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CONSEQUENCE ANALYSIS OF SPENT NUCLEAR FUEL RECONFIGURATION SCENARIOS

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ABSTRACT

High-burnup fuel has different characteristics than low-burnup fuel with respect to cladding oxide thickness and hydride content, radionuclide inventory and distribution, heat load, fuel grain size, fuel fragmentation, and fission gas release to the rod plenum. High-burnup fuel may have a greater potential to reconfigure than low-burnup fuel because of higher irradiation damage. In addition, it appears that the fuel will be stored in a dry storage condition beyond the initial license. As the fuel decay heat decreases during storage times, fuel rod mechanical performance may be affected as a result of cladding material properties changes associated with hydride reorientation. The fuel cladding may become less ductile once the temperature decreases below the cladding's ductile-to-brittle transition temperature.

To improve understanding of the implications of potential fuel failure on the continued safety of storage casks and transportation packages, the U.S. Nuclear Regulatory Commission (NRC) has initiated a project with Oak Ridge National Laboratory (ORNL) to evaluate the potential consequences of fuel rod failure and reconfiguration with respect to satisfying the regulatory requirements for SNF storage and transportation systems in the areas of criticality, shielding (dose rates), containment, and thermal. The technical approach includes (1) development of credible failed fuel configurations and (2) evaluation of the impact of fuel failure/reconfiguration on the storage and transportation systems with respect to satisfying the 10 CFR 71 and 72 regulatory requirements for criticality safety, shielding, containment, and thermal design. The evaluated scenarios include possible effects from individual rod break, general cladding failures, rod/assembly deformation, and gross failure as a function of fuel type, burnup, and decay time.

INTRODUCTION

Commercial spent nuclear fuel (SNF) in the United States is now regularly irradiated to high-burnup values (>45 GWd/MTU), and is expected to remain in storage for periods beyond 20 years. Very little data is available to characterize the cladding and fuel mechanical properties and aging phenomena for longer storage times and higher fuel burnup, and this results in uncertainty in fuel performance analysis of SNF in storage and transportation conditions. Traditional SNF storage and transportation package design certifications assume that the fuel remains in its original geometric configuration under storage and normal transport conditions. However, this assumption may no longer be assured for high-burnup fuel or fuel that has been in storage beyond the certification period. Therefore, the possibility of fuel failure during normal, off-normal and accident conditions of spent fuel storage and normal and accident conditions of transportation should be considered. Depending on the severity, fuel failure can result in changes to the geometric configuration of the fuel, and this can have safety and regulatory implications. This paper discusses potential consequences of fuel rod failure and fuel reconfiguration with respect to the safety of SNF storage and transportation systems in four technical disciplines: criticality, shielding (dose rates), containment, and thermal.

To evaluate the potential regulatory implications of fuel rod failure and fuel reconfiguration that can result from normal, off-normal and accident conditions of spent fuel storage and normal and accident conditions of transportation, three reconfiguration categories were considered: (1) cladding failure; (2) rod/assembly deformation without cladding failure; and (3) changes to assembly axial alignment without cladding failure. The likelihood of any particular reconfiguration and the impact of reconfiguration on each technical discipline are dependent on many factors, including storage and transportation conditions, the fuel assembly characteristics, and the storage and/or transportation system characteristics. Predicting the likelihood of reconfiguration is outside the scope of this work. Attention was placed on identifying fuel reconfigurations that may be caused by initiating event(s) (e.g., side/horizontal drop), rather than developing hypothetical worst case scenarios. Hence, a series of schematic analyses were used to characterize the impact of potential fuel reconfigurations in each technical discipline. The stylized analysis treated parameters of each reconfiguration category that would have the most significant implications in each technical discipline. The effects of parameter variations over a wide range were modeled and evaluated for each reconfiguration to observe the sensitivity of the parameter to system safety function response. Consequences of reconfiguration in each technical discipline were assessed relative to the nominal intact configuration reference cases.

RECONFIGURATION CATEGORIES

Category 1 – Cladding Failure. The structural integrity of the cladding for high-burnup fuel may become challenged under a variety of cladding degradation mechanisms. Several mechanisms that could lead to degradation of cladding structural integrity are driven by mechanical property changes due to hydrogen-related phenomena. For example, hydride reorientation can result in degradation of the ductility of the cladding, potentially resulting in radial-hydride-induced embrittlement. The temperature at which embrittlement occurs is referred to as the ductile-to-brittle transition temperature (DBTT) [1], which varies by cladding material type, and may be dependent upon a number of parameters such as burnup, irradiation history, and drying-storage history (e.g., stress at maximum temperature).

If the cladding temperature drops below the DBTT, the SNF rods are more susceptible to failure under load impacts. Load impacts from natural phenomena such as earthquakes or tornados resulting in cask tip-over, and drops can result in cladding breaches and fuel particle relocation. Additionally, when the cladding temperature is below the DBTT, fuel rod cladding failure within the package may occur under normal conditions of transport (NCT) as prescribed in 10 CFR 71.71, and as a result of impacts from hypothetical accident conditions (HAC) as specified in 10 CFR 71.73. Predicting the exact behavior of the fuel assemblies in a package can be a substantive challenge because of the uncertainty and variability of fuel rod material properties (note that this is the subject of on-going research, i.e., to better predict cladding performance). Hence, different degrees of fuel rod cladding failure followed by release of material into the canister cavity during normal handling and transfer operations are being evaluated to understand the implications of cladding failure with respect to the different technical disciplines.

Considerations for the cladding failure category were designed to represent the effects of two scenarios – (S1a) breached spent fuel rods, where the cladding has failed to the extent to allow for a loss of gas and fuel particles from single or multiple locations such that the rod segment and fuel fragments collect at different locations within the canister, and (S1b) damaged SNF, where the cladding has failed to the extent to allow free movement of fuel particles and pellets within a basket cell. The parameters for each scenario considered to have the most significant implications in each technical discipline are listed in Table 1.

Table 1. Parameters considered for Reconfiguration Category 1

Technical discipline	Scenario	
	S1a – Breached spent fuel rods	S1b – Damaged SNF
Criticality	Lattice positions where fuel particulate could be displaced from	Geometry changes and modeling homogenous versus heterogeneous representations of fuel debris mixture
Shielding	Fraction of fuel redistributed and canister basket cavity regions where particulate accumulates	Regions within canister volume where fuel is redistributed to
Containment	Fraction of failed fuel rods; in addition for high-burnup fuel, varying release fractions for the contributors to the releasable activity and pellet region from which the radioactive material originates	For high-burnup fuel, varying release fractions for the contributors to the releasable activity and pellet region from which the radioactive material originates
Thermal	Fraction of fuel rods experiencing cladding failure that releases fission product and rod backfill gases (varied from 0-100%)	The number of assemblies (1 or 32 (all)) and the packing fraction of the debris (0.612-0.313) to investigate the impact of fuel redistribution on component temperatures

Category 2 - Rod/Assembly Deformation Without Cladding Failure. A number of studies and tests have been performed over the years to investigate the impact of NCT and HAC on fuel assemblies that are contained within a transportation package [2, 3, 4]. Overall, the analyses and tests have indicated that during horizontal (side) drop the fuel rods are primarily subjected to loads that can result in fuel rod bending or some degree of plastic deformation so that the lattice pitch of the fuel assemblies tends to reduce. In a vertical (end) drop orientation, the axial loading can lead to buckling of the fuel rods.

Effects that can influence material strength and structural integrity of the cladding and fuel assembly include neutron fluence (e.g., grid spring relaxation, irradiation hardening, growth, cladding creep down), corrosion (e.g., thinning, oxidation, hydrogen uptake), operating conditions (e.g., temperature), and drying conditions (e.g., temperature, residual moisture). Analytical methods for calculating load responses and characteristics of fuel rods during and after impact events require assumptions that are difficult to verify, such as percentage of fuel mass that is bonded to or participates with the cladding during the buckling process as discussed in Ref. [5]. Additionally, boiling water reactor (BWR) and pressurized water reactor (PWR) fuel assemblies are designed differently, and some of these differences result in different mechanical responses on the fuel rods under impact events. Besides the BWR fuel assembly being channeled, the fuel rods are connected to an upper and lower tie plate. In the PWR fuel assembly, the fuel rods are not directly connected to the upper and lower end-fitting leaving a small gap between the ends of the rods and the end-fittings. Under horizontal drop events, this design difference does not result in noticeable differences on fuel rod response between a PWR and BWR fuel assembly; however, it can alter the response under vertical drop events [3].

Analyses in this reconfiguration category evaluate the impact of fuel rod and assembly deformation when the fuel cladding is able to absorb the loads of the impact event and remain intact (i.e., cladding does not fail). All reconfigurations that involve cladding failure are evaluated in the cladding failure category. Two reconfiguration scenarios are considered: S2(a) – configurations associated with side/horizontal drop; and S2(b) – configurations associated with end/vertical drop. The parameters for each scenario considered to have the most significant implications in each technical discipline are listed in Table 2.

Table 2. Parameters considered for Reconfiguration Category 2

Technical discipline	**Scenario**	
	S2a – Side/horizontal drop	**S2b – End/vertical drop**
Criticality	Assembly pin pitch contraction	Uniform and non-uniform radial and axial pin pitch changes
Shielding	Source/fuel location	N/A – Bounded by Category 3
Containment	Fraction of crud that spalls off cladding (varied from 0.05 to 1.0)	N/A – Same as Scenario S2a
Thermal	Assembly pin pitch contraction	Assembly pin pitch expansion

Category 3 - Changes To Assembly Axial Alignment Without Cladding Failure. Different types of overpacks are used for storage and transportation. Storage systems typically consist of a thick storage overpack made of steel, concrete, or a combination of the two that is used to fully encompass the spent fuel canister. For transportation, the overpack typically consists of a layered shell with several different materials to provide shielding for gamma rays and neutrons, as well as to provide a means for heat removal. The safety features of the canister fuel basket and the transportation overpack (e.g., neutron absorber plates, gamma and neutron shield, cooling fins) have typically been designed based on the expectation that the fuel assembly remains in a fixed geometric location within the package under normal and accident conditions of transportation. For example, the axial extent of the package radial neutron shielding does not always extend the full length of the containment vessel because the package needs an allowance for attaching the impact limiters on the ends. Hence, the cavity volume of the canister may not be fully covered with shielding material. Additionally, most neutron absorbers present in currently deployed systems are in plate form, do not extend the full length of the basket, and are held in place by a thin gauge stainless steel sheath. Fuel spacers are used to restrain the fuel assembly within the basket cell to ensure axial alignment of the active fuel region within the neutron absorber envelope. A schematic representation of a fuel assembly within a basket cell is illustrated in Figure 1. Some of the more recent basket designs made of metal matrix composites integrate the absorber into the basket material, in which case the absorber does extend the full length of the basket [6].

While the degree of axial movement should be limited, post-buckling bending deformations, regions of lattice expansion, and interaction with deformed nozzles can result in changes to the axial alignment after an impact event, resulting in a loss of axial geometry control from the as-designed configurations. Additionally, residual moisture that may be present in the canister after drying can promote corrosion and/or stress corrosion cracking of fuel assembly hardware components while the SNF is in dry storage. Intergranular stress-corrosion cracking (IGSCC) is a known failure mechanism that can result in dislocation of the top nozzle end-fitting from the remainder of the assembly [7], leaving space for fuel rod axial shifting.

Analyses in this reconfiguration category evaluate the impact of changes to assembly axial alignment assuming that the fuel cladding remains intact (i.e., cladding did not fail). All reconfigurations that involve cladding failure are evaluated in the cladding failure category. The parameters considered to have the most significant implications in each technical discipline are listed in Table 3.

Table 3. Parameters considered for Reconfiguration Category 3

Technical discipline	Scenario S3
Criticality	Fuel assembly axial position (varied between canister base plate and top lid)
Shielding	Fuel assembly axial position (source shifted towards top lid or towards bottom lid)
Containment	N/A (same as Scenario S2a where fraction of crud that spalls off cladding is varied)
Thermal	Fuel assembly axial position

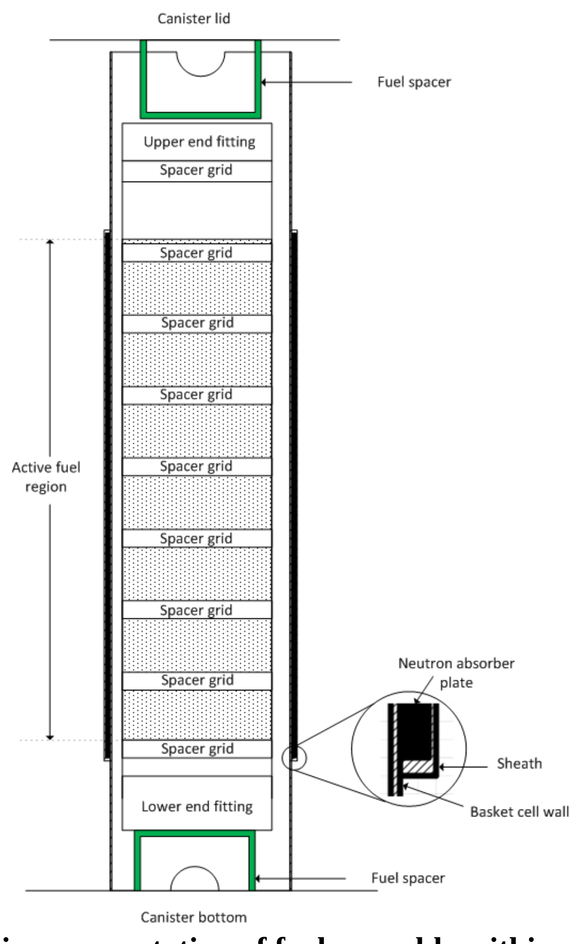
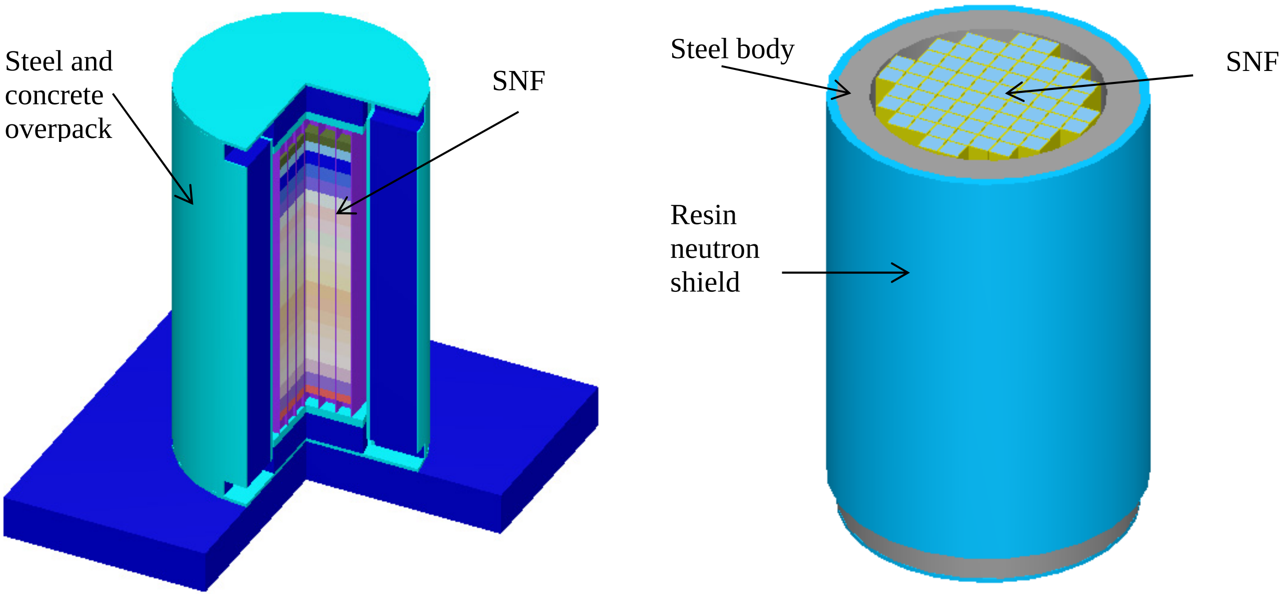


Figure 1. Schematic representation of fuel assembly within a typical basket cell

RECONFIGURATION ANALYSIS

Generic PWR and BWR storage cask/transportation package models used in previous studies [8, 9] were adapted for use in the analyses. The PWR canister/cask contains 32 PWR fuel assemblies representative of a Westinghouse (W) 17×17 optimized fuel assembly (OFA) design and the BWR canister/cask contains 68 BWR fuel assemblies representative of a 10×10 General Electric-14 (GE14) design. The PWR and BWR storage cask/ transportation package models are referred to as generic burnup credit (GBC)-32 and GBC-68, respectively. The GBC-32 and GBC-68 cask/package models were originally developed for criticality safety analyses of PWR and BWR SNF, respectively, and needed to be modified to facilitate analyses for the different reconfiguration categories. Example modifications are shown in Figure 2 that included adding a concrete overpack for the vertical storage cask configuration and radial neutron shielding for the transportation package configuration. Detailed descriptions of the modified models used for the analyses are provided in a NUREG/CR report that will be issued in the future [10].



(a) Storage cask model (b) Transportation package model

Figure 2. Example modified GBC-32 and GBC-68 models

The SCALE code system [11] was used to develop the irradiated fuel compositions and the thermal and radiation source terms. The KENO V.a and KENO-VI Monte Carlo codes were used for the criticality calculations within the appropriate CSAS5 and CSAS6 sequences with the 238-group neutron data library based on Evaluated Nuclear Data Files, Part B (ENDF/B)-VII.0, distributed with the SCALE system. The SCALE shielding analysis sequence MAVRIC (Monaco with Automated Variance Reduction using Importance Calculations) (Ref. [11], Sect. S06) and the SCALE 27N-19G ENDF/B-VII shielding library were used to perform Monte Carlo transport and dose rate calculations. The containment analysis results were developed in accordance with the calculation methodology described in NUREG/CR-6487 [12] and the containment acceptance criteria provided in 10 CFR Part 71. The thermal analyses used the thermal hydraulic analysis code COolant Boiling in Rod Arrays–Spent Fuel Storage (COBRA-SFS) [13]. Several different initial fuel enrichments and discharge burnup values and decay times were considered in the analyses. The ^{60}Co activation sources in the assembly hardware region, and typical PWR and BWR axial burnup profiles were accounted for in the models for source and isotopic composition distributions. Specific cases evaluated for each category were tailored with respect to each technical discipline (i.e., criticality, shielding, containment, and thermal) to reflect the extent of considerations governed by the requirements of the separate nuclear safety analyses. Consequences of reconfiguration in each technical discipline were assessed relative to the nominal intact configuration cases. The following is a brief summary of some the consequences observed to date regarding fuel failure/reconfiguration for the different technical disciplines. A more thorough discussion and details of the cases evaluated will be available when the NUREG/CR report is published [10].

Criticality. A summary of the criticality analysis results is provided in Table 4. Configurations associated with cladding failure (Category 1) had the most impact on the calculated results relative to the nominal intact configurations. Damaged fuel configurations where the fuel is considered completely rubblized provided the largest amount of variability for modeling (e.g., hydrogen to fissile mass ratio) and resulted in the largest consequences when optimum moderation conditions were represented. However, this type of model is considered non-credible under NCT as prescribed in 10 CFR 71.71, or as a result of impacts from accident loads specified in 10 CFR 71.73, but is being used as a bounding representation designed to evaluate the maximum change in reactivity possible. Combination models of fuel rod failure and displaced fuel distributions at or beyond the ends of the assembly provided similar changes in k_{eff} (~5% Δk_{eff}) as were observed for the reconfigurations in Category 2 that resulted in the highest k_{eff} changes.

Table 4. Summary of criticality analysis results

Scenario	Case description	Parameter varied	Maximum k_{eff} increase (% Δk_{eff}) (GBC-32/GBC-68)
Category 1: Cladding Failure			
S1(a) – breached spent fuel rods	Multiple rod removal	Multiple missing rod combinations until an upper limit is identified	1.86/2.40
	Combination of multiple rod removal and rubble extended beyond absorber envelope (displaced fuel volume fraction=0.341)	Number of missing rods and distribution of displaced fuel outside neutron absorber envelope	4.91[b]/NC[c]
S1(b) – damaged SNF	Uniform pellet array[a] distributed throughout basket cell	The pellet spacing, and thus the debris bed size, was varied to find the largest Δk_{eff}	21.37/34.40[b]
Category 2: Rod/Assembly Deformation			
S2(a) – configurations associated with side drop	Pin pitch compression	Pin pitch reduced from nominal intact configuration	Not applicable. Nominal intact configuration bounding for this reconfiguration
S2(b) – configurations associated with end drop	Uniform pin pitch expansion	Pitch of all rods expanded the same amount until outer pins are in contact with basket cell wall	2.65/2.09(channeled), 13.22 (unchanneled)
	Non-uniform radial pin pitch expansion	Pin pitch of different radial regions expanded	3.90/2.80 (channeled), 13.31(unchanneled)
	Non-uniform axial pin pitch expansion (birdcaging)	Pin pitch of different axial regions expanded	3.90/13.02 (unchanneled)
Category 3: Changes to Assembly Axial Alignment			
S3 – axial displacement of intact fuel	Assembly shift exposing active fuel outside neutron absorber envelope	Length of active fuel above or below the neutron absorber plate	3.64 at 20 cm / 6.29 at 20 cm

[a]Case is bounding but considered non-credible as the model represents the fuel floating in water in an ordered array at near optimum moderation.
[b]Maximum value from cases evaluated but optimum missing fuel and volume fraction distribution was not determined.
[c]NC: not calculated.

The effects of axial alignment shift of intact fuel assemblies (Category 3) are further illustrated in Figure 3. The cases evaluated indicated that larger reactivity changes can occur for higher burnups and longer cooling times than for low burnup fuel and short cooling times. Overall the results indicate that displacement towards the canister bottom is inconsequential, and displacement towards the top increases with increased active fuel exposure beyond the basket neutron absorber envelope. The difference in k_{eff} response between axial displacement towards the top or bottom is primarily a result of the spent fuel fission density being driven by the top of the SNF assembly due to the axial distribution of fuel burnup during irradiation resulting in higher plutonium generation and lower burnup at the top of the active fuel length [14]. Under most credible scenarios the available displacement distance will be limited due to the presence of assembly end-fitting components and fuel spacers.

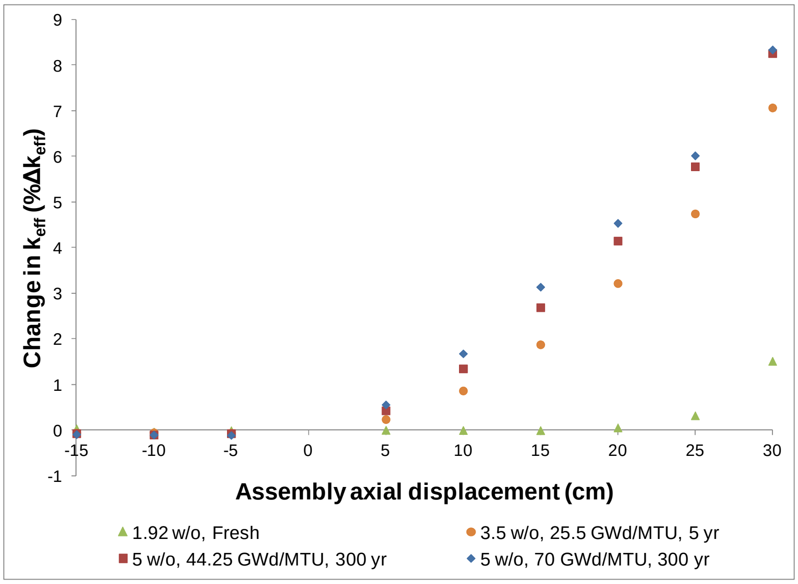


Figure 3. Change in k_{eff} in GBC-32 as a function of assembly axial displacement

Shielding. A summary of shielding analysis results are provided in Table 5 for a radiation source term for 5.0 wt % ^{235}U initial enriched fuel at a burnup of 65 GWd/MTU and 40-year decay time. Similar to the criticality evaluations, configurations associated with cladding failure (Category 1) had the most impact on the calculated results relative to the nominal intact configurations. The shielding analysis showed that fuel redistribution within the axial middle section of the fuel basket cause relatively small changes in the maximum dose rates at the cask external surfaces relative to the nominal intact fuel configuration. However, fuel redistribution towards the cask bottom and/or top regions significantly increases the dose rates at the cask/package surfaces. The neutron and gamma dose rates were also calculated at the controlled area boundary 100 meters from a generic 4×2 PWR storage cask array. The controlled area boundary evaluations indicated that when the fuel mixture is represented as homogeneously distributed within the entire canister cavity, the site boundary dose rate increased by a factor of ~2.4 for gamma radiation and by a factor of ~2.7 for neutron radiation, relative to the nominal intact configuration. However, this type of source distribution is considered non-credible under normal, off-normal, and accident conditions of storage, but was used as a bounding representation to maximize the impact on dose.

Table 5. Summary of shielding analysis results

Scenario	Case description	Parameter varied	Relative change in maximum dose rate[a]	
Category 1: Cladding Failure				
S1(a) – breached spent fuel rods	Transportation package; combination of multiple rod failure and source relocation maintained within the active fuel region, distributed to the top end-fitting, or distributed to the bottom end-fitting	Number of missing rods and distribution of displaced fuel particulates at middle of active fuel region	Insignificant	
		Number of missing rods and distribution of displaced fuel particulates to top or bottom of assembly	PWR Top: 6.2(n) 11.6 (g); Radial: 3.5(n); Bottom: 2.8(n) 5.2 (g)	BWR Top: 21.8(n) 84.6 (g); Radial: 2.4(n); Bottom: 4.4(n) 24.4 (g)
	One meter from a storage cask; multiple rod failure and source relocation distributed to the bottom end-fitting	Source distribution	PWR[b] Radial: 1.7 (n) 2.9 (g)	BWR[b] Radial: 2.2 (n) 2.9 (g)
S1(b) – damaged SNF	Transportation package; homogeneous fuel mixture distribution settled at bottom or uniformly distributed throughout the package cavity	Source distribution	PWR Top: 6.7(n) 14.1 (g); Radial: 3.9(n) 1.2 (g); Bottom: 4.2(n) 7.3 (g)	BWR Top: 23.5(n) 84.2 (g); Radial: 3.3(n) 1.4 (g); Bottom: 6(n) 32.2 (g)
	One meter from a storage cask; homogeneous fuel mixture distribution settled at bottom	Source distribution	PWR[b] Radial: 2.7 (n) 4.2 (g)	BWR[b] Radial: 3.2 (n) 9.6 (g)
	4×2 storage array evaluation at controlled area boundary	Source distribution	2.4 (g); 2.7 (n)	
Category 2: Rod/Assembly Deformation				
S2(a) – configurations associated with side drop	Pin pitch compression with fuel rods collapsed against fuel basket plates	Source distribution within basket cell	~20% (radial), ~50% (axial neutron dose rate)	
S2(b) – configurations associated with end drop	None. This case is bounded by Category 3 representations	None.	NC[c]	
Category 3: Changes to Assembly Axial Alignment				
S3 – axial displacement of intact fuel	Transportation package; assembly shift allowing fuel assemblies to reach top or bottom surface of the cask cavity	Source location	PWR radial: 30% (g) 15% (n); Bottom: 40% (n) 80% (g); Top: 30%(n) 70%(g)	BWR radial: No change; Bottom: 20% (n) 55% (g); Top: 15%(n) 40%(g)
	One meter from a storage cask; assembly shift allowing fuel assemblies to reach bottom surface of the inner cavity	Source location	PWR[b] radial: 1.4 (n) 2.7 (g)	BWR[b] radial: 1.2 (n) 1.2 (g)

[a](n) indicates neutron dose rate, (g) indicates gamma dose rate.
[b]Locations that receive radiation streaming through cask air vents.
[c]NC: Not calculated.

For Category 2 configurations, calculation results indicate that a collapsed fuel lattice causes a relatively small increase in the radial neutron and gamma dose rates relative to the nominal intact fuel configuration for both PWR and BWR SNF assemblies. For the specific models used in this study, the maximum radial dose rate increase was ~20%. Note that the collapsed fuel lattice can also cause an increase in neutron dose rate at the package bottom and top surfaces as a result of neutron streaming.

The effects of axial alignment shift (Category 3) are dependent on the available volume above or below the fuel assembly and result in increases in the neutron and gamma dose rates at the package axial surface in the direction of the axial shift.

The neutron and gamma dose rates were also calculated at one meter from either a PWR or BWR storage cask in the vertical orientation. The changes (dose rate ratio) in the maximum neutron and gamma dose rates at one meter from the storage cask indicate that fuel reconfiguration causes significant dose rate changes relative to the nominal intact configuration in the cask outer regions that face air vent locations, i.e., receive radiation directly from streaming through the air vents. At locations away from air vents, the change in radiation dose rate is either small (e.g., 30% for damaged fuel configurations) or negligible.

Containment. Containment analyses showed that the allowable leakage rate exhibits the greatest sensitivity to changes in the mass fraction of fuel released as fuel fines due to cladding breach. Crud was found to be an important factor in the calculation of the allowable leakage rate for the time interval 5 to 40 years after fuel discharge from the reactor because of its relatively large contribution to the total releasable activity. Allowable radionuclide release rate and leakage rate for high-burnup fuel vary as a function of the pellet regions from which fuel fines are released. Fuel fines released from the pellet peripheral region produced smaller allowable leakage rates than fuel fines released from the entire fuel pellet. The importance of the pellet region from which releasable activity originates increases with increasing decay time and fraction of failed fuel rods as illustrated in Figure 4.

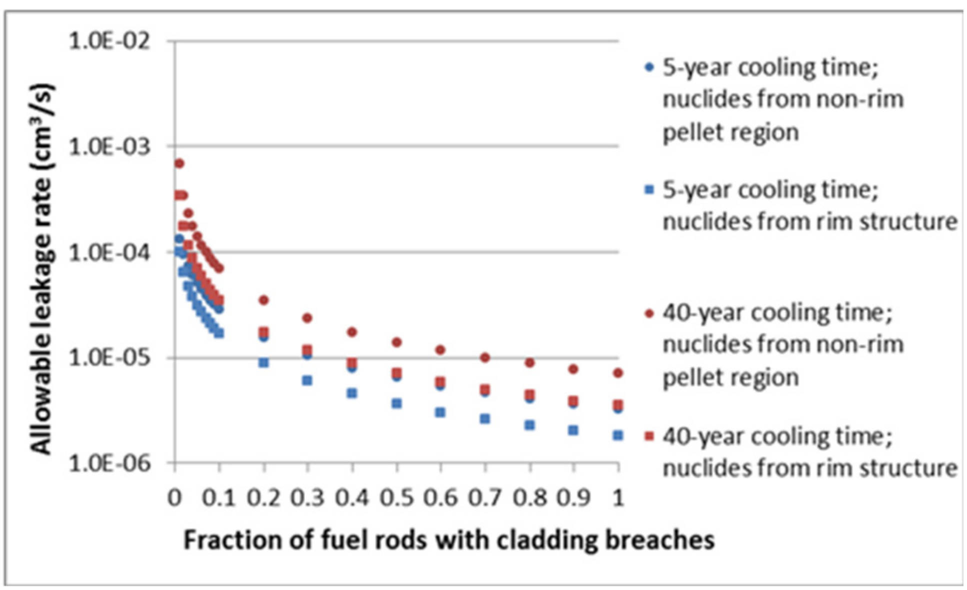


Figure 4. Illustration of fuel fine origination location on allowable leakage rate under NCT

Thermal. For the thermal analyses, the decay heat and radioisotope inventory was calculated from a 5 wt % ^{235}U initial enrichment PWR assembly with a burnup of 65 GWd/MTU at various decay times. The results presented in Table 6 correspond to a 40-year decay time unless identified otherwise. The fuel reconfiguration evaluations associated with Category 1 had the largest impact on the component temperatures relative to the nominal intact fuel configuration. For the vertical cask investigated, which primarily relies on internal convective heat transfer, the release of fission product gases from breached fuel rods into the canister reduced peak internal temperatures, and caused an overall flattening of the axial temperature profiles within the cask. However, for the horizontal cask orientation, where the cask investigated relies on conductive and thermal radiation heat transfer, the release of fission product gases caused a significant increase in internal temperatures. When the fuel failure was represented as damaged SNF, combining fission product gas release with formation of a debris bed, there was a significant increase in the internal component peak temperatures. Fuel reconfiguration scenarios that don't involve additional SNF rod breaches (Categories 2 and 3) have only minor impacts on cask component temperature changes.

Table 6. Summary of thermal analysis results

Vertical package orientation			
Category	**Description**	**Cases compared**	**Peak cladding or neutron absorber* temp. variation (Δ˚C)**
-	Decay time	(20 to 60 years)	-221
-	Decay time	(40 to 60 years)	-45
-	No insolation	(yes to no insolation)	-10
1	Failure of one assembly: only gaseous release	(0% to 100% failure)	-14
	Failure of one assembly: gaseous release and particle bed	(0% to 100% failure, 0.626 packing fraction)	-14
	Failure of all assemblies: only gaseous release	(0% to 100% failure)	-71*
	Failure of all assemblies: gaseous release and particle bed	(0% to 100% failure, 0.626 packing fraction)	+127*
2	Rod pitch to diameter ratio	(1.38 to 1.16)	-51
3	Shifting all assemblies	(top to bottom)	-11
Horizontal package orientation			
Category	**Description**	**Cases compared**	**Peak cladding or neutron absorber* temp. variation (Δ˚C)**
-	Decay time	(20 to 60 years)	-226
-	Decay time	(40 to 60 years)	-51
-	No insolation	(yes to no)	-8
1	Failure of one assembly: only gaseous release	(0% to 100% failure)	+4
	Failure of one assembly: gaseous release and particle bed	(0% to 100% failure, 0.313 packing fraction)	+3
	Failure of all assemblies: only gaseous release	(0% to 100% failure)	+42*
	Failure of all assemblies: gaseous release and particle bed	(0% to 100% failure, 0.417 packing fraction)	+31*
2	Rod pitch to diameter ratio	(1.40 to 1.25)	-12
3	Shifting all assemblies	(top to bottom)	+3

SUMMARY

Overall, the reconfiguration scenarios involving cladding failure (Category 1) repeatedly exhibited the largest impacts relative to the nominal intact configurations for each of the technical disciplines evaluated. The NRC staff is currently working with scientists and engineers at ORNL to examine the effects of fuel reconfiguration scenarios, and to assess the potential safety impact on current storage casks and transportation packages. These analyses are intended to be used in conjunction with other ongoing work, such as described in *Reversible Bending Fatigue Test System for Investigating Vibration Integrity of Spent Nuclear Fuel during Transportation* [15], to gain a better understanding of the mechanical response of high-burnup fuel under NCT as well as the potential implications on safety if fuel reconfigures.

DISCLAIMER

This is a joint ORNL and NRC staff paper. The views expressed herein are preliminary and do not constitute a final judgment or determination of the matters addressed, and do not reflect a regulatory position of the NRC.

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