooded the package, the package could ever form a critical configuration. A companion requirement could impose a condition that k_{eff} be shown to be below 1 when using a *best-estimate* burnup credit methodology.

Of the two options, moderator exclusion would seem 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 from 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 in-leakage is not credible [21].

4 ISSUES RESOLUTION APPROACH SUMMARY

Risk information indicates that the probability of a critical event during transportation is essentially zero.

1. Casks are designed and their contents so limited that under the most reactive conditions, with pure water in the fuel cavity, the k_{eff} for the reactivity system must be calculated to be less than or equal to 0.95 using very conservative assumptions. The criticality analysis assumes that each fuel assembly in the cask is at its minimum required burnup for its enrichment as specified in the loading curve contained in the CoC. This is required to ensure that the licensing basis criticality analyses for the cask are bounding for all combinations of fuel permitted by the CoC for loading. Other physical parameters used in the criticality analysis are also assumed to be at their limiting value that maximizes the reactivity of the system. In a properly loaded cask, at least some of the fuel is burned to higher levels than the minimum CoC requirement, which provides additional safety margin beyond that shown in the criticality analyses. EPRI work shows that the actual reactivity of a properly loaded 24-assembly cask with fixed neutron absorbers and considering the burnup credit from five fission products would be on the order of k_{eff} = ~0.8 – 0.9, as illustrated in Figure 3-1. This represents significant additional criticality safety margin before any misloading event is considered.
2. Transportation accidents that are severe enough to result in an opening in the transportation package in the presence of water are very low probability events.
3. Assuming that the package contents maintain their normal configuration during the accident, misloading of under-burned fuel does not result in a critical configuration, except under extremely unlikely assumptions (misloading of *fresh* fuel enriched at ~5%).
4. Assuming that the package contents experience reconfiguration as a result of the accident, it is highly likely that reconfiguration will result in a lower k_{eff}. Reconfiguration may lead to slightly higher k_{eff} under extremely unlikely assumptions. However, (1) best-estimate evaluation of fuel damage under accident conditions indicates that damage is limited and far from approaching the type of damage assumed for extremely unlikely scenarios; and (2) the potential for any reconfiguration is driven by high burnup, but high burnup equals low residual nuclear reactivity.

5 PROPOSED RESOLUTION

5.1 Short-Term Resolution Path

Risk information indicates that the probability of a critical event during transportation is essentially zero. This conclusion is valid as long as k_{eff} is lower than, or equal to, 0.95 for nominal configuration [Part 71.55(b)].

5.1.1 Application to the "High-Burnup" Issue

The "criticality" issue under hypothetical accident conditions can be dismissed on the basis of the risk information considerations. The impact of this resolution path is to essentially expand the applicability of ISG-11 to transportation, and not only storage.[8] ISG-19 should be withdrawn, as it would no longer serve any useful purpose.

5.1.2 Application to Burnup Credit

It is recommended that the industry proposes an alternate and practical approach relying on tools well tested in the context of reactor operations. In addition, based on the risk information discussion, moderator exclusion (due to the low probability of transportation accidents severe enough to create an opening in the waste package) provides defense-in-depth.

5.1.3 Application to Burnup Measurement

The NRC should revise its existing regulatory guidance to provide alternatives to the requirement for in-pool measurement when it is unnecessary. Fuel assembly burnup information is already well characterized:

- Quality records -- in-core measurements – already exist;
- Comparisons show good agreement relative to diverse records (calculated burnup);
- Burnup measurements are not consistent with current regulatory practices

[8] Some fine tuning of ISG-11 may be required, more specifically in the areas of thermal cycling and guidance for low burnup fuel.

In addition, in-pool measurements have adverse impacts (fuel assembly manipulations, more personnel access to vital area, occupational exposure, low-level waste generation)

5.1.4 Application to Moderator Exclusion

There is sufficient guidance presently available for designing such systems in the US NRC and IAEA regulations. The design approach should be augmented by a defense-in-depth argument consisting in showing that the specific contents of a given transportation package would result in a k_{eff} <1 if flooded, by using a *best-estimate* burnup credit methodology.

5.1.5 Application to Loading under Part 72

There is no technical justification at this time for continuing the “fresh fuel” practice for cask loading under Part 72.

5.2 Long-Term Resolution Path

It would be highly desirable to modify Part 72 through rulemaking by adding a section specifically applicable to commercial spent nuclear fuel distinct from the present approach that essentially regulates commercial spent fuel in the same manner as for fissile materials such as enriched uranium and plutonium. This lack in differentiation results in applying the same criticality safety considerations (methodologies) to such materials as highly enriched uranium, plutonium, or commercial spent nuclear fuel. The degree of rigor implemented in the methods presently espoused by the criticality safety community is certainly justified when dealing with shipment of highly enriched uranium or plutonium, which can readily form critical configuration, but not for shipment of commercial spent nuclear fuel.

A predictable regulatory treatment of moderator exclusion should also be revisited at the Commission’s level, and possibly by rulemaking.

6 CONCLUSIONS

Within the boundary limits of the analyzed cases, the EPRI-sponsored work shows that there are no credible combinations of accident events, accident locations, and fuel misloading or reconfiguration that would result in a critical configuration during the transportation of spent nuclear fuel. For most transportation package designs, criticality during hypothetical transportation accidents should be a regulatory non-issue given the extraordinarily low probability of the concomitant occurrence of the conditions required for providing a situation conducive to criticality in the cask. The non-mechanistic criticality evaluation performed in the as-loaded or as-designed configuration can be considered the bounding case for all conditions of transportation because this hypothetical reactivity case bounds all those normal and hypothetical accident cases that can *credibly* exist for a spent fuel transportation packages.

In the U.S., the present lack of regulatory guidance for analyzing hypothetical transportation accident conditions is largely based on the paucity of mechanical property data available for spent fuel irradiated above 45 GWd/MTU. The reactivity of a spent fuel transportation package with re-configured fuel and water in the fuel cavity is of particular concern to the regulators. However, realistic and achievable configurations of nuclear fuel materials following an impact accident are more likely to result in a nuclear reactivity decrease.

Non-radiological events, while very low risk themselves, provide the over-riding level of comparative risk. These non-radiological risks are directly proportional to the number of spent fuel shipments; that is, a higher number of shipments means a higher risk of accidents and other events. To minimize the number of shipments and related risk, the number of spent fuel assemblies per shipment should be maximized. High-capacity rail casks represent the lowest risk method of transporting spent nuclear fuel, regardless of the enrichment or burnup of the fuel.

7
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A
REGULATORY CONSIDERATIONS

A.1 Transportation Regulations

A.1.1 Definitions

- *Containment system* means the assembly of components of the packaging intended to retain the radioactive material during transport.
- *Fissile material* means the radionuclides uranium-233, uranium-235, plutonium-239, and plutonium-241, or any combination of these radionuclides. Fissile material means the fissile nuclides themselves, not material containing the fissile nuclides. Unirradiated natural uranium and depleted uranium and natural uranium or depleted uranium that has been irradiated in thermal reactors only, are not included in this definition.
- *Package* means the packaging together with its radioactive contents as presented for transport.
- *Packaging* means the assembly of components necessary to ensure compliance with the packaging requirements of [Part 71]. It may consist of one or more receptacles, absorbent materials, spacing structures, thermal insulation, radiation shielding and devices for cooling or absorbing mechanical shocks. The vehicle, tie-down system, and auxiliary equipment may be designated as part of the packaging.
- *Spent nuclear fuel or Spent fuel* means fuel that has been withdrawn from a nuclear reactor following irradiation, has undergone at least 1 year's decay since being used as a source of energy in a power reactor, and has not been chemically separated into its constituent elements by reprocessing. Spent fuel includes the special nuclear material, byproduct material, source material, and other radioactive materials associated with fuel assemblies.
- *Type A quantity* means a quantity of radioactive material, the aggregate radioactivity of which does not exceed A_1 for special form radioactive material, or A_2, for normal form radioactive material, where A_1 and A_2 are given in Table A-1 of [Part 71], or may be determined by procedures described in Appendix A of [Part 71].
- *Type B quantity* means a quantity of radioactive material greater than a Type A quantity.

A.1.2 Package Approval Standards

Subpart E to Part 71 (encompassing 10 CFR 71.41 through 71.65[9]) provides the standards for approval of radioactive material packages. 10 CFR 71.41 through 71.47 provide the requirements for the design and certification of all radioactive material transportation packages. 10 CFR 71.51 through 71.65 include particular requirements for Type B packages, fissile material packages, and shipments of plutonium. Spent fuel transportation packages are both Type B and fissile material packages and they are considered plutonium shipments. Regulations applicable to Type B packages containing more than a 10^5 A2 quantity of radioactive material also apply to spent fuel packages.

This report focuses on the surface transportation of spent nuclear fuel. There are several requirements in Subpart E that are either not applicable to surface transportation of spent nuclear fuel (e.g., §71.64, "Special Requirements for Plutonium Air Shipments) or are not applicable to the form of special nuclear material contained in spent fuel (e.g., §71.55(g), "Packages containing uranium hexafluoride…). These requirements, and any others not applicable to spent fuel transportation casks, are not germane to this report, and will be listed, but not discussed in detail. In addition, regulations pertaining to package tie-down requirements will not be discussed because they are not central to the objective of this report.

A.1.3 General Standards for All Packages

10 CFR 71.43(f) requires all radioactive material packages to be designed, constructed and prepared for shipment so that under the tests specified in §71.71 ("Normal Conditions of Transport") there would be no loss or dispersal of radioactive contents, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging. The so-called "tests specified in §71.71" establish the limiting conditions for cask performance during normal transportation. These "tests" and the acceptance criteria in §71.43(f) ensure that the package will perform its design functions and meet all regulatory limits for the period of transportation, including all reasonably likely hazards to which the package could be subjected.

A.1.4 Additional Requirements for Type B Packages

10 CFR 71.51 establishes additional design requirements for Type B packages involving the release of radioactive material under normal conditions of transport (§71.71) and under hypothetical accident conditions (described in §71.73).

A.1.5 Requirements for Fissile Material Packages

10 CFR 71.55 establishes several requirements for fissile material packages under various conditions. Each subsection to this section is discussed below.

[9] Not all subsection numbers are used in the subpart.

A.1.5.1 General Requirements

10 CFR 71.55(a) provides a general statement that fissile material packages must be designed and constructed in accordance with 10 CFR 71.41 through 71.47 and, based on the amount of radioactive material in the package, it also needs to be designed and constructed in accordance with §71.51. A spent fuel cask is a fissile material package and meets the threshold for applicability of §71.51.

A.1.5.2 Moderator Intrusion

10 CFR 71.55(b) requires the package to be designed and constructed and its contents so limited such that it will remain sub-critical if water were to leak into the containment system or liquid contents were to leak out of the containment system. Historically, the containment system as considered by the designers under this part of the regulation has been the vessel containing the spent fuel. This may have been the bolted-lid bare fuel cask, which was also the containment boundary considered for compliance with the balance of the Part 71 regulations, or the welded canister, which was not considered the containment boundary for canister-based packages. In canister-based system the canisters containing the fuel are packaged in an outer, bolted-lid overpack, which is designed and analyzed as the package containment boundary. In theory, the designers and analysts could have modeled water intrusion into the outer overpack, but not the inner canister and complied with this regulation. To date, however, no designer has attempted to license such an approach for a canister-based system.

It is important to note that the requirement for moderator to be assumed to flood the containment system in the criticality analysis is neither a normal condition of transport (§71.55(d)) nor a hypothetical accident condition (§71.55(e)) that the package must be designed to withstand. It is a non-mechanistic assumption required by this regulation independent of, and in addition to the normal and accident conditions.

A.1.5.3 Moderator Exclusion

10 CFR 71.55(c) permits the Commission to approve exceptions to the moderator intrusion requirement if the shipping 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 does not leak. The NRC has made it clear that they will not generically license a package design under this exception, but would consider individual shipments, or perhaps multiple shipments of a small group of the same packages under this exception if strict, specifically licensed controls on the contents and shipping route are in place.

The so-called "moderator exclusion" rule is an artifact of a past time when all shipping packages for spent nuclear fuel were of the bare fuel design. Canister-based shipping packages are designed to preclude breach of the canister under all normal and accident conditions of transport. Nevertheless, the non-mechanistic nature of the moderator intrusion requirement and the NRC's reluctance to grant an exception on a generic, package design basis requires the designers to continue to perform the criticality analysis assuming the fuel cavity is flooded with unborated water. The Nuclear Regulatory Commission has informed the staff that rulemaking to modify the moderator intrusion requirement is not to be pursued in the foreseeable future.[10]

A.1.5.4 Normal Conditions of Transport

10 CFR 71.55(d) provides the performance requirements for fissile material packages during normal conditions of transport that are defined by the tests described in §71.71.

1. The contents must remain 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, including:
   i. No more than 5 percent reduction in the total effective volume of the packaging on which nuclear safety is assessed;
   ii. No more than 5 percent reduction in the effective spacing between the fissile contents and the outer surface of the packaging; and
   iii. No occurrence of an aperture in the outer surface of the packaging large enough to permit the entry of a 10 cm (4 in) cube.

The normal conditions of transport are defined by a series of tests the package must withstand as listed in 10 CFR 71.71(c):

1. *Heat.* An ambient temperature of 38°C (100°F) in still air and insolation.
2. *Cold.* An ambient temperature of -40°C (-40°F) in still air and shade.
3. *Reduced external pressure.* An external pressure of 25 kPa (3.5 lbf/in^2) absolute.
4. *Increased external pressure.* An external pressure of 140 kPa (20 lbf/in^2) absolute.
5. *Vibration.* Vibration normally incident to transport.
6. *Water Spray.* Water spray that simulates exposure to rainfall of approximately 5 cm/hr (2 in/hr) for at least 1 hour.
7. *Free drop.* Between 1.5 and 2.5 hours after the conclusion of the water spray test, a free drop through the distance of 0.3 meter, or 1 foot (for packages weighing over 15,000 kg (33,100 lbs)) onto a flat, essentially unyielding, horizontal surface, with the package striking the surface in a position for which maximum damage is expected.

[10] Reference: NRC, Staff Requirements – SECY-07-185 "Moderator Exclusion in Transport Packages," December 18, 2007

8. *Corner drop.* Not applicable to rail car-sized spent fuel transport
9. *Compression.* Not applicable to rail car-sized spent fuel transport packages.
10. *Penetration.* Impact of the hemispherical end of a vertical steel cylinder of 3.2 cm (1.25 in) diameter and 6 kg (13 lbs) mass, dropped from a height of 1 m (40 in) onto the exposed surface of the package that is expected to be most vulnerable to puncture. The long axis of the cylinder must be perpendicular to the package surface.

A.1.5.5 Hypothetical Accident Conditions

10 CFR 71.55(e) requires that the 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, the package would be subcritical, subject to the following assumptions:

- 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;
- 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
- There is full reflection by water on all sides, as close as is consistent with the damaged condition of the package.

The hypothetical accident conditions of are defined by a series of tests the package must withstand as listed in 10 CFR 71.73(c):

1. *Free drop.* A free drop of the specimen through a distance of 9 m (30 ft) onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which the maximum damage is expected.
2. *Crush.* Not applicable to rail car-sized spent fuel transport packages.
3. *Puncture.* A free drop of the specimen through a distance of 1 m (40 in) in a position for which maximum damage is expected, onto the upper end of a solid, vertical, cylindrical, mild steel bar mounted on an essentially unyielding, horizontal surface. The bar must be 15 cm (6 in) in diameter, with the top horizontal and its edge rounded to a radius of not more than 6 mm (0.25 in), and of a length as to cause maximum damage to the package, but not less than 20 cm (8 in) long. The long axis of the bar must be vertical.
4. *Thermal.* Exposure of the specimen fully engulfed, except for a simple support system, in a hydrocarbon fuel/air fire of sufficient extent, and in sufficiently quiescent ambient conditions, to provide an average emissivity coefficient of at least 0.9, with an average flame temperature of at least 800°C (1475°F) for a period of 30 minutes, or any other thermal test that provides the equivalent total heat input to the package and which provides a time averaged environmental temperature of 800°C. The fuel source must extend horizontally at least 1 m (40 in), but may not extend more than 3 m (10 ft), beyond any external surface of the specimen, and the specimen must be positioned 1 m (40 in) above the surface of the fuel source. For purposes of calculation, the surface absorptivity coefficient must be either that value which the package may be expected to possess if exposed to the fire specified or 0.8, whichever is greater; and the convective coefficient must be that value which may be demonstrated to exist if the package were exposed to the fire specified. Artificial cooling

may not be applied after cessation of external heat input, and any combustion of materials of construction, must be allowed to proceed until it terminates naturally.

5. *Immersion – fissile material.* For fissile material subject to §71.55, in those cases where water inleakage has not been assumed for criticality analysis, immersion under a head of water of at least 0.9 m (3 ft) in the attitude for which maximum leakage is expected. As previously discussed, spent fuel shipping cask design and their contents are not generically licensed for moderator exclusion, but may be licensed as such on a per shipment basis.
6. *Immersion – all packages.* A separate, undamaged specimen must be subjected to water pressure equivalent to immersion under a head of water of at least 15 m (50 ft). For test purposes, an external pressure of water of 150 kPa (21.7 lbf/in^2) gauge is considered to meet these conditions.

A.1.5.6 High Radioactive Content Type B Packages

10 CFR 71.61 requires packages containing more than 10^5 A2 [of radioactive material], (with A2 being the quantities of specific isotopes in the package specified in Appendix A to 10 CFR 71) to be designed for an increased level of external water pressure than other fissile material packages. Railcar-sized spent fuel shipping packages contain more than 10^5 A2 of radioactive material and are, therefore, subject to the provision of this rule. The package must be design so that its undamaged containment system can withstand an external water pressure of 2 MPa (290 psi) for a period of not less than 1 hour without collapse, buckling, or inleakage of water.

A.1.5.7 Plutonium Shipments

10 CFR 71.63 establishes one additional requirement for packages designed for shipment of more than 20 Curies of plutonium. Because spent fuel assemblies contain greater than 20 Curies of plutonium as fission products in the fuel pellet matrix, this requirement applies to spent fuel shipping casks. The requirement is simply that the contents must be in solid form. Spent fuel assemblies meet this requirement by their nature.

A.1.5.8 Assumptions as to Unknown Properties

10 CFR 71.83 states the following: "When the isotopic abundance, mass, concentration, degree of irradiation, degree of moderation, or other pertinent property data of fissile material in any package is not known, the licensee shall package the fissile material as if the unknown properties have credible values that will cause the maximum neutron multiplication." This is a general stipulation requiring designers to make conservative assumptions in the package design and analysis that maximize the computed k-effective when isotopic data are scarce or not supported by test data of sufficient quality.

A.1.5.9 Other Fissile Material Package Regulations

Several additional regulatory requirements apply to fissile material packages but do not apply to rail-car sized spent fuel packages intended exclusively for over-land or water transport, which are the subject of this report. These regulations are listed below but not summarized or discussed further in this report.

- 10 CFR 71.55(f): Fissile material package designs to be transported by air.
- 10 CFR 71.55(g): Packages containing uranium hexafluoride.
- 10 CFR 71.59: Arrays of fissile material packages. [Railcar-sized spent fuel casks are single package shipments.]
- 10 CFR 72.64: Special requirements for plutonium air shipments.
- 10 CFR 71.74: Accident conditions for air transport of plutonium.
- 10 CFR 71.75: Qualification of special form radioactive material. [Spent fuel assemblies are not special form material.]

A.2 Regulatory Guidance

A.2.1 Standard Review Plan

Chapter 6 of NUREG-1617 provides the staff with criticality evaluation review guidance for the package.

Section 6.4.3 of the SRP suggests the analysis take credit for no more than 75% of the specified minimum neutron poison concentration [in fuel basket absorber panels] and no credit should be taken for burnable absorbers. Both of these suggestions are areas of conservatism as there will be more neutron absorption occurring than that modeled in the criticality analysis.

Section 6.4.8 of the SRP delineates the staff's position circa 2000 (i.e., up to and including ISG-8, Rev. 1) on the subject of burnup credit evaluations for commercial spent fuel transportation packages. The key points of the review guidance applicable to the licensing basis analysis are summarized below.

A.2.1.1 Section 6.4.8.1 – Limits for the Licensing Basis

Guidance in this SRP section has been superseded by Revision 2 to ISG-8.

A.2.1.2 Section 6.4.8.2 – Code Validation

- Bias and uncertainties associated with predicted actinide compositions determined from benchmarks of fuel assembly measurements
- Bias and uncertainties associated with the calculation of k-effective derived from benchmark experiments representing important features of the cask design and spent fuel
- The set of nuclides used to determine the k-effective value is limited to that established in the validation process

- Particular consideration given to bias uncertainties arising from lack of critical experiments for spent fuel in a cask

A.2.1.3 Section 6.4.8.3 – Licensing Basis Model Assumptions

- Actinide compositions calculated using fuel design and in-reactor operating parameters selected to provide conservative estimated k-effective
- Account for axial and horizontal burnup profiles in the fuel in the cask, including consideration of less-burned ends of the fuel
- Consider higher actinide reactivity of fuel burned with fixed [neutron] absorbers, or with control rods fully or partially inserted

A.2.1.4 Section 6.4.8.4 – Loading Curve

- Minimum fuel cooling time of five years

A.2.1.5 Section 6.4.8.5 – Assigned Burnup Loading Value

- Administrative loading procedures include an assembly measurement that confirms the reactor record assembly burnup
- The measurement should match the reactor record within a 95% confidence interval based on measurement uncertainty to be considered “confirmed”
- The “assigned burnup value” is the confirmed reactor record value as adjusted by reducing the record value by the combined uncertainties in the records and the measurement

A.2.1.6 Section 6.4.8.6 – Estimate of Additional Reactivity Margin

- Estimate the additional reactivity margins available from fission products and actinides not included in the licensing safety basis
- The analysis methods used to estimate these reactivity margins should be verified using available experimental data (e.g., isotopic assay data) and computational benchmarks that demonstrate the performance of the applicant’s methods in comparison with independent methods and analyses
- Resulting estimates of design-specific margins should be assessed against:
  - Any uncertainties not directly evaluated in the modeling or validation processes for actinide-only burnup credit, and
  - Any potential non-conservatisms in the models for calculating the licensing basis actinide inventories

A.2.2 Interim Staff Guidances

A.2.2.1 ISG-1

The NRC issued Revision 2 of ISG-1 in May 2007.[11] This ISG provides guidance to cask designers on defining spent fuel as intact or damaged based on function. Damaged fuel would be required to be stored or transported in a damaged fuel container to ensure the geometry of the fuel in the cask remains bounded by the assumptions in the criticality analysis.

A.2.2.2 ISG-8

The NRC issued Revision 2 of ISG-8 in September 2002.[12] This ISG is applicable to actinide-only burnup credit for PWR assemblies enriched to 5.0 wt% or less. A key issue is that the assigned burnup based on reactor records must be confirmed by a measurement. Assigned burnup values are obtained from reactor records, but are adjusted by reducing the reactor record value by a combination of the uncertainties in the record value and in the measurement.

As discussed in the previous Section, the current version of the SRP for spent fuel transportation package licensing includes the revised guidance of ISG-8, Revision 1. Revision 2 of ISG-8 is the current revision. Thus, only the changes in the key technical areas specified in ISG-8, Revision 2 are summarized here.

A.2.2.2.1 Limits for the Licensing Basis

The NRC guidance continues to recommend burnup credit only for actinide compositions associated with UO_2 fuel irradiated in PWR power plants. Applicants requesting fission product burnup credit would need to justify deviating from the guidance with appropriate data for the isotopes of interest. However, some clarification and relaxation on other limits in the recommendations of the guidance is provided in ISG-8, Revision 2:

1. The upper limit for the amount of fuel burnup available to be considered is increased to 50 GWD/MTU.
2. The minimum fuel cooling time is modified from a mandatory five years to a cooling time as short as one year and as long as 40 years.
3. Fuel that was exposed to burnable absorbers in the reactor is permitted for burnup credit consideration.
4. The requirement for a 1 GWD/MTU burnup offset per 0.1 wt% increase in enrichment for fuel enriched over 4.0 wt. % ^{235}U is removed.

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[11] NRC, Interim Staff Guidance 1, Rev. 2, "*Classifying the Condition of Spent Nuclear Fuel for Interim Storage and Transportation Based on Function*," May 11, 2007.

[12] NRC-SFST Interim Staff Guidance 8, Rev. 2, "*Burnup Credit in the Criticality Safety Analyses of PWR Spent Nuclear Fuel in Transport and Storage Casks*," September 27, 2002.

A.2.2.2.2 Code Validation

Two changes in the guidance pertaining to bias and uncertainties were made:

1. Bias and uncertainties associated with the calculation of k-effective should be derived from benchmark experiments that closely [emphasis added] represent the important features of the cask design and spent fuel contents.
2. The licensing basis safety analysis should utilize bias and uncertainty values that can be justified as bounding based on the quantity and quality of the experimental data. This replaces the previous guidance that simply stated the bias and uncertainties should be applied in a way that ensures conservatism in the licensing safety analysis.

A.2.2.2.3 Licensing-Basis Model Assumptions

This section is expanded to address fuel that was irradiated in the reactor in the presence of burnable absorbers and/or full or partially inserted control rods. The guidance continues to require accounting for, and effectively modeling axial and horizontal burnup distribution based on reactor operating conditions. A reference to an Oak Ridge National Laboratory paper is added to assist the analyst in developing the fuel burnup profile. The guidance further suggests a licensing basis modeling assumption whereby the assemblies are exposed during irradiation to the maximum neutron absorber loading of burnable poison rods for the maximum assembly burnup in order to encompass all assemblies, whether or not they have been exposed to burnable absorbers. Assemblies exposed to “atypical” insertions of control rods should not be loaded unless the safety analysis explicitly considers such operational conditions.

A.2.2.2.4 Loading Curve

The guidance recommends that separate loading curves should be established “for each set of licensing conditions.” For example, separate, cooling time-specific loading curves should be developed for each cooling time considered in the cask loading to bound various fuel types [and] burnable absorber loading. The guidance further recommends that only one loading curve be used for each cask loading “[t]o limit the opportunity for misloading.”

A.2.2.2.5 Assigned Burnup Loading Value

The guidance continues to recommend that administrative procedures be established to ensure the cask will be loaded with fuel with burnup that is within the specifications of the approved contents in the package certificate of compliance. The administrative procedures should include a measurement that confirms the reactor record for each assembly with a 95 percent confidence interval. In ISG-8, Revision 2, a new provision for sampling is added. Sampling may be used if a database of measured data is provided to justify the adequacy of the procedure in comparison to procedures that require a burnup measurement of each assembly.

A.2.2.2.6 Estimate of Additional Reactivity Margin

The SRP discussion regarding performance of a design-specific analysis that estimates the additional reactivity margin available for fission products and actinides in the fuel but not credited in the licensing basis analysis is augmented. The additional guidance is a discussion about the experimental database relevant to the use of burnup credit in the safety analysis of a PWR cask not being as extensive as that for licensing analyses of unirradiated fuel. Extensive discussion of uncertainties not directly evaluated in the actinide-only burnup credit analysis is also provided.

A.2.2.3 ISG-11

The NRC issued Revision 3 of ISG-11 in November 2003.[13] Most of ISG-11 provides guidance on the thermal analysis of the cask and the temperature limits for fuel cladding under normal, off-normal and accident conditions. Importantly, no guidance is provided for high burnup fuel in transportation casks, which have to be designed to withstand more severe accident conditions (e.g., a nine meter drop) than a storage cask. The guidance states that the NRC staff is reviewing data and technical reports in an attempt to refine the guidance for transportation of high burnup fuel. Until then, transportation of high burnup fuel will be addressed by the NRC on a case-specific basis through the review of licensing submittals.

For the purpose of this report, the guidance pertaining to the prevention of fuel failure and relocation of the fuel inside the cask is important. The geometry of the fuel considered in the licensing basis criticality analysis for the cask system must bound all potential geometries of the fuel while in the cask during storage and/or transportation to provide reasonable assurance of safety to the NRC reviewer.

The limited amount of discussion about transportation of high burnup fuel in ISG-11 pertains to the concentration of hydrogen in the cladding and the size, distribution, orientation, and location of zirconium hydrides in the metal. Hydrides with the right combination of these traits may create points of reduced fracture toughness in the cladding, leading to cladding fracture under stress conditions such as those caused by hypothetical accident conditions.

A.2.2.4 ISG-19

The NRC issued ISG-19 in May 2003.[14] This ISG offers options for cask designers to address moderator intrusion into the cask containment during transportation accident conditions, based on the design features of the cask. The two options are:

1. Perform a structural evaluation to determine a bounding fuel reconfiguration based on the design of the cask, the irradiated fuel cladding properties, and the accident impact loads

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[13] NRC-SFST Interim Staff Guidance 11, Rev. 3, "*Cladding Considerations for the Transportation and Storage of Nuclear Fuel*," November 17, 2003.

[14] NRC, Interim Staff Guidance 19, "*Moderator Exclusion under Hypothetical Accident Conditions and Demonstrating Subcriticality in Spent Fuel under the Requirements of 10 CFR 71.51(e)*," May 2, 2003.

imposed on the fuel. Then perform a criticality analysis of the bounding fuel reconfiguration assuming moderator intrusion into the cask cavity.

2. Show that there will be no moderator intrusion into the fuel cavity under accident conditions based on the design features of the cask and appropriate physical testing.

In both cases, the designer must further demonstrate that the cask closure bolts do not exceed yield stress and any there is no inelastic deformation of the cask closure system under accident loadings.

B
CLADDING PERFORMANCE UNDER TRANSPORTATION ACCIDENT CONDITIONS – A SUMMARY

B.1 Background

Transportation accidents postulated for spent fuel shipments are bounded by the regulatory hypothetical accident of a 9-meter drop of a rail transport cask onto an unyielding surface. Of the three possible drop orientations of the cask at impact, namely the end-on drop, the corner drop with slap-down and the side drop; the latter is the most severe because it activates the cladding failure modes with the highest failure potential. A schematic depiction of these failure modes is shown in Figure B-1. The EPRI study simulates the effects of the hypothetical transportation accident on cladding behavior in all three modes. A summary of the results is given here; details can be found in the following reports:

1. 1009929, June 2005: "Spent Fuel Transportation Applications: Fuel Rod Failure Evaluation under Simulated Cask Side Drop Conditions."
2. 1011817, December 2005: "Spent Fuel Transportation Applications: Global Forces Acting on Spent Fuel Rods and Deformation Patterns Resulting from Transportation Accidents."
3. 1013447, October 2006: "Spent-Fuel Transportation Applications: Modeling of Spent-Fuel Rod Transverse Tearing and Rod Breakage Resulting from Transportation Accidents."
4. 1013448, December 2006: "Spent Fuel Transportation Applications: Longitudinal Tearing Resulting from Transportation Accidents – A Probabilistic Treatment."
5. 1011816, September 2005: "Application of Critical Strain Energy Density to Predicting High-Burnup Fuel Rod Failure – Response to Comments from the Nuclear Regulatory Commission Staff."
6. 1015048, December 2007: "Spent Fuel Transportation Applications – Assessment of Cladding Performance: A Synthesis Report"

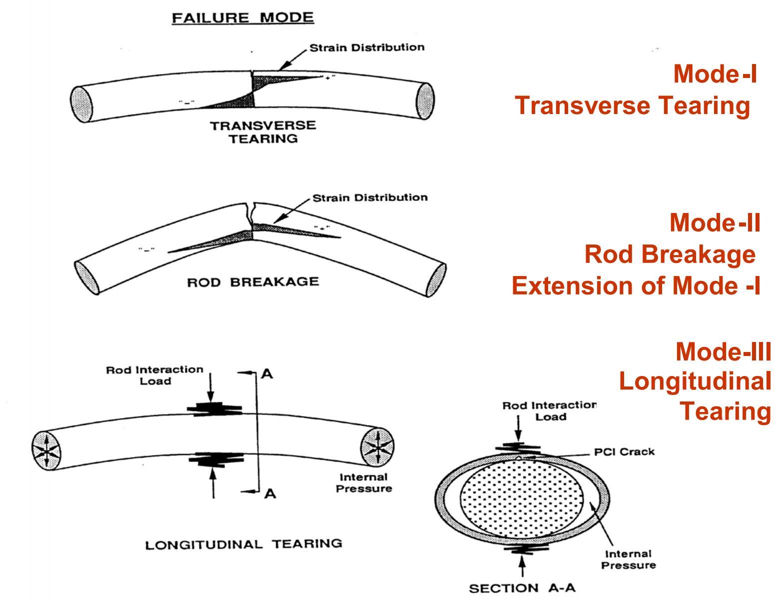

Figure B-1
Definition of Potential Failure Modes under Hypothetical Accident Conditions

B.2 Dynamic Forces Acting on Fuel Rods

Figures B-2 through B-4 show frequency distributions for the dynamic forces acting on fuel rods during a 9-m side drop of the modeled transportation package. These forces were obtained assuming the fuel rods can undergo elastic-plastic deformations without failing; consequently they constitute upper bound force distributions for calculating failure probabilities for the three modes depicted in Figure B-1. The distributions for the axial force and bending moment, which affect Modes I and II, are shown in Figures B-2 and B-3, with largest axial force of 2.8 kips, and largest bending moment of 300 in-lb, respectively. Figure B-4 shows the distribution for the pinch force, which affects Mode-III, with a maximum value of 7.5 kips. Force-time history plots for the dynamic forces in the fuel assembly experiencing the highest impact loading are shown in Figure B-5, which illustrates the impulsive nature of the loading.

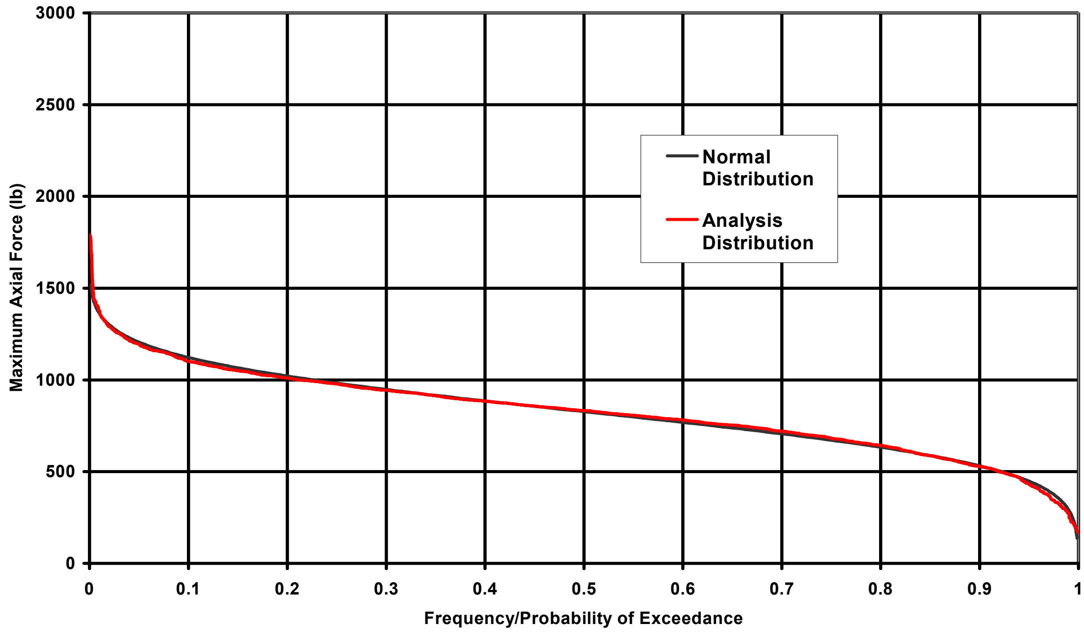

Figure B-2
Frequency Distribution of Axial Force

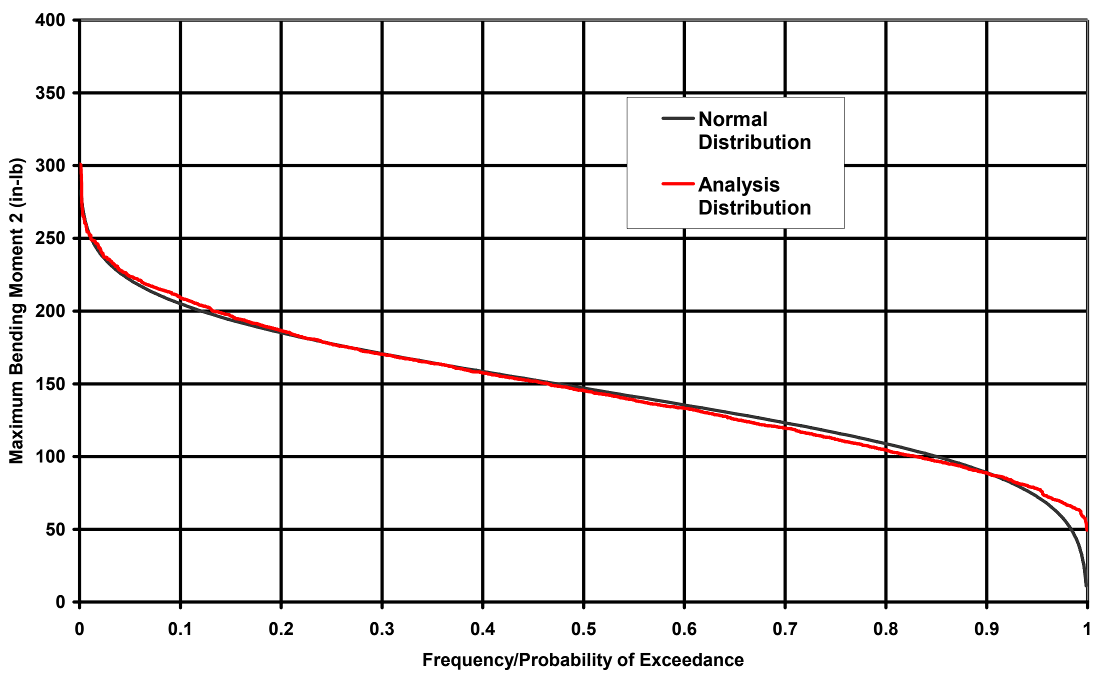

Figure B-3
Frequency Distribution of Bending Moment

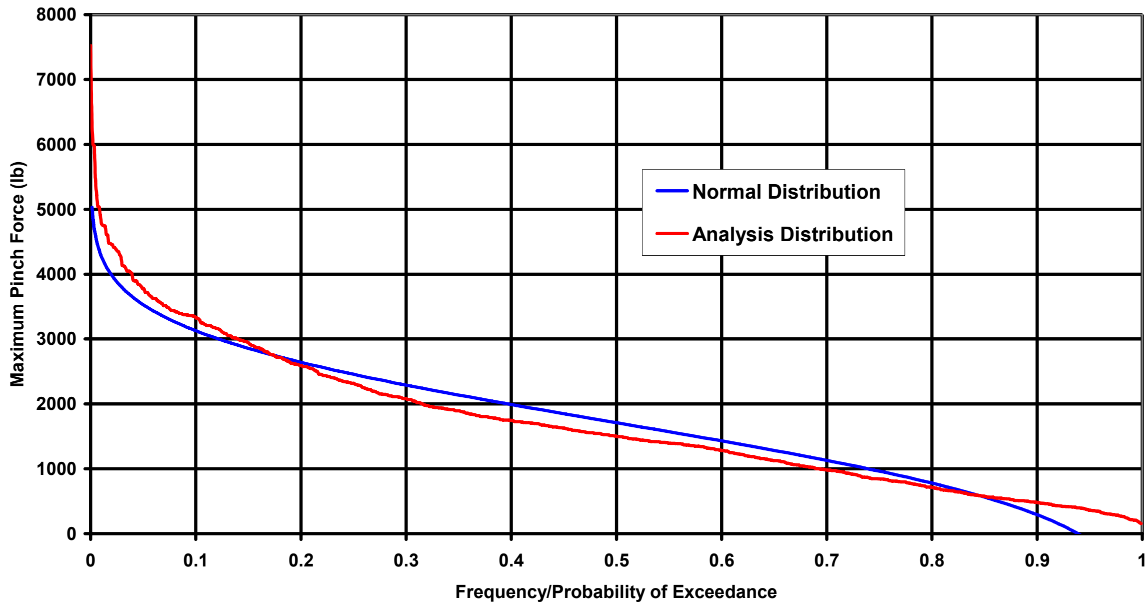

Figure B-4
Frequency Distribution of Pinch Force

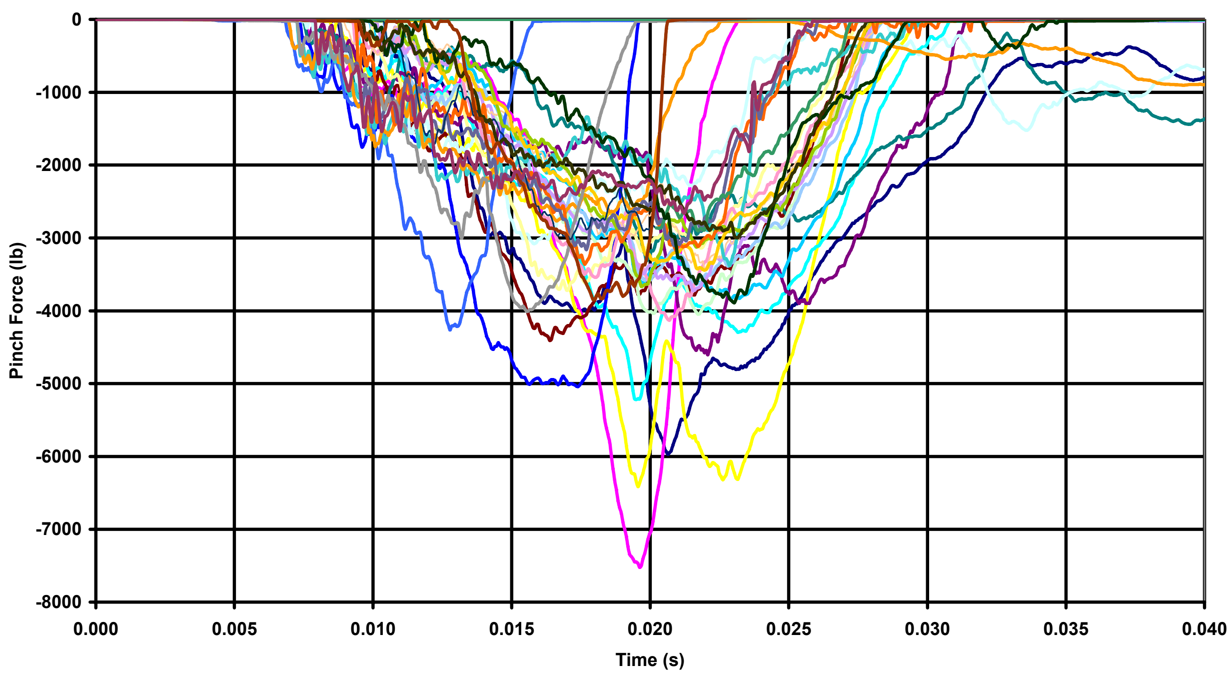

Figure B-5
Pinch Force Time Histories for a Spent Fuel Assembly Receiving the Highest Impact

B.3 Effects of End-of-Dry-Storage Conditions on Cladding Failure Behavior

There are three types of physical and material conditions of spent fuel rods at the end of dry storage that could have an effect on cladding failure behavior under transportation accident conditions. These are:

(a) Burnup-dependent conditions, such as the dependence of cladding mechanical properties on irradiation, and cladding thickness loss due to the OD corrosion layer, which affect the magnitude of cladding deformations.

(b) Dry-storage conditions, such as creep-induced fuel-cladding gap and hydrides re-orientation, which affect cladding resistance and vulnerability to failure.

(c) Cladding defects, such as OD hydride lens and incipient ID cracks, behave as precursors for cladding failure initiation.

These are incorporated in the EPRI methodology, and their effects are reflected in the results. The results indicate that type (a) conditions play an indirect role in cladding failure behavior, namely, through their effects on cladding deformations. Type (b) conditions, however, play a very direct role in cladding failure behavior in two ways: firstly, through the effects of radial hydrides on cladding fracture resistance, and, secondly, through the effect of the fuel-cladding gap size on limiting cladding deformations due to fuel pellets participation in resisting the load. This latter effect of gap size plays a similar role in the behavior of type (c) defects, where cladding contact with the fuel pellets prevents the propagation of ID cracks or surface defects to through-wall failures. This behavior is illustrated in Figure B-6, which depicts the relative distribution of the Strain Energy Density (SED), (a measure of the material's response to the loading), showing crack-tip local concentration of SED. This figure illustrates the inhibiting effect of pellet-cladding contact on potential tearing emanating from ID flaws.

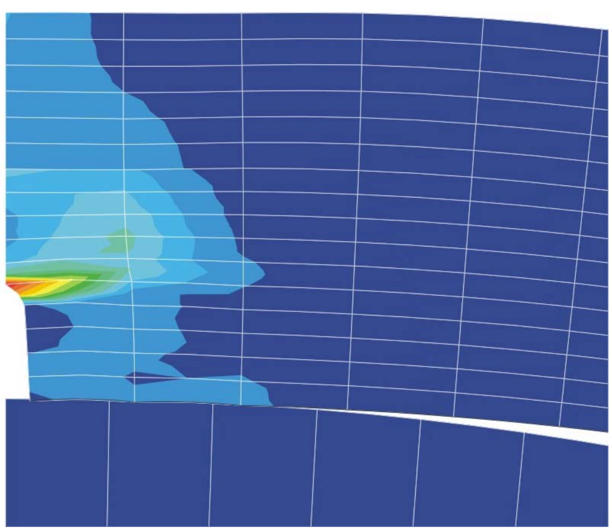

Figure B-6
Effect of Pellet-Cladding Contact under Pinch Load on Local Strain Energy Density Distribution in the Vicinity of a PCI Crack Tip

B.4 Cladding Failure Probabilities

Cladding sensitivity to failure depends to a large extent on the direction of the applied stress relative to the hydrides' orientation in the cladding. For example, Mode-III, which is activated by the pinch force, is highly sensitive to the presence of small concentration levels of radial hydrides, whereas radial hydrides have almost no effect on Modes I and II. This is because the applied stress in Mode-III is normal to the plane of the hydrides, which act as embedded brittle barriers to the stress flow. By contrast, the applied stress for Modes I and II is in the plane of the hydrides, in which case the hydrides act as interstitials and the stress flows around them in the ductile material. The results for all three modes are individually described below.

B.4.1 Longitudinal Tearing – Mode-III

Cladding failure in the longitudinal tearing mode, Mode-III, was found to be bimodal: a state of failure initiation at the cladding ID remaining as part-wall damage, with less than 2% probability of occurrence; and a through-wall failure with a probability of 1E-5. This bi-modal behavior is strongly dependent on the magnitude of cladding displacement before fuel-cladding contact begins to transfer the force to the pellets. This behavior is illustrated in Figure B-7, which depicts the strain energy density (SED) as function of the pinch force for various pellet-cladding gaps. The figure shows nearly unbounded behavior (solid-dashed curve) prior to gap closure, and a self-limiting behavior after gap closure. It is interesting to note that the magnitude of the pinch force is bounded by a value of ~4000 lbs regardless of gap size. Comparing this value to the distribution in Figure 4 indicates that, while as much as 96% of the rods could potentially experience such a force, only 0.001% of those rods would experience through-wall failure. This relatively benign accident consequence is primarily due to fuel pellets participation.

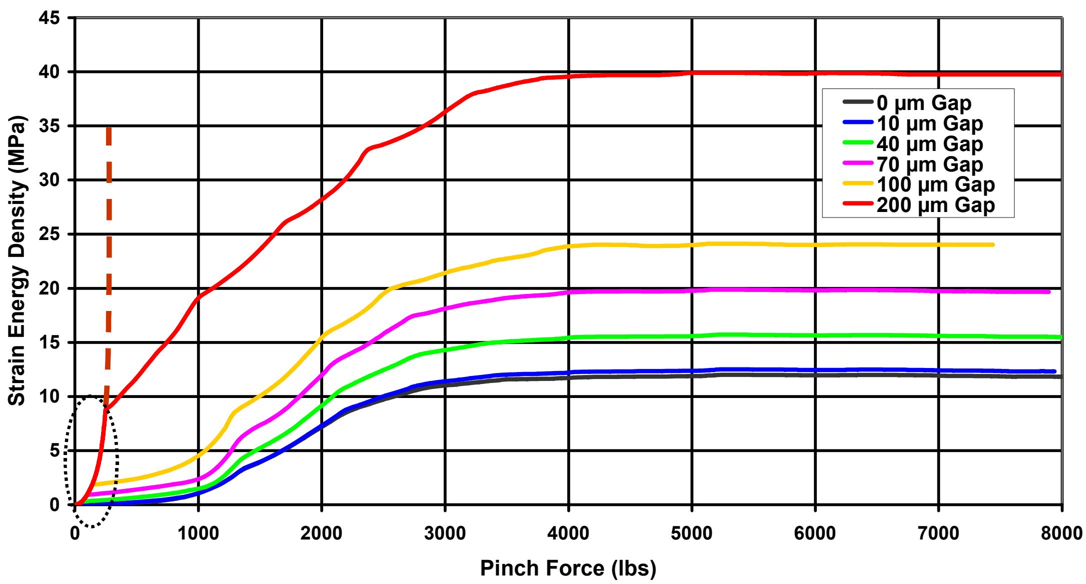

Figure B-7
SED at Cladding ID as a Function of Pinch Force – Note that the solid-dashed curve is an indirect simulation of a ring compression test.

B.4.2 Transverse Tearing and Rod Breakage – Modes I and II

These two modes are interdependent, with Mode-I implying a pinhole failure and Mode-II indicating an intermediate level of damage between a pinhole and a guillotine break. The axial force and bending moment distributions depicted in Figures B-2 and B-3 constitute bounding values because the analysis did not account for cladding failure during the event. However, by allowing damage to occur during the event, the actual magnitudes of axial force and bending moment that the fuel rod can attract are shown in Figures B-8 and B-9. As can be seen in these figures, the force and moment capacities of the rod are 1200 lb and 150 in-lb, respectively. It is important to note that the drop of the axial force to zero in Figure B-8 does not mean that the rod has been severed, but rather that a state of pure bending is reached, as shown by the nearly constant moment line in Figure B-9. The peak value in these figures signifies the initiation of pinhole Mode-I failure.

Comparing the axial force and bending moment capacities in Figures B-8 and B-9 to their corresponding distributions in Figures B-2 and B-3, respectively, indicates that about 7% of the rods may experience an axial force of 1200 lbs or greater, and about 50% of the rods may experience a bending moment of 150 in-lb or greater. If we conservatively assume that of the 50% of the rods that exceed their force capacity, 7% of those rods would exceed their moment capacity at exactly the same time, then, without rigorous analysis, an estimate of a failure probability of 3.5% can be obtained for Mode-I/Mode-II failure configuration. Moreover, the fact that an equilibrium state was reached at the end of the damage process with about 45% of the cladding remaining intact, as illustrated in Figure B-10, indicates that fuel re-configuration is not an expected outcome of the hypothetical transportation accident.

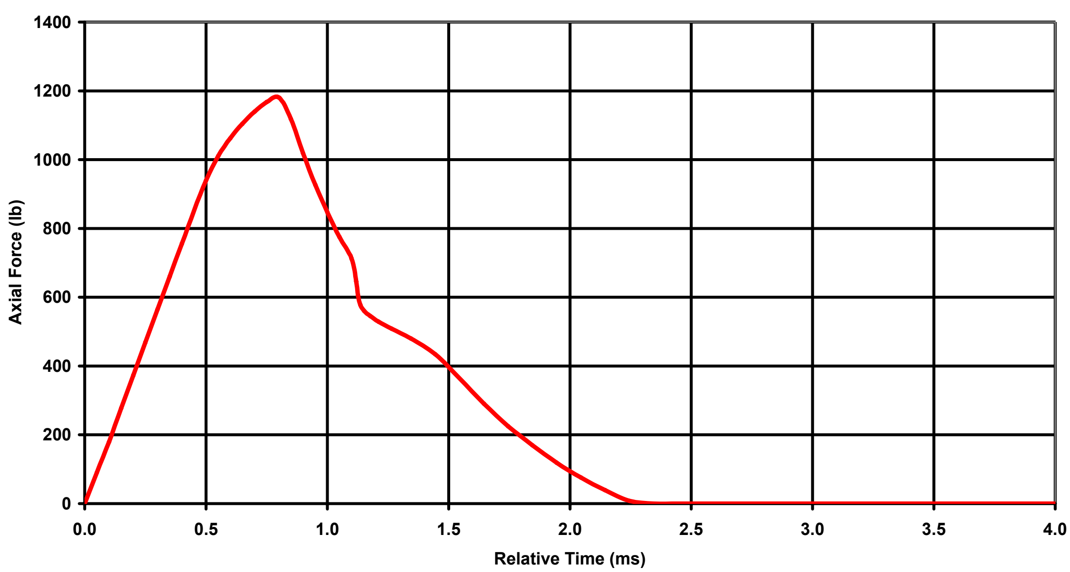

Figure B-8
Time History of the Axial Force that the Fuel Rod Was Able to Attract, when Combined with the Bending Moment Shown in Figure B-9

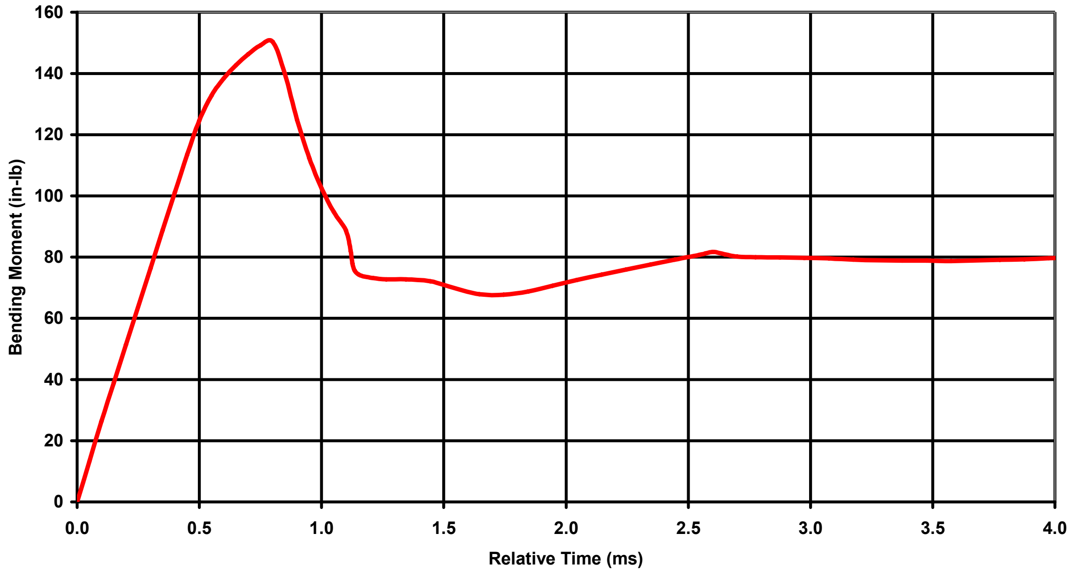

Figure B-9
Time History of the Bending Moment that the Fuel Rod Was Able to Attract, when Combined with the Axial Force Shown in Figure B-8

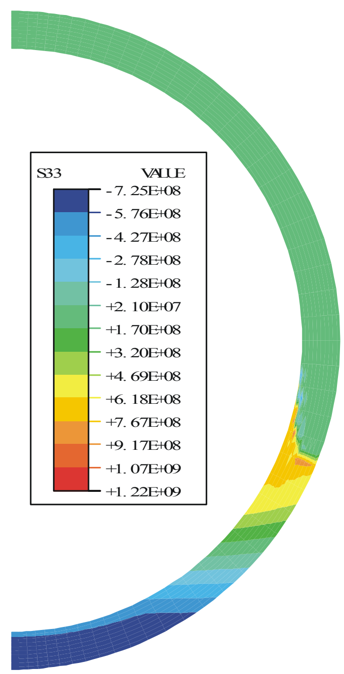

Figure B-10
Stress Distribution at 4.0-ms Relative Time at the State of Equilibrium at the End of the Analysis Sequence when Damage Progression Stops – Note that the Zero-stress Contour line (Green Color) Indicates the Extent of Fracture

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