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United States Department of Energy
Spent Nuclear Fuel Storage and Transportation R&D Activities

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Abstract

As of August 2015, the US had 72 Independent Spent Fuel Storage Installations located in 34 states around the country. (U.S. Nuclear Regulatory Commission, 2015) Collectively, more than 2,000 dry storage casks, of different weights, dimensions, and ages, are currently storing spent nuclear fuel (SNF). The majority of these are dual purpose casks (designed for both storage and transportation), but were not licensed for the long storage times and subsequent transportation that are now needed.

The Department of Energy's strategy is to build at least one consolidated storage facility to store the dry storage canisters before a final disposal location is chosen and operational (U.S. Department of Energy, 2013). This necessitates two large radioactive waste transportation campaigns in the future; one from the current storage sites to the consolidated storage facility and another from the consolidated storage facility to the geologic disposal site (assuming the repository is not co-located with the storage facility). Current Research & Development is developing the technical basis to show that the fuel will have the structural integrity and confinement capability to be transported safely on roads and rail after extended storage.

This current research is focused on increasing our understanding of: 1) material properties of fuel cladding after wet storage, drying, and prolonged dry storage; 2) the thermal conditions the canister and fuel is exposed to in order to better understand the stress and chemical changes the fuel and fuel storage system may undergo over years of storage; 3) any environmental conditions that could degrade the dry storage system over its storage time; and 4) the loads and strains placed on individual fuel rods during conditions of normal transport. This paper will show progress being made in each of these areas and how they synergize to develop the technical basis for the safe transport of spent fuel.

1.0 Background

Current Spent Nuclear Fuel Inventory & Diversity of the Inventory

The Inventory of Commercial Spent Nuclear Fuel:

As of April 2016, there were approximately 80,150 metric tons of heavy metal (MTHM)[1] of commercial spent nuclear fuel (SNF) in the US, of which 54,750 MTHM was in pool storage and

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[1] Under current law, the statutory limit for Yucca Mountain is 70,000 MTHM.

1 Sandia National Laboratories is a multi-mission laboratory operated by Sandia Corporation, a wholly owned subsidiary of Lockheed Martin Corporation, for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000. SAND2016-8539C.

25,400 MTHM was in dry storage (Figure 1). The 99 reactors currently operating in the US discharge about 2,200 metric tons of newly created SNF each year, and because most reactor pools are reaching near full capacity in the US, dry storage capacity must increase at an equivalent rate. This means adding about 160 new storage canisters each year to the approximately 2,080 already in service at active and decommissioned reactor sites. Assuming no new reactors are built and all currently operating reactors receive full renewals of their Nuclear Regulatory Commission (NRC) licenses, and also assuming that no permanent disposal facility becomes available, by the time the last of the current reactors is retired in mid-century the US will have approximately 140,000 metric tons of SNF in storage. Essentially all of this fuel will be in approximately 9,000 dry storage systems distributed across approximately 75 different locations.

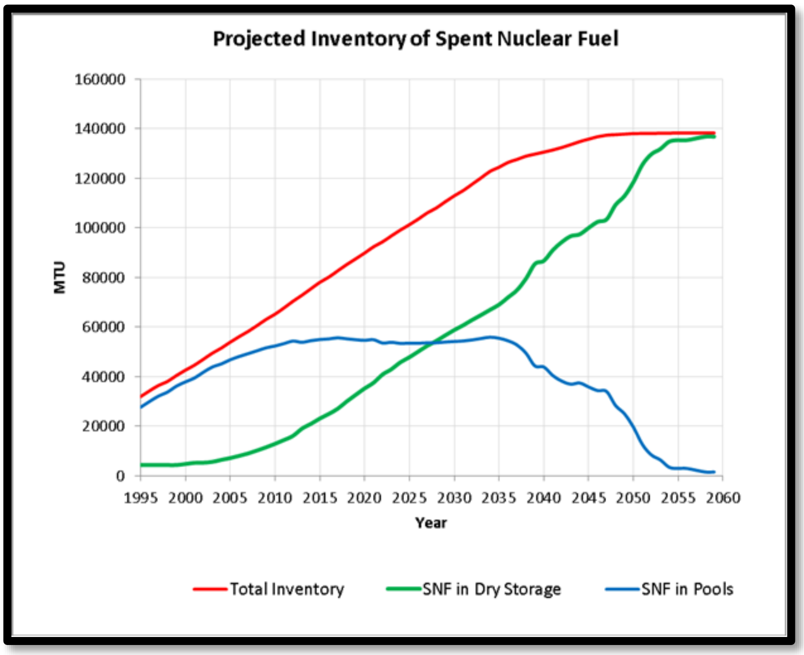

Figure 1. Existing & Projected Inventory of US Commercial SNF. Projections are conditional on an assumption of full license renewals of operating reactors, no new reactor construction, and no permanent disposal of SNF.
(Source: Updated from Figure 6 of Rechard et al., 2015)

The Dual Purpose Canisters (DPCs) themselves are large: up to 2 meters in diameter and 5 meters in length, and the largest currently in use accommodate up to 37 intact fuel assemblies from Pressurized Water Reactors (PWRs), accounting for about two-thirds of the U.S. reactor fleet. A loaded canister may weigh on the order of 70 metric tons, and adding the transportation overpack may increase the total weight to 150 metric tons. Because it is economically advantageous for nuclear power plants to load larger canisters, the canister size continues to increase beyond sizes and weights practical for transportation and subsequent disposal. Engineering solutions for hoist, ramp, and transporter operations appear to be feasible, but need to be accounted for in planning.

There are three main vendors providing multiple dry storage systems in the US. There are both canistered and non-canistered storage systems. Canistered systems share common design features in that spent fuel assemblies are loaded into baskets integrated into a large stainless-steel canister which is welded closed, drained, dried, backfilled with helium, and then placed in a thick-walled concrete overpack (Figure 2 and Figure 3).) for storage. In non-canistered (often called “bare-fuel”) systems, spent fuel assemblies are placed directly into a basket that is integrated into the cask which is sealed with a bolted lid after backfilling with helium and drying. This system does not have an additional overpack. Transportation of the canistered systems would entail removing the welded canister from the storage overpack and emplacing it in a transportation overpack designed specifically for the canister type.

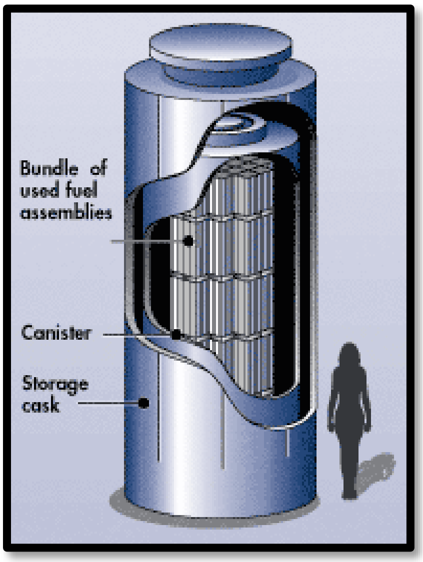

Figure 2. Schematic Illustration of Representative Design for Vertically-Stored Canisterized Dry Storage System

(Source: http://www.nrc.gov/waste/spent-fuel-storage/diagram-typical-dry-casks)

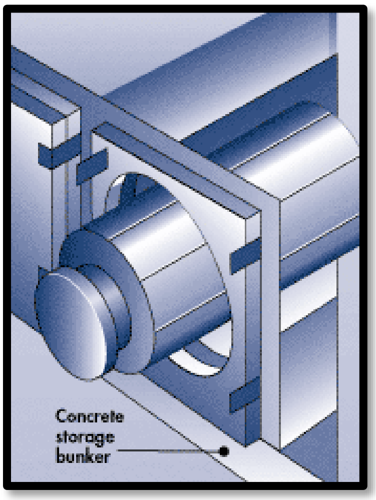

Figure 3. Schematic Illustration of Representative Design for a Horizontally-Stored Canisterized Dry Storage System. (Source: http://www.nrc.gov/waste/spent-fuel-storage/diagram-typical-dry-cask-system.html).

2.0 R&D for Fuel Structural Integrity

2.1 Consolidated Interim Storage

Consolidated interim storage has been proposed as a potential temporary approach to better manage spent fuel storage and could also be used to provide flexibility in repackaging options for ultimate disposal (Source: DOE 2013). Consolidated storage facilities will introduce technical issues associated with the mechanical effects of repeated transportation and storage. The overarching concern is to provide assurance that the fuel will not contact the outside environment.

The spent nuclear fuel has numerous layers of protection from exposure to the environment. The major ones are the:

- hard ceramic structure of the fuel pellet
- metal cladding holding the fuel pellets into a long rod configuration
- ridged assembly hardware designed to maintain the fuel in a fixed geometry
- stainless steel canister and storage and transport overpack (canistered fuel) or the integrated basket and cask (non-canisterized fuel).

There is ongoing research at numerous institutions in each of the areas above to understand how the extended storage time-frame and potential additional transportation will affect the overall mechanical robustness of the storage and transportation system. That research is indicating that the spent fuel appears to be more robust than previously thought.

2.2 Material Properties of Fuel Cladding

Fuel cladding is vital for keeping fuel intact and maintaining configuration and retrievability during handling, storage, and transportation.

The factors thought to have the greatest impact on spent fuel cladding integrity are: radial hydride formation, which is determined by peak cladding temperatures and rod internal pressures; ductile to brittle transition temperatures; pellet-to-pellet and pellet-to-clad interaction; and the number of fatigue cycles and loads to which the cladding is subjected. Research is being performed in each of these areas and the research community is gaining a deeper understanding of how these separate effects work together to affect the strength of spent nuclear fuel (Figure 4).

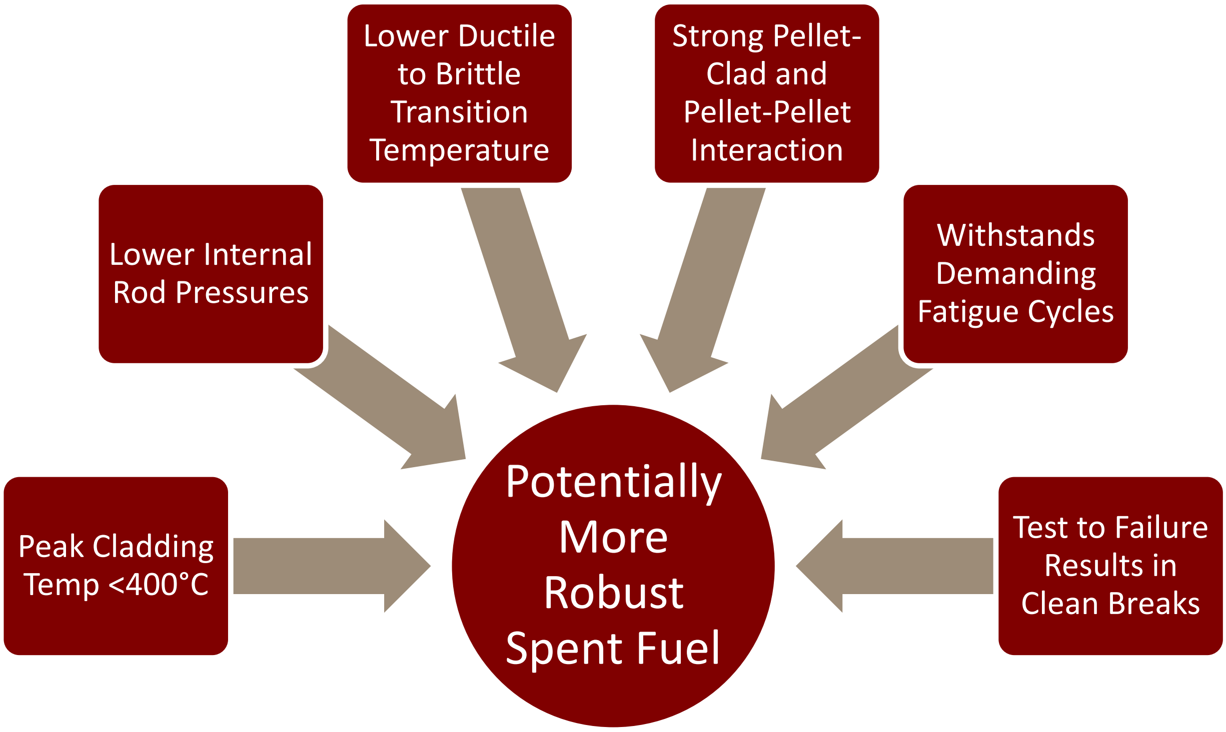

Figure 4. R&D Results from Current Separate Effects Tests and Analyses. The combination of these separate effects all point to more robust fuel than previously thought.

2.2.1 Radial Hydride Formation Which Is Determined by Peak Cladding Temperatures and Rod Internal Pressures: The formation of radial hydrides within the fuel cladding is a concern for long-term storage and subsequent transportation of spent nuclear fuel. After discharge from the reactor, the fuel assemblies are stored in the spent fuel cooling pool where the hydrogen in the spent fuel cladding (most in the form of hydride platelets) is aligned in the circumferential direction. When the fuel is removed from the cooling pools and transferred to the dry storage casks, the cladding temperature increases during cask drying and during the early stages of thermal stabilization. At these higher temperatures, the hydrogen moves into solution. The increasing temperature causes an increase in the rod internal pressure, which causes increased hoop stress in the cladding. As the fuel temperature cools following the drying process, the dissolved hydrogen reprecipitates as hydrides in the cladding. Due to the high hoop stress, the new hydrides may align in the radial direction forming radial hydrides. Extensive radial hydrides reduce clad ductility and lower the cladding's ability to withstand pinch-type loadings which could be experienced if the fuel hits spacer grids or other fuel rods during transportation. Experimental and analytic results are indicating that hydride reorientation is less or a problem than originally thought.

2.2.2 Ductile to Brittle Transition Temperatures: Argonne National Laboratory is currently performing ring-compression tests to simulate pinch-type loadings at the fuel assembly spacer grids, using defueled cladding from various reactors to determine the ductile-to-brittle transition temperature under different temperature and pressure conditions. The results are indicating cladding retains its ductility at the more realistic lower peak cladding temperatures (PCTs) and lower rod internal pressures (and thus lower hoop stresses) expected for standard PWR rods. A lower ductile to brittle transition temperature can indicate that the fuel will have more ductility at lower temperatures. If the fuel is stored for longer times, it will be cooler when it is ultimately transported, so understanding this transition temperature range is important. Results indicate much of the cladding may not enter the brittle regime even after years or decades of dry storage.

2.2.3 Pellet-To-Pellet and Pellet-To-Clad Interaction, Fatigue Cycles, and Clean Breaks: Oak Ridge National Laboratory (ORNL) is conducting cyclic bending tests to quantify strength and fatigue of irradiated fuel segments. The effects of fatigue on fuel rods are of concern when considering transportation events after long-term storage. The ORNL findings indicate that high-burnup fuel shows increased strength potentially due to pellet-clad and pellet-pellet interactions (Figure 5). Results show the fuel is able to withstand tens of thousands to millions of fatigue cycles before breaking. In addition, when the fuel breaks, the breakage is often a clean break between fuel pellets resulting in few fuel particulates being released (Source: Wang et al. 2014) (**Error! Reference source not found.**Figure 6).

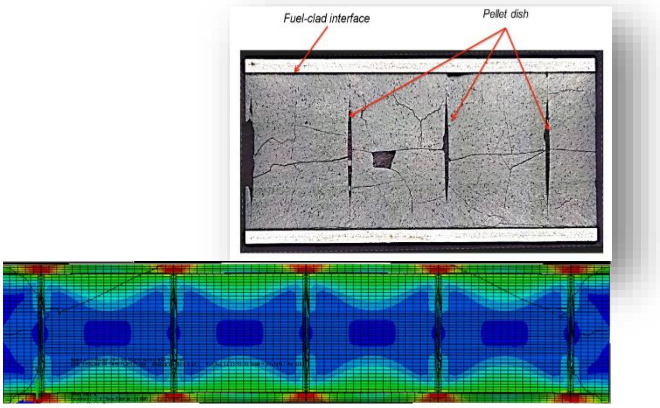

Figure 5. Stress Distribution in Fuel Showing the Fuel Pellets Supporting the Clad Due to Cohesive Bonding. (Source: Wang et al. 2014; ORNL).

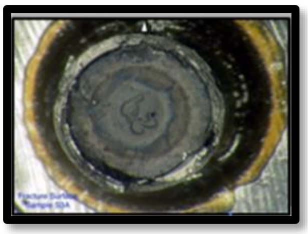

Figure 6. Cladding Clean Break. After tens of thousands of fatigue cycles, the cladding usually breaks cleanly between fuel pellets resulting in the release of few particulates. (Source: Wang et al. 2014; ORNL)

2.3 Thermal Conditions

2.3.1 Modeling of Peak Cladding Temperatures: The temperature of the cladding is an important driver for chemical reactions within the cladding and mechanical properties relating to the robustness of the cladding. To protect from deleterious chemical and mechanical changes in the clad, a regulatory guidance limit of 400°C has been set. To better understand how close actual PCTs are to this limit, Pacific Northwest National Laboratory (PNNL) and ORNL have worked together to produce more realistic and detailed thermal modeling results of PCT in the planned high burnup Research Project Cask and in dry cask storage systems loaded to date. Results indicate that PCTs are significantly lower than previously thought and most likely will be substantially lower than the 400°C regulatory limit (Source: Fort et al. 2016) in one system that has been analyzed in detail (Figure 7). In order to obtain these more realistic clad temperatures, PNNL used actual decay heat calculations as opposed to typically-used calculations by removing many of the known conservativisms within the models used by industry.

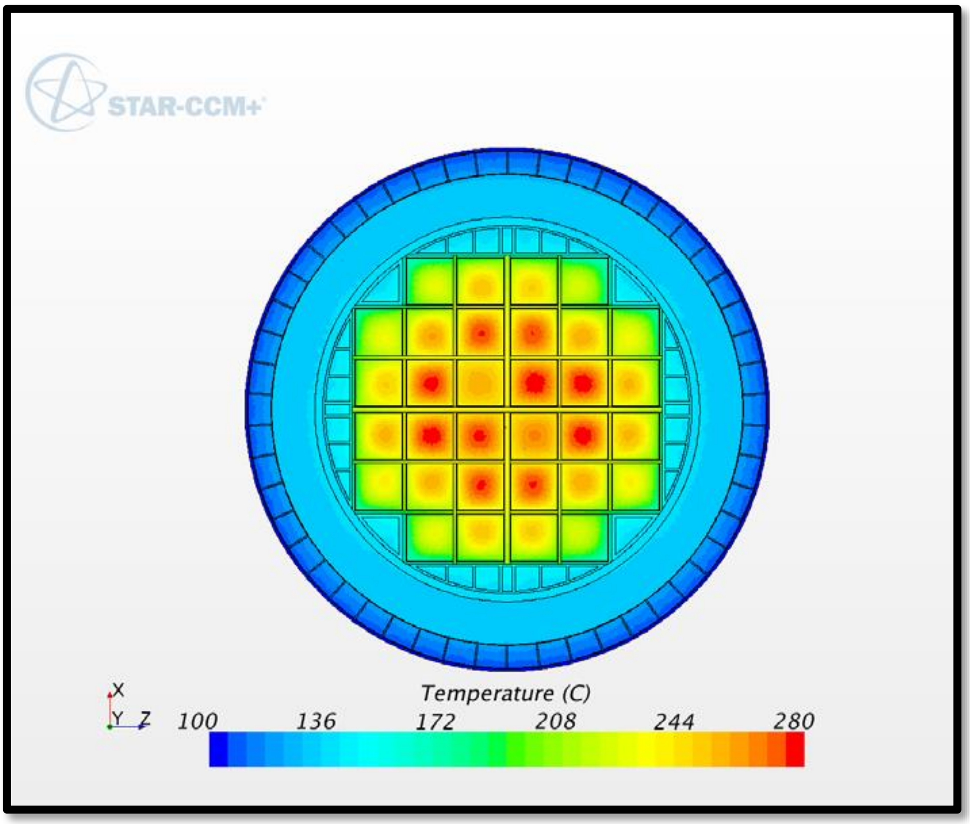

Figure 7. Radial Section View of Component Temperatures at Peak Cladding Temperatures. This shows that predicted temperatures are well below the 400^0 C regulatory guidance limit. (Source: Fort et al. 2016)

Understanding peak cladding temperature is important because a lower PCT will result in less hydrogen going into solution during the drying phase. In addition, a lower clad and fuel temperature will result in lower rod internal pressures and thus lower hoop stress in the cladding. If there is less hydrogen in solution and lower hoop stress, the risk of forming radial hydrides during drying is reduced. Fewer radially oriented hydrides reduce the risk of potential cladding failures during pinch-loading and fatigue.

2.3.2 Experiments to Validate Thermohydraulic Analyses to Experimental Peak Clad Surface Temperatures: In a Department of Energy (DOE) and NRC jointly-funded experiment, Sandia National Laboratories (SNL) is obtaining additional data to validate assumptions in computational fluid dynamics thermohydraulic calculations for BWR spent fuel cask thermal design analyses. Data from these experiments are used to determine steady-state cladding temperatures in above (Figure 8) and belowground configurations (Figure 9) of vertical, dry-cask systems with canisters. This data is needed to evaluate cladding integrity throughout the storage cycle. In this experiment, a 9 x 9 BWR surrogate fuel assembly containing 74 Incoloy® clad fuel rods, 2 water rods, and 7 spacers is heated with electric heater rods. The heater rods simulate heat output from real BWR spent fuel in dry storage with varying burnups and helium backfill conditions.

Figure 8. Aboveground Storage.

(Source: www.nrc.gov/reading-rm/doc-collections/fact-sheets/storage-spent-fuel-fs.html).

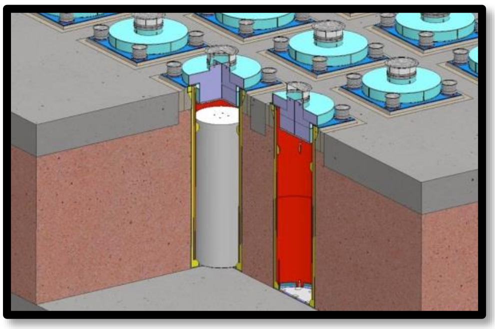

Figure 9. Belowground Storage.

(Source: www.holtecinternational.com/productsandservices/wasteandfuelmanagement/
hi-storm/).

2.4 High Burnup Spent Fuel Data Project

DOE and the Electric Power Research Institute are collaborating on the High-Burnup Spent Fuel Data Project whose goal is to obtain data on physical properties of high-burnup spent fuel during drying and subsequent dry storage of at least ten years. The objective is to generate data to increase our understanding of the cask internal environment and spent fuel degradation processes during storage. This project involves loading a commercially licensed TN-32B storage cask in 2017, with high burnup fuel from a utility storage pool. The fuel is well-characterized and consists of four common cladding alloys, AREVA M5®, Westinghouse ZIRLO®, Westinghouse low-tin Zircaloy-4 rods, and Westinghouse standard Zircaloy-4 rods. To obtain temperature measurements during drying and storage, the cask will be instrumented with 63 thermocouples located axially and radially inside the cask. Samples of the gas within the cask will be taken during the drying phase before going to the storage pad and periodically during storage. The cask will be dried and stored according to the industry's normal operating procedures and will be stored on-site for ten years. The canister will then be transported to a facility for opening and the mechanical properties of the rods will be tested and compared against the baseline properties.

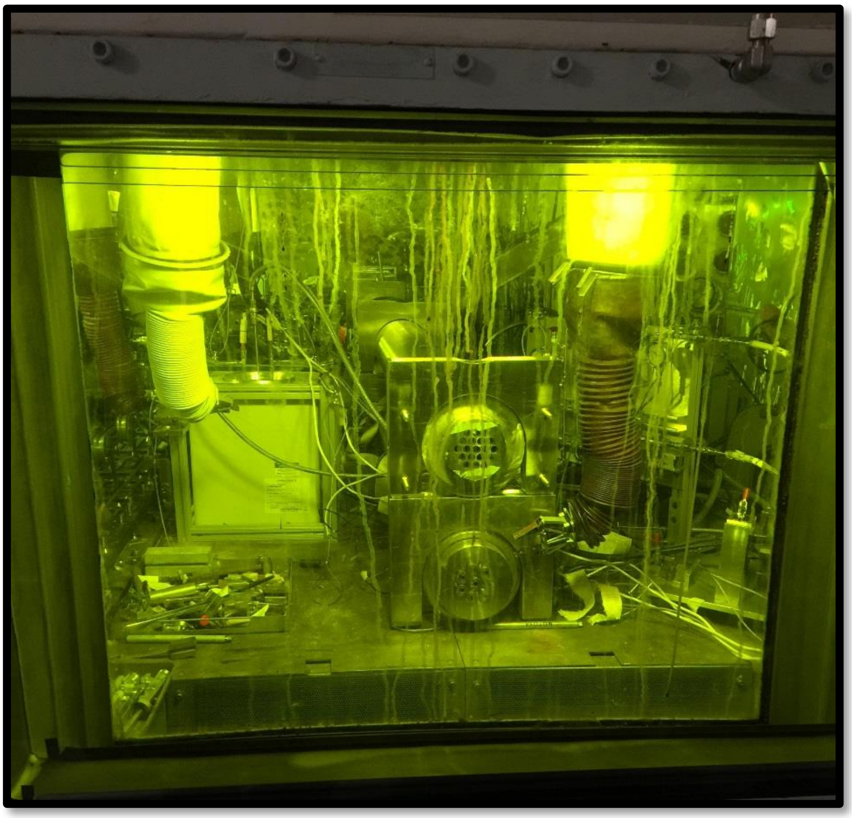

Figure 10. Twenty-five High-Burnup Sister Rods in an ORNL Hot Cell Awaiting Mechanical Property Testing. (Source Photo: Sylvia Saltzstein, SNL)

In order to obtain baseline data of the mechanical properties of the fuel cladding, 25 fuel rods with similar histories will be characterized and tested at ORNL (Figure 10). The results of the baseline cladding mechanical property testing, actual temperature measurements within the cask during drying and ten years of cooling, combined with the cladding mechanical property tests after ten years of dry storage will provide an increased understanding of degradation mechanisms effecting spent fuel during storage and validation of our separate effects testing and analysis.

3.0 Loads and Strains Placed on Fuel Rods During Transport

Understanding the mechanical integrity of the spent nuclear fuel cladding must be coupled with an understanding of the shock and vibration loads that the fuel will realistically experience during its lifetime. Transportation will most likely be the time when the highest shock and vibration loads will be experienced. SNL and PNNL have been collaborating to perform tests and develop models to quantify the loads that fuel will see during Normal Conditions of Transport (NCT). Quantifying these conditions, coupled with the resulting validated models, will allow the community to predict the stresses from shocks and vibrations that fuel will see in different transportation conditions and environments, such as within different casks; on different transportation trucks, rail cars, (Figure 11) and barges; during the transfer between transportation modes; as well as placement of the cask on a storage pad;, and during different off-normal conditions.

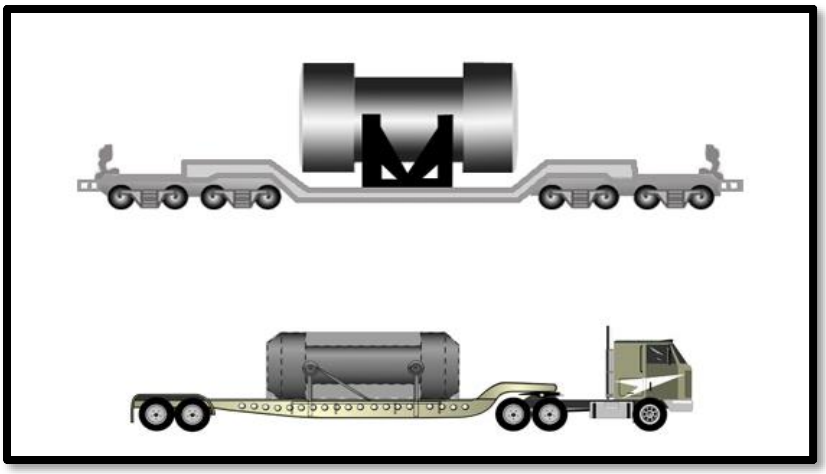

Figure 11. Rail and Truck Transport. These will be the most common transportation modes. (Source: energy.gov/pictures)

3.1 Recent Transportation Tests

Two sets of shaker-table tests were performed both using a 17 x 17 PWR surrogate assembly which was placed within a surrogate truck-cask basket. The first shaker table test was performed in 2013, at SNL using a shaker table with one-degree of freedom (vertical). The second test was performed in 2015 at Dynamic Certification Laboratories (DCL) with a multi-axis (six-degrees of freedom) shaker table. The SNL shaker table tests simulated NCT for only truck shock and vibration using data from NUREG/CR-0128 (NRC, 1978) while the DCL tests simulated NCT truck and rail shock and vibration.

During the SNL tests, all rods were filled with lead rope because simulating the mass of uranium oxide was the primary factor for simulating the dynamic response of a real assembly. Five of the rods were made of Zircaloy-4 and each of those rods was instrumented with accelerometers and strain gauges. During the DCL test, three of the Zircaloy-4 rods were instrumented. One filled with lead “rope,” one with lead pellets, and one with molybdenum pellets in order to quantify the difference between the lead rope and the pellets of different densities and stiffness. During these tests, the assembly, basket, and shaker table were instrumented with accelerometers only. Rail data was obtained from measured shock and vibration data on a 50,000 lb. coal car and translated into those expected to be seen on a S-2043-compliant rail car (AAR, year unknown) using Transportation Technology Center, Inc.’s (TTCI) NUCARS code.

Between the two shaker table tests, in 2014, an over-the-road truck test was performed in Albuquerque, New Mexico (Figure 12). The same surrogate spent fuel assembly was used and all of the rods were filled with lead rope. One of the Zircaloy-4 tubes was instrumented with strain gauges and accelerometers.

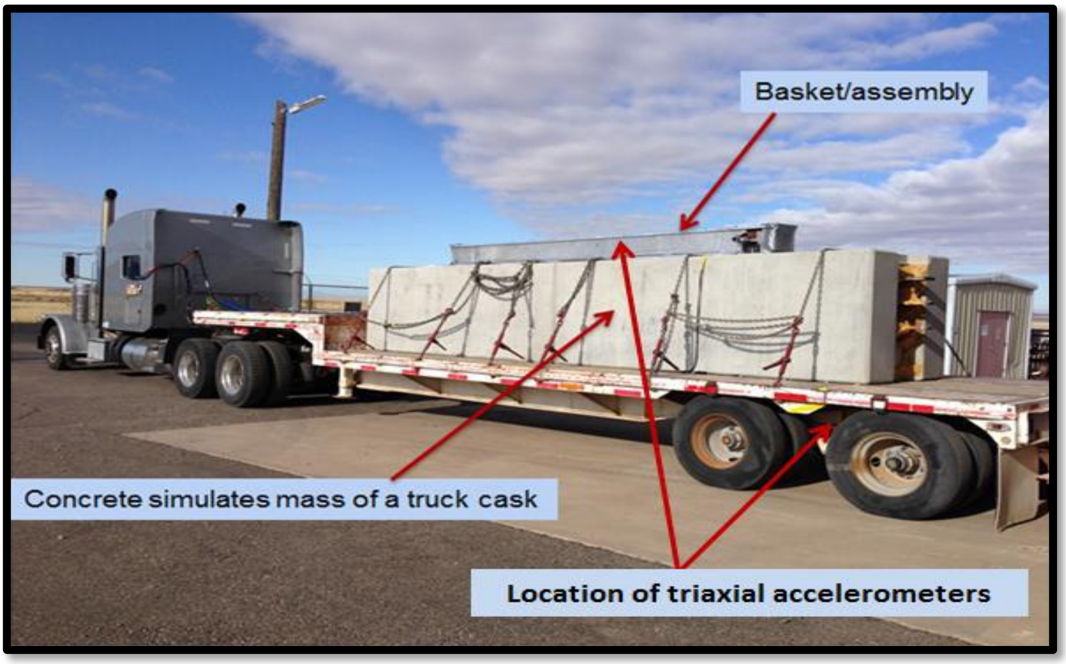

Figure 12. Over the Road Truck Test. Surrogate assembly is within an aluminium basket on top of concrete blocks which simulates the mass of a truck cask. (Photo Credit: Paul McConnell, SNL)

Strains on Zircaloy-4 rods when subjected to simulated NCT (truck or rail) were consistently very low – well below the elastic limit of the alloy of 7000-9000 µm/m. Maximum strains during the transportation tests were typically 100 – 300 µm/m for shock and typically below 10 µm/m for vibration.

All test inputs and results were provided to PNNL for modeling. PNNL modeled the assembly and rods used in the testing. Results from two shaker table tests, and one truck test all produced results well below the yield point for unirradiated or irradiated clad and within the range of the shaker and truck tests (Figure 13).

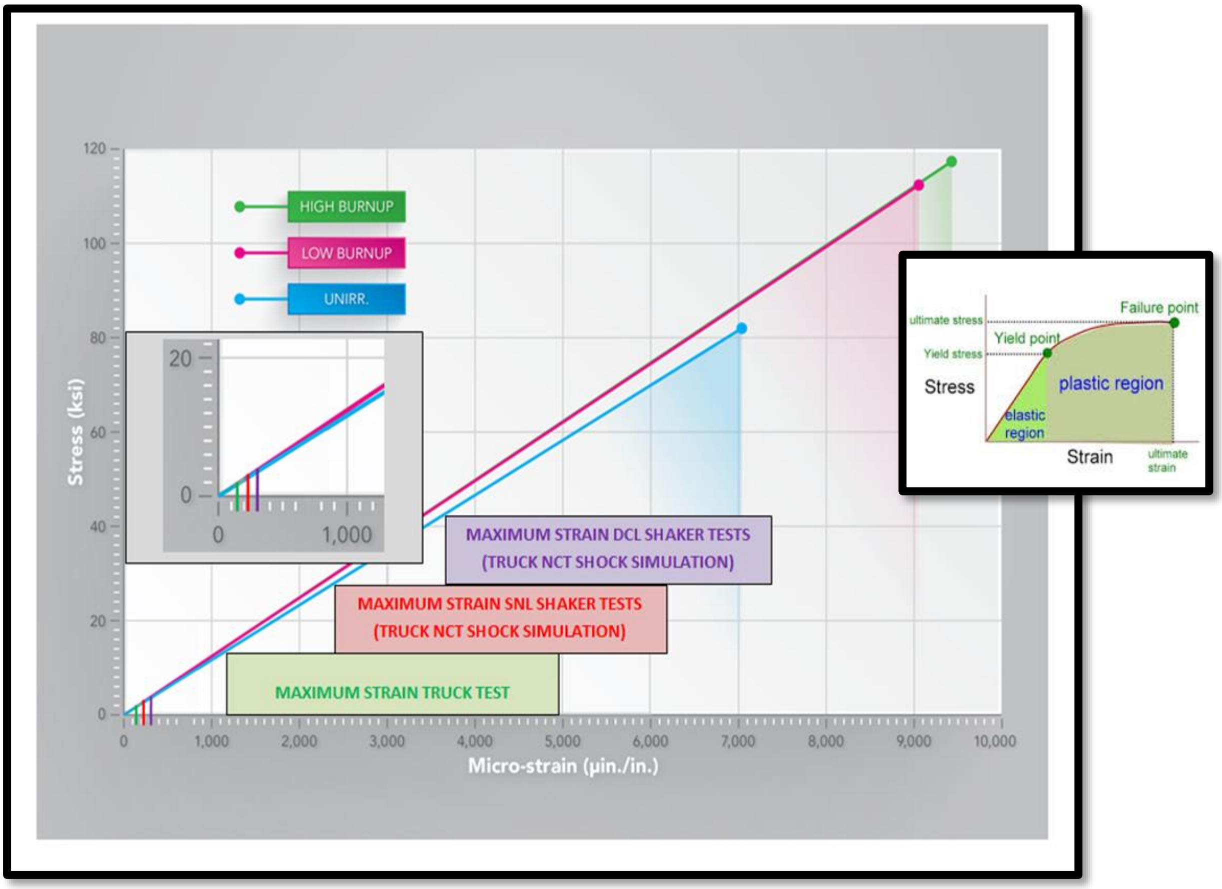

Figure 13. Results from Two Shaker Table Tests & One Truck Test. All produced results well below the yield point for unirradiated or irradiated clad.
(Source: McConnell et al. 2016; SNL and PNNL)

The test results were compared with fatigue testing performed at ORNL on irradiated rods. The calculated bending moment, curvature of the rod, and the strain on the rod in the SNL tests were far lower than those which caused fatigue failure in the ORNL tests (Figure 14).

ORNL Zircaloy-4 fatigue test data

Specimen	Burnup (GWd/MTU)	Applied bending moment (N-m)	Curvature of rod (m^{-1})	Strain on rod (μm/m)	Fatigue cycles $x10^6$	Rod Failure?
D2	63.8	5	0.16	862	6	NO
D4	66.5	7.6	0.23	1239	11	NO
D5	66.5	9	0.22	1185	2.3	YES
D9	66.5	35	1.2	6464	0.007	YES
D13		13.72	0.44	2370	0.129	YES
D14		8.89	0.27	1454	0.27	YES
D15		7.62	0.22	1185	22.3	NO
SNL NCT assembly tests						
		0.7	**0.04**	**≈ 200**		

Q: Cycles-to-failure for a rod subjected to NCT?

A: Cycles-to-failure estimated to be ≫ 22 x10^6

Figure 14. Comparison Between ORNL Irradiated Fuel Fatigue Data & SNL Transportation Load Data.

3.2 Future Multi-Modal Transportation Test

Starting in April 2017, SNL and PNNL will collaborate with Equipos Nucleares S.A. (ENSA) on a multi-modal transportation test that involves: heavy-haul truck transport in Spain. Intercostal barge from Santander, Spain to a northern European port, transoceanic shipment to Baltimore, Maryland, and cross-country rail transport to TTCI near Pueblo, Colorado (Figure 15). Once the cask is in Pueblo, controlled rail tests will be performed before the package makes the return trip. Data will be collected at each leg of the trip and at the transfers (i.e. from truck to coastal ship, coastal ship to ocean liner, ocean liner to rail). The transportation package will consist of approximately two surrogate unirradiated assemblies within a 32-cell basket in an ENUN-32P storage and transportation cask. The remaining fuel cells will be filled with a dummy assemblies made of concrete to simulate the mass of a real assembly. Each surrogate assembly will have rods instrumented with strain gauges and accelerometers. The basket, cask, cradle, and transport platform (e.g., rail car) will be instrumented with accelerometers. Data will be collected during each leg of the trip, including the transfers between modes. This data set will provide the most realistic data to date of all of these different modes of transportation including the shocks and vibrations seen during the transfer operations.

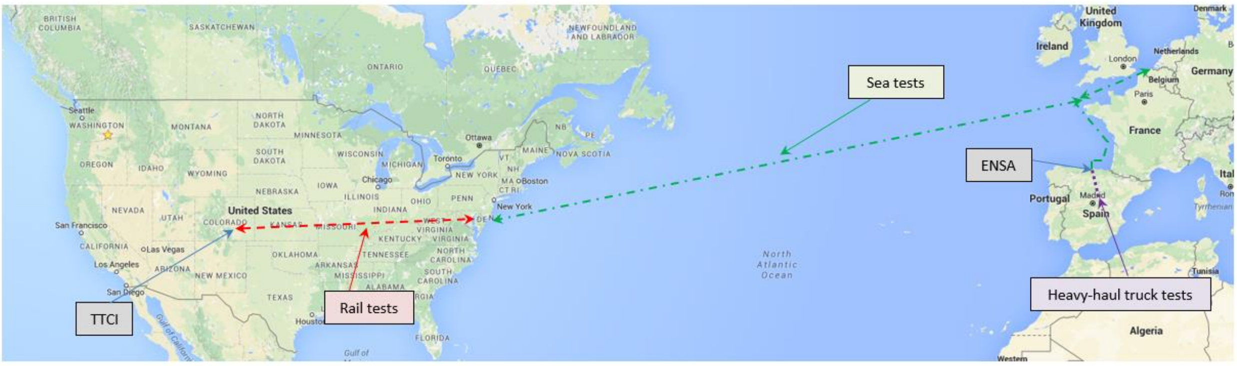

Figure 15. Route at Dates for the 2017 Multi-Modal Transportation Test.
(Source: Paul McConnell, SNL)

4.0 How it Comes Together

Numerous different, seemingly loosely connected areas of research are coming together to show that SNF is more robust than previously thought (Figure 16). This paper provided a brief overview of the results of some of those separate effects. In summary, more realistic thermal modeling is indicating that peak cladding temperatures may be substantially lower than the 400°C regulatory guidance limit. Because temperature is a driving force for mechanical changes in the cladding, this is an important finding. When peak clad temperatures are lower, less hydrogen will move into solution during the drying phase and consequently less hydrogen will be available to precipitate back into the cladding in the radial orientation as the cladding cools. Secondly, if the fuel and cladding are cooler, the rod internal pressure will be lower resulting in lower cladding hoop stress. Lower hoop stress also reduces the formation of radially oriented hydrides. These are important findings because fewer radial hydrides in the cladding allow the fuel to better withstand pinch-type loads that could be experienced during transportation.

Research on quantifying the ductile-to-brittle transition temperature is revealing that cladding is more ductile when subjected to these more realistic PCTs and hoop stresses. In the absence of significant radial hydride formation, the ductile-to-brittle transition temperature remains very low. Research on the fatigue strength of cladding shows that cladding can withstand at least tens of thousands of fatigue cycles that are much more extreme than fuel would experience during NCT. When breakage occurs, it is often a clean break between pellets which results in little particulate release. All of these separate effects point to a stronger fuel/clad system than previously predicted. Results from the High Burnup Spent Fuel Data Project are hypothesized to confirm many of these results.

Lastly, even brittle fuel typically won't break unless it experiences an external stress to initiate failure. Results from two shaker table and one over-the-road truck test show that the stress from shocks and vibrations during transport are well below that experienced by irradiated high-burn up fuel in laboratory conditions. A multi-modal transportation test in 2017 will collect more data on all transportation modes as well as the transfers between modes.

The combination of these separate effects all point to more robust fuel than previously thought.

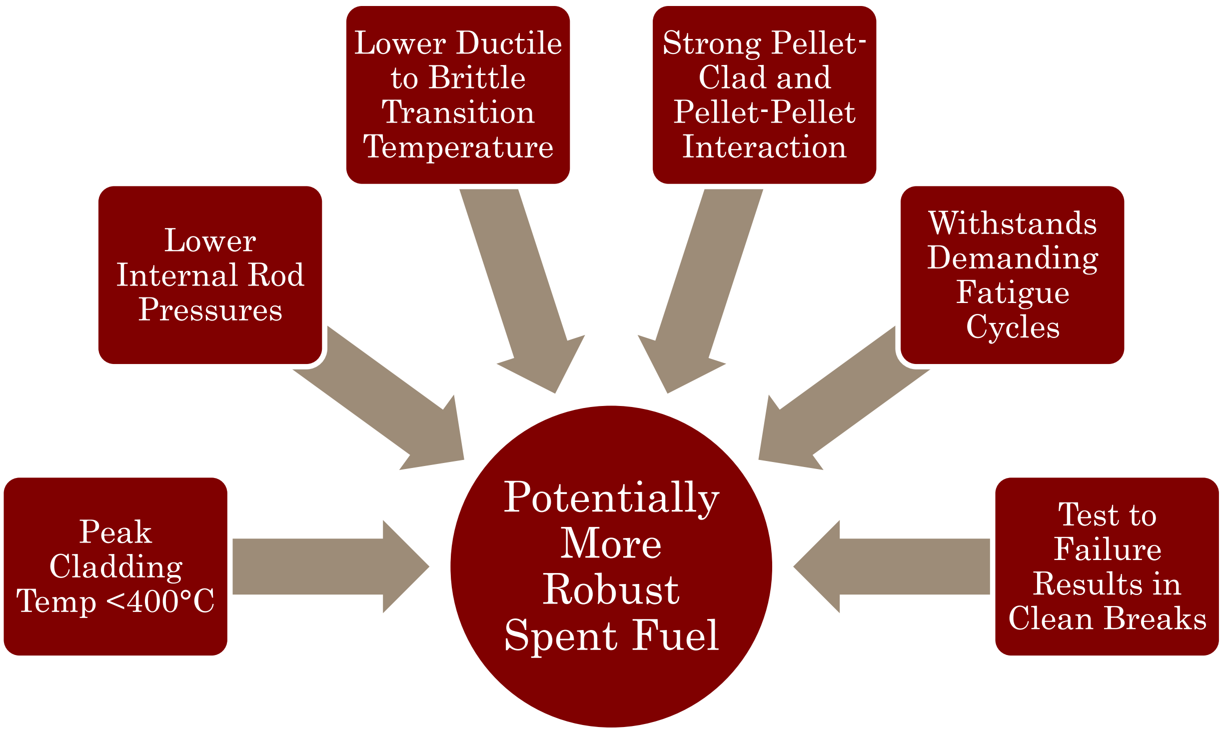

Figure 16. R&D Results from Current Separate Effects Tests and Analyses.

5.0 Acknowledgements

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6.0 Acronym List

BWR	Boiling Water Reactor
DCL	Dynamic Certification Laboratory
DOE	U.S. Department of Energy
ENSA	Equipos Nucleares S.A., Spain
ISFSI	Independent Spent Fuel Storage Installation
MTHM	Metric Tons of Heavy Metal
NCT	Normal Conditions of Transport
NRC	U.S. Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PNNL	Pacific Northwest National Laboratory
PWR	Pressurized Water Reactor
R&D	Research and Development
SNF	Spent Nuclear Fuel
SNL	Sandia National Laboratories
TTCI	Transportation Technology Center, Inc.

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