USED FUEL DISPOSITION CAMPAIGN

Used Nuclear Fuel Loading and Structural Performance Under Normal Conditions of Transport - Modeling, Simulation and Experimental Integration RD&D Plan

Fuel Cycle Research & Development

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Used Fuel Disposition Campaign

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EXECUTIVE SUMMARY

Under current U.S. Nuclear Regulatory Commission regulation, it is not sufficient for used nuclear fuel (UNF) to simply maintain its integrity during the storage period, it must maintain its integrity in such a way that it can withstand the physical forces of handling and transportation associated with restaging the fuel and moving it to treatment or recycling facilities, or a geologic repository. Hence it is necessary to understand the performance characteristics of aged UNF cladding and ancillary components under loadings stemming from transport initiatives. Researchers would like to demonstrate that enough information, including experimental support and modeling and simulation capabilities, exists to establish a preliminary determination of UNF structural performance under normal conditions of transport (NCT).

This research, development and demonstration (RD&D) plan describes a methodology, including development and use of analytical models, to evaluate loading and associated mechanical responses of UNF rods and key structural components. This methodology will be used to provide a preliminary assessment of the performance characteristics of UNF cladding and ancillary components under rail-related NCT loading. The methodology couples modeling and simulation and experimental efforts currently under way within the Used Fuel Disposition Campaign (UFDC). The methodology will involve limited uncertainty quantification in the form of sensitivity evaluations focused around available fuel and ancillary fuel structure properties exclusively. The work includes collecting information via literature review, soliciting input/guidance from subject matter experts, performing computational analyses, planning experimental measurement and possible execution (depending on timing), and preparing a variety of supporting documents that will feed into and provide the basis for future initiatives.

The methodology demonstration will focus on structural performance evaluation of Westinghouse WE 17×17 pressurized water reactor fuel assemblies with a discharge burnup range of 30-58 GWd/MTU (assembly average), loaded in a representative high-capacity (≥32 fuel rod assemblies) transportation package. Evaluations will be performed for representative normal conditions of rail transport involving a rail conveyance capable of meeting the Association of American Railroads (AAR) S-2043 specification. UNF modeling is anticipated to be defined to the pellet-cladding level and take in to account influences associated with spacer grids, intermediate fluid mixers, and control components. The influence of common degradation issues such as ductile-to-brittle-transition will also be accounted for. All model development and analysis will be performed with commercially available software packages exclusively. Inputs and analyses will be completely documented, all supporting information will be traceable, and bases will be defendable so as to be most useful to the U.S. Department of Energy community and mission. The expected completion date is the end of fiscal year (FY) 2013.

Success of this initiative is defined by exercising the computational models developed as part of this initiative to evaluate loading and associated mechanical responses of UNF rods and key structural components under NCT. In doing so, a preliminary assessment of the post-transport

integrity of the UNF rods and effects on subsequent geometry, criticality, and retrievability will have been established.

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ACRONYMS

AAR	Association of American Railroads
ANL	Argonne National Laboratory
BWR	boiling water reactor
CRIEPI	Central Research Institute of Electric Power Industry, a research institute of the Japanese nuclear industry
DBTT	ductile-to-brittle-transition temperature
DOE	U.S. Department of Energy
DOE-NE	U.S. Department of Energy Office of Nuclear Energy
FEM	finite element method
FY	fiscal year
g	gravitational acceleration
GWd	gigawatt-days
HAC	hypothetical accident condition
IFM	intermediate fluid mixer
INL	Idaho National Laboratory
M&S	modeling and simulation
MP	material properties
MPC	multi-purpose canister
MTU	metric tons (Tonnes) of uranium
NCS	normal conditions of storage
NCT	normal conditions of transport
NFST	Nuclear Fuels Storage and Transportation
NRC	U.S. Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PNNL	Pacific Northwest National Laboratory
PWR	pressurized water reactor
RD&D	research, development, and demonstration
SAR	safety analysis reports
SME	subject matter expert

SNL	Sandia National Laboratories
SOW	statements of work
TL	Transportation Loading
TTCI	Transportation Technology Center, Inc.
UFDC	Used Fuel Disposition Campaign
UNF	used nuclear fuel
UQ	uncertainty quantification
WBS	work breakdown structure

USED FUEL DISPOSITION CAMPAIGN
Used Nuclear Fuel Loading and Structural Performance Under Normal Conditions of Transport - Modeling, Simulation and Experimental Integration RD&D Plan

1. INTRODUCTION

The U.S. Department of Energy Office of Nuclear Energy (DOE-NE), Office of Fuel Cycle Technology has established the Used Fuel Disposition Campaign (UFDC) to conduct the research and development activities related to storage, transportation, and disposal of used nuclear fuel (UNF) and high-level radioactive waste. Under U.S. Nuclear Regulatory Commission (NRC) regulations, it is not sufficient for UNF to simply maintain its integrity during the storage period. It must maintain its integrity in such a way that it can withstand the physical forces of handling and transportation associated with restaging the fuel and moving it to a different location (such as an interim storage site, a geologic repository, or a treatment/recycling facility). Hence, understanding mechanical performance under cumulative loading stemming from normal conditions of storage (NCS), transfer (from storage container to transport container if needed), and normal conditions of transport (NCT) are necessary as it establishes the safety basis via maintaining the fuel confining boundary (geometry), maintains criticality safety, and is one of the critical components to the preservation of retrievability. Because of this, an important part of UFDC research and development is related to the mechanical loads on used nuclear fuel, cladding, and key structural components of the fuel assembly during NCT and NCS, and the response of the used fuel and assembly to those loads.

1.1 BACKGROUND

In the United States, the UNF inventory continues to increase as nuclear power generation, part of the nation's commercial power generation portfolio, continues to assist in meeting the country's energy demands. At the end of 2012, it was estimated that the commercial nuclear industry generated approximately 70,000 metric tons (Tonnes) of uranium (MTU) contained in about 245,000 UNF assemblies (140,000 from boiling water reactors [BWRs] and 105,000 from pressurized water reactors [PWRs]). By 2020, the projected total UNF discharges will be approximately 88,000 MTU (Carter et al. 2012). By then, roughly 35,000 MTU is expected to be in dry storage with the remaining 53,000 MTU in the reactor pools. At the time waste acceptance starts, the fuel in dry storage represents a legacy that must be dealt with regardless of what approach is taken to manage newly discharged fuel going forward. By 2060, when all currently licensed reactors will have reached the end of their operational licenses, assuming a 60-year maximum, there will be approximately 140,000 MTU of UNF discharged from the reactor fleet (Carter et al. 2012).

Of particular interest are assemblies that have achieved high burnup, because technical questions have been raised relative to cladding integrity of high-burnup fuel, as discussed in NRC Interim Staff Guidance Memorandum 11 (NRC 2003). The current average discharge burnup for PWRs is approximately 48 GWd/MTU, and for BWRs it is approximately 43 GWd/MTU (EPRI 2010). However, by 2020 it is projected that the average discharge burnups will be 58 GWd/MTU for PWRs and 48 GWd/MTU for BWRs.

As the burnup increases, a number of changes occur that may affect the performance of the fuel, cladding, and assembly hardware in storage and transportation. These changes include increased thickness of the cladding corrosion layer, increased hydrogen content in the cladding, increased creep strain in the cladding, increased fission gas release, and the formation of the high burnup structure at the surface of the fuel pellets. Because of these changes and the lack of fuel performance data at higher burnups, especially under design basis accident conditions, the current maximum rod-averaged burnup is limited by the NRC to 62 GWd/MTU (OECD 2012). Newer cladding materials such as ZIRLO™ and M5® were developed to mitigate the effects on cladding associated with these higher burnups. However, because these materials are relatively new, very limited data are available publicly that can be used to determine how these materials may perform under storage and transportation conditions (Hanson et al. 2012).

Depending on the drying process and/or storage conditions, the ductile-to-brittle-transition temperature (DBTT) of specific types of high burnup cladding may increase substantially because of hydride reorientation. As the fuel cools during NCS, it may cool below the DBTT before the fuel is handled and transported at the end of interim storage. If the UNF cladding temperature at the time of transport is below the DBTT, the chances for damage to the fuel cladding under NCT will increase.

The implementation of consolidated interim storage of UNF, consistent with one of the Blue Ribbon Commission on America's Nuclear Future recommendations (BRC 2012), and DOE's recently published *Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste* (DOE 2013), would necessitate the implementation of a large-scale transportation program. Some of the used fuel in the inventory may be transported at least twice to get it to a repository—once from the reactor to the consolidated interim storage facility and then to a repository for final disposal—after an unknown storage duration. Given the uncertainty in material properties of high burnup UNF, variability in storage duration, and the potential variability in the magnitude and duration of normal loading during transport, it is appropriate to investigate whether or not a single or multiple transports would have a negative impact on fuel integrity and its suitability to meet the regulations retrievability after transport. Hence, understanding performance characteristics of UNF cladding and ancillary components under cumulative loading stemming from NCS, transfer (from storage container to transport container if needed), and NCT is necessary as it establishes the safety basis via maintaining the fuel confining boundary (geometry), maintains criticality safety, and is one of the critical components to the preservation of retrievability.

Researchers would like to demonstrate that enough information, experimental support and modeling and simulation capabilities exist to establish a preliminary determination of UNF structural performance under NCT loading. A steering team composed of national laboratories, DOE, and NRC staff met to discuss project feasibility. The group identified the basic information required and established a preliminary path for a successful research, development, and demonstration (RD&D) plan. The Steering Team Meeting took place on December 11-12, 2012. The "Method and Approach" document (Adkins 2013) was developed to identify the decisions and proceeds of that meeting. The Steering Team participants are listed in Appendix A. An Implementation Group composed of national laboratories, DOE, NRC, and Transportation Technology Center, Inc. (TTCI) staff met February 21-22, 2013 to identify specific inputs needed to develop a focused overall RD&D project implementation plan, based on input from subject matter experts selected for their expertise related to this effort. The Implementation Group list of participants is provided in Appendix B. Both meetings took place at the DOE National Nuclear Security Administration (NNSA) Nevada Site Office. This document was assembled to capture and communicate the proceedings of the two working meetings and to identify that plan.

A subset of the subject matter experts (SMEs) has been selected as the Implementation Team and statements of work (SOWs) have been assembled for the purpose of identifying laboratory-specific work scope and responsibilities. These are shown in Appendix C. This document will serve as the RD&D project implementation plan. The plan will contribute to an overarching blueprint for resolving the numerous technical challenges related to extended storage and transportation of UNF.

1.2 OBJECTIVES

The objective of the work is to determine the mechanical loads, with quantified uncertainties, on used nuclear fuel, cladding, and key structural components of the fuel assembly during NCT and NCS, and to assess the response of the UNF and assembly hardware to those loads. The work scope will support development and integration of UNF data and analysis capabilities, as well as support the UFDC mission regarding scientific research and technology development to strengthen the technical basis for storage and transportation of UNF. The proposed work scope includes, but is not limited to, collecting information via literature review, soliciting input and contributions from SMEs, developing and demonstrating a methodology, performing computational analyses, planning and executing experimental measurement, and preparing a variety of supporting documentation that will feed into and provide the basis for future initiatives. The completion date for the work scope identified herein is the end of fiscal year (FY) 2013. The fundamental near-term objectives of this initiative are:

- **Perform literature reviews to establish the quantity and type of information available in three specific areas: mechanical loading during storage and transportation; system and UNF material properties; and relevant modeling and simulation techniques.** This review process will serve to identify, assemble, and document applicable data and information, and identify information gaps. The resulting information will be used to establish databases to support work performed under this initiative, as well as future programs and tasks. Specific details and associated guidance are provided in subsequent sections of this document.
- **Develop a database of information required for modeling via the literature reviews and associated topical influencing factors.** Information related to material properties, specified

loading conditions, applicable boundary conditions, etc., will be consolidated, and sensitivity ranges will be selected for use in future modeling and simulation (M&S) efforts. Uncertainty quantification (UQ) will be limited to UNF mechanical properties during this initiative because of the volume of work scheduled to be conducted within the defined performance period.

- **Apply information from the literature reviews to construct models for performing high-resolution deterministic structural evaluations.** These models (also referred to as M&S tools) will be constructed and documented in a fashion that is conducive to future upgrades and modifications/alterations for performing alternate simulations, and readily accommodate new information as it becomes available. Fundamentally, the models will be constructed so that they may be readily used in future initiatives/assessments to address emergent issues or questions.
- **Select and perform one or more validation cases to establish the credibility of the methodology as well as the models' predictive capability.** The number and type of validation cases will be selected based on availability and pertinence to the deterministic predictions intended to be performed for this initiative.
- **Provide an initial demonstration of the developed model's capabilities by performing preliminary deterministic evaluations of moderate-to-high burnup UNF mechanical performance under NCS and NCT conditions.** The completion of this demonstration will serve to identify data and information gaps that might exist, and the types of testing that might be needed to fill those gaps. It will also demonstrate the development and integration of UNF data and analysis capabilities, as well as couple M&S and experimental efforts with focused sensitivity evaluations. Future sensitivity evaluations, through simulation, can provide focus for future material testing and examination studies to refine correlations and relationships critical to understanding UNF structural performance and behavior.

The long-term goals and objectives of the initiative are to:

- provide an analytical assessment of UNF integrity when subjected to NCT
- identify the type of ductility demands that would be required to ensure adequate high burnup UNF performance and survivability under a normal transport campaign
- answer questions relative to the ability of high burnup UNF to maintain its integrity and retrievability as it moves through each step of the waste management process (storage, transportation, repackaging, and disposal)
- develop validated models and information to aid in making decisions regarding determination of storage and disposal paths
- identify tests that would be sufficient to address technical issues that need to be resolved
- contribute to an overarching blueprint for resolving the numerous technical challenges related to extended storage and transportation of UNF.

2. PERSPECTIVES AND CONSIDERATIONS

Understanding performance characteristics of UNF under loads stemming from NCS, transfer (from storage container to transport container if needed), and NCT and quantifying cumulative effects is important to maintaining safety bases and preserving fuel retrievability. However, integration of all possible loading influences is beyond the scope of the current initiative. With this in mind, the Steering Team and Implementation Group elected to focus on constructing the computational models to accurately evaluate transportation loading scenarios. Because this initial effort is focused on computational model development, existing literature data will be used wherever possible and actual experimental work will be limited. Initial fuel material condition/state will be assumed based on the best information available to date. The intent is to address cumulative effects through a multi-staged approach. This will be handled by modularizing the computational models so that the capability to account for storage and transfer conditions outside those assumed in this study can be readily incorporated in the future.

It is anticipated that the M&S tools developed during this initiative will be capable of performing accident-related performance assessments; however, no accident-related evaluations will be performed for this initiative. This decision is influenced by the fact that storage and transportation packages involved in an event (beyond the definition of normal or off-normal conditions per 10 CFR 71 or 10 CFR 72) would have a unique and different disposition path than normal packages. Additionally, despite a possible safety basis shift from the fuel cladding to the canister level or possible alteration in the definition of retrievability, understanding of UNF structural performance is still necessary to demonstrate that regulatory safety basis requirements are met.

2.1 TRANSPORTATION MODES

Understanding the influence of rail and over-the-road transportation modes is necessary. The most recent national transportation plan (DOE 2009) established that a majority (possibly even greater than 90 percent) of the UNF inventory will be transported by rail. Per this document, DOE selected the mostly rail scenario as the transportation mode to be analyzed in a repository-related environmental impact statement. Additionally, The Office of Civilian Radioactive Waste Management issued a policy stating that dedicated trains will be the usual mode of rail transportation for UNF and high-level radioactive waste. This provides the basis for focusing on the rail transportation mode for this work.

UNF rail transport conveyances will need to meet the Association of American Railroads (AAR) *Performance Specification for Trains Used to Carry High-Level Radioactive Material*, AAR Specification S-2043 (AAR 2003). An example of this type of conveyance is the Naval UNF Transporter M-290-0001 Cask Car shown in Figure 2.1 that is in the process of commissioning. The specified peak accelerations provided by a conveyance meeting this standard are also reduced substantially in comparison to those measured in previous studies (such as Magnuson 1978) for truck transport, or from loading that would be anticipated from a typical conveyance device designed to meet the ANSI N14.23 restraint requirement. Over-the-road transportation information will be collected during the literature review portion of this initiative. Truck-transport-associated loading is anticipated to bound that of rail and will eventually require evaluation. However, rail will remain the primary focus at least until an order of magnitude deterministic assessment on the ductility requirements of UNF under this mode of transport is fully realized.

Figure 2.1. Naval UNF Transporter M-290-0001 Cask Car (under commissioning to meet AAR S-2043 [AAR 2003])

2.2 SELECTION OF A REPRESENTATIVE TRANSPORT CONFIGURATION AND USED NUCLEAR FUEL ASSEMBLY FOR MODEL DEVELOPMENT AND ANALYSIS

For economic reasons, the nuclear industry is currently using large dry storage systems with canister capacities up to 37 PWR and 80 BWR fuel assemblies, with larger capacity canisters being considered for future use.

According to Leduc (2012), "…approximately 84% of [used] commercial fuel [in dry storage] in the U.S. is stored in single welded canisters inside individual concrete or steel-encapsulated concrete cylindrical storage overpacks or rectangular horizontal storage modules." A total of 1,570 loaded canisters in dry storage systems are currently in use at active or decommissioned reactors. Figure 2.2 shows the proportion of the total canisters made up by each canister type. These same data are re-plotted in Figure 2.3 with the canister types grouped based on design and vendor. Canister systems from a single vendor often share design features such as physical dimensions and material compositions. The five most currently used canisters, in descending order, are the HI-STORM MPC-68 (Holtec), the NAC-UMS UMS-24 (NAC International), the NUHOMS 24P (Transnuclear), the HI-STORM MPC-32 (Holtec), and the NUHOMS 61BT

(Transnuclear). When broken down by vendor/design, three vendors have provided approximately 75 percent of the total canisters in use. These are, in descending order: NUHOMS (Transnuclear), HI-STORM (Holtec), and NAC-UMS UMS-24 (NAC International) (Miller et al. 2012).

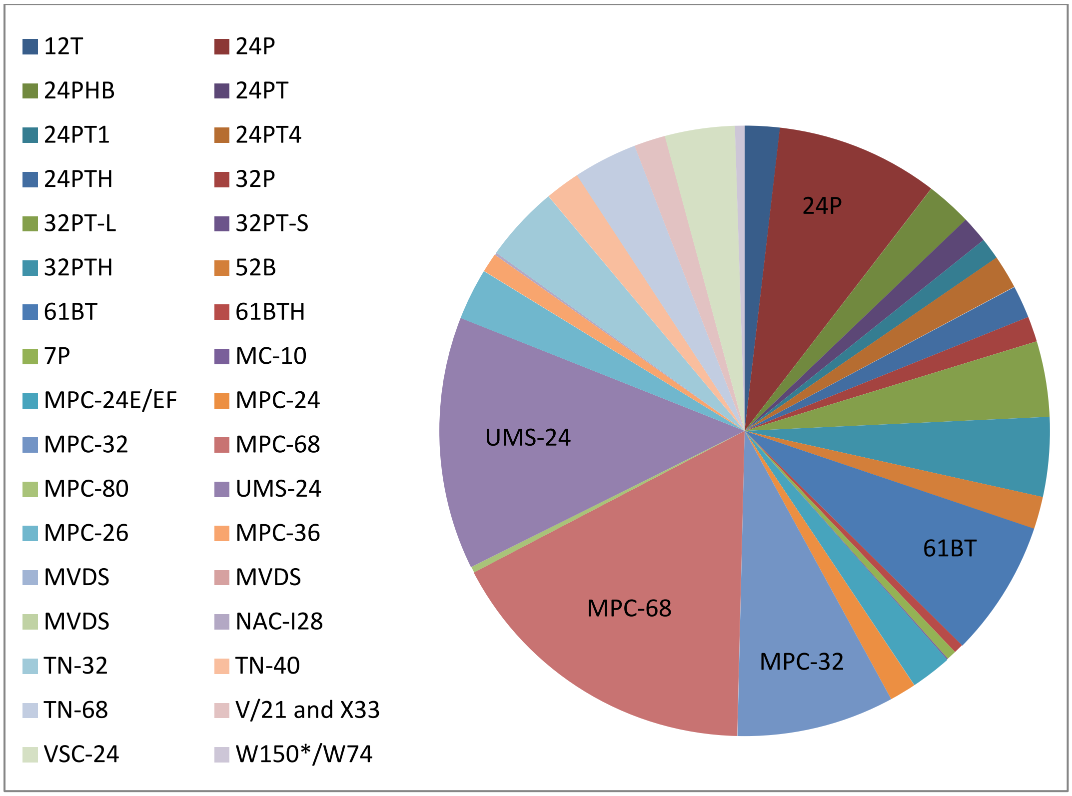


Figure 2.2. Relative Frequency of Storage Systems in Use (Source: Miller et al. 2012)

As shown in Figure 2.2 an abundance of UNF canister configurations requiring future transport exist. Of the light water reactor fuel assemblies, approximately two-thirds of the existing inventory is from PWR reactors (on a MTU basis) (Carter et al. 2012). As such, the PWR is targeted as a configuration of high interest. As of 1987, 35 percent of all PWR UNF inventory was WE 17×17 type fuel assemblies (Weihermiller and Allison 1979). By 1993, 50 percent of all PWR UNF inventory was WE 17×17 type fuel assemblies (Energy Information Administration 1995). Now, almost all PWRs (except combustion engineering plants) use WE 17×17 type fuel assemblies. As such, the WE 17×17 PWR fuel assemblies, or permutations of this base assembly configuration, will represent the greatest fraction of PWR UNF inventory.

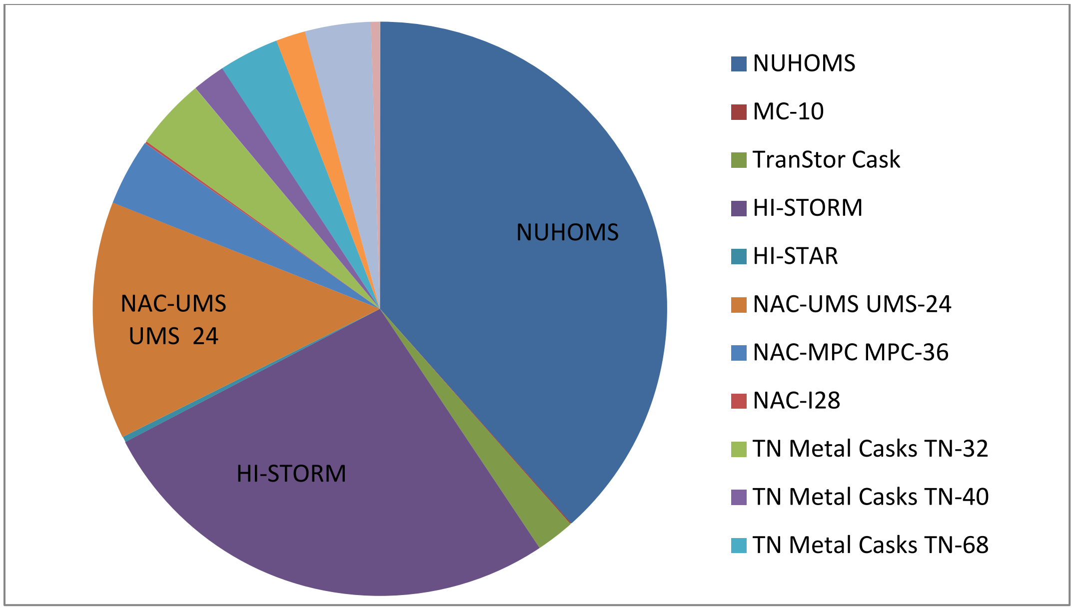


Figure 2.3. Relative Frequency of Existing Storage Systems Grouped by Design and Vendor (Source: Miller et al. 2012)

The WE 17×17 fuel assembly type should represent the lesser mechanically robust configuration in comparison to other multi-rod PWR configurations because of reduced fuel rod diameter and cladding thickness. As such, the Steering Team, with Implementation Group concurrence, decided that an initial transport packaging candidate must be representative of a high-capacity PWR (≥32 assembly payload) transportation package and would be defined to contain a canisterized payload initially comprising WE 17×17 fuel assemblies. This assembly definition would also leave room to evaluate variants of the WE 17×17 design without making substantial changes to the initial M&S tool definitions.

Given the wide range of dry cask storage systems currently in use and the abundance of WE 17×17 PWR fuel, a generic 32 PWR canister/basket model will be developed and used for the analyses. Finally, generic transportation packaging configurations must be used wherever practical, and yet the packaging configurations must resemble one that could readily be considered for obtaining a Certificate of Compliance for transport purposes. As such, all components that would influence the structural performance need to be captured and accounted for during model generation. An applicable example of this would be the generic cask configuration with a 32-PWR assembly capacity, referred to as the generic burnup cask (GBC)-32 and described in NUREG/CR-6747 (Wagner 2001). It was previously developed to serve as a computational benchmark for criticality burnup credit studies. This configuration will be adapted for use in this study. The features of the GBC-32 model include a basket with 32 cells that are 365.76-cm-tall and 19.05-cm-wide Boral® (0.0225 g 10B/cm^2) panels between and on the external faces of each cell. The cells have inner dimensions of 22 cm by 22 cm and are spaced on 23.76 cm centers. The cell walls are constructed of a variety of stainless steel. The cells sit 15 cm above the bottom of a stainless steel cask having an inner radius of 87.5 cm and internal height of 410.76 cm. The radial thickness of the side walls is 20 cm, and the cask bottom and lid are 30-cm thick.

2.3 ADDITIONAL INFLUENCING FACTORS

The Steering Team and Implementation Group members clearly identified that integration of the capability to capture the influence of degraded fuel assembly components such as spacer grids, intermediate fluid mixers, and control components is a specific requirement. Also, the influence of interaction between fuel cladding and fuel pellets, and other influencing factors such as DBTT, etc., must also be considered based on the viability and availability of information required to do so. Temperature-dependent material properties must be identified, based on realistic thermal profiles stemming from the anticipated rail-related NCT. The thermal profiles must also be based on prototypic decay heating established from reactor burn, pool cooling, and dry storage durations. Evaluations to establish possible decay heat loading and assembly material property definitions need to consider a spectrum of credible discharge burnup values. In the late 1970s assembly average discharge burnup was around 30 GWd/MTU. At that time, the energy policy in the United States was changed to focus on more efficient utilization of uranium in light water reactors. In response to this change, a number of programs were introduced under the sponsorship of the DOE, the Electric Power Research Institute (Smith et al. 1993), fuel vendors, and utilities. These programs collected data that demonstrated reliable fuel performance at increasing burnup. By the mid-1980s it had been demonstrated that assembly average burnups of 45 GWd/MTU were possible (Smith 1983, Dideon and Bain 1983) and by 1990, it had been demonstrated that assembly average burnup of 57 GWd/MTU was possible (Smith et al. 1993). Since this time, fuel assemblies have been discharged with a burnup of around 55-58 GWd/MTU. The peak rod average burnup limit is 62 GWd/MTU (OECD 2012). Based on this history, values ranging from 30 to 58 GWd/MTU assembly average burnup are viewed to be credible for covering the current UNF candidates, as they include assemblies discharged between 1970 and 2005.

All analytical software packages anticipated to be selected for model development for this initiative will be commercially available and have a proven predictive capability for the intended application.

2.4 QUALITY ASSURANCE

Inputs and analyses will be completely documented, all supporting information traceable, and bases defendable so as to be most useful to the DOE community and mission. In an effort to maintain cost effectiveness and provide the greatest impact within the limited period of performance, the Steering Team deemed that the laboratories' or participants' quality assurance programs are suitable for the current scope of work anticipated. The Implementation Group was in concurrence with this decision.

3. IMPLEMENTATION PROCESS

The implementation process for this initiative drives the methodology, including development and use of analytical models, to evaluate loading and associated mechanical responses of UNF rods and key structural components. The methodology involves coupling M&S with experimental efforts currently under way within UFDC. It includes limited UQ in the form of sensitivity evaluations focused around available fuel and ancillary fuel structure properties exclusively. The work includes, but is not limited to, collecting information via literature review, soliciting contributions from SMEs, developing and demonstrating model capability, performing computational analyses, planning experimental measurement and possible execution (depending on timing), and preparing a variety of supporting documents that will feed into and provide the basis for future initiatives. A flow chart of the process is shown in Figure 3.1. This document represents the level of completion of "Finalize RD&D Implementation Plan (Roadmap)" within this flow chart. As previously identified, the completion date for the work scope identified herein is the end of FY 2013. A Gantt chart showing interacting activities within the specified work scope and associated deliverable milestones is presented in Appendix D.

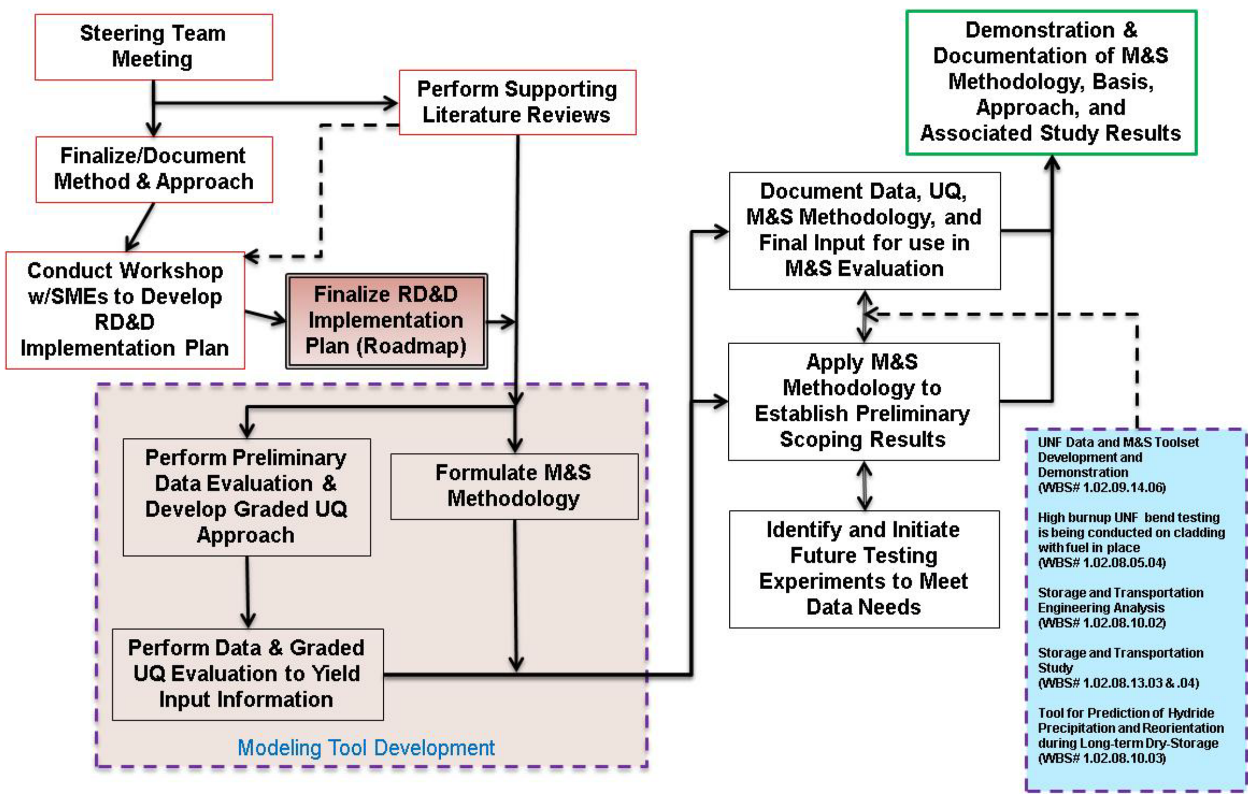


Figure 3.1. Implementation Process Flow Chart

Integration of information from DOE's UFDC M&S and experimental programs is vital to the success of this initiative. Successful integration will mitigate duplication of efforts, potentially backfill previously identified information gaps, and aid in collectively evaluating and identifying additional information gaps that might need to be addressed. Work that will be evaluated for possible integration is described as follows.

Storage and Transportation Engineering Analysis (WBS 1.02.08.10.02)

A number of storage and transport systems have been studied under this work breakdown structure (WBS) element. Pre-test temperature distribution predictions have been, and will continue to be performed for full-storage systems and associated components. The purpose of this work is to support storage site inspections. Thermal and stress profile work has been performed and is under way for prototypic storage and transport systems as well. Thermal and structural sensitivity evaluations are also being performed to establish and support the understanding of degradation mechanisms.

Storage and Transportation Study (WBS 1.02.08.13.03 & .04)

Shaker table testing, M&S, and evaluations involving a standard WE 17×17-type PWR are being performed under these WBS elements. Preliminary pre-test modeling has been performed to aid in determining instrumentation locations. Some pre-test simulations are under way and others are pending finalization of mockup material property verification. This work will involve more discrete modeling of mockup fuel pins such as explicit definition of the materials selected to mimic the cladding and fuel pellets. Some supporting scoping cask body influence evaluations need to be performed for this to directly tie to the initiative described herein. Physical testing is pending mockup completion. Algorithms converting LS-DYNA models into PRESTO (Livermore Software Technology 2007, SNL 2011) models are being considered for construction. It is envisioned that these algorithms will aid in providing baseline analyses for future benchmarking and validation of modeling techniques involving PRESTO as well as cross-comparison between LS-DYNA and PRESTO. This work may serve as part of the validation of the M&S.

UNF Data and M&S Toolset Development and Demonstration (WBS 1.02.09.14.06)

Radiation and shielding code SCALE (ORNK 2011), and the thermal hydraulics code COBRA-SFS (Michener et al. 1995), are being coupled to yield an assessment tool that can be used to establish fully detailed radionuclide inventory and highly resolved shielding, dose, and temperature distribution characteristics for a specific UNF package. This work is being conducted under Nuclear Fuels Storage and Transportation (NFST) sponsorship and direction.

High Burnup UNF Bend Testing Conducted on Cladding with Fuel in Place (WBS 1.02.08.05.04)

This testing was being conducted through FY 2012 and a portion of FY 2013 under NRC sponsorship and direction on H.B. Robinson WE 15×15 rod segments with 63-67 GWd/MTU average assembly burnup. UFDC is collaborating and funding follow-on research in FY 2013.

Tool for Prediction of Hydride Precipitation and Reorientation during Long-term Dry-Storage (WBS 1.02.08.05.01 & 1.02.08.10.03)

Hydride reorientation and cladding degradation modeling as well as other supporting cladding testing and experimentation are under way and will be incorporated as the information becomes available.

High Burnup UNF Cladding Material Properties (WBS 1.02.08.05.01)

Testing and analysis being conducted at Argonne National Laboratory (ANL), using existing equipment and available high burnup PWR cladding specimens (Zircaloy-4, ZIRLO™, and M5®), to determine the DBTTs and mechanical properties for the PWR high burnup cladding alloys.

4. WORK SCOPE

The sections that follow define how subtasks of this initiative will be conducted and identify specific actions/activities to be performed based on SME recommendations.

4.1 LITERATURE REVIEW

Literature reviews have been initiated and will be coming to conclusion in the months to come. The objective is to establish the quantity and type of information available in three specific areas germane to this initiative. They are mechanical loading during storage and transportation, system and UNF material properties, and relevant M&S techniques. In each area the following will be established by a representative team:

- mechanical loading that UNF cladding and key structural components could be subjected to during NCT and NCS. Mechanical loading inputs will take into account the possible types of conveyances and packaging configurations targeted for road and rail transport.
- applicable material properties for storage, transfer, and transport packages, conveyances, and properties of UNF cladding and ancillary components. However, transport-related items remain the primary focus.
- relevant M&S methodologies and their accompanying evaluation techniques commensurate with the objectives. Identify, gather, and consolidate literature on M&S (materials models, possible failure criteria and constitutive models, methodologies, computational frameworks, benchmark studies, and possible methodology validation information, etc.).

The resulting documented information will be used to establish databases to support work performed under this as well as future programs and tasks.

4.1.1 TRANSPORTATION AND STORAGE LOADING, BOUNDARY CONDITIONS, AND APPLICABLE REGULATIONS

A literature review is being performed to identify all loadings and governing regulations applicable to establishing preliminary performance characteristics of UNF during normal conditions of storage and transport in the United States. In particular, available measured data relative to mechanical loads on cladding and key assembly structural components are being identified, with an emphasis on data needed to perform analyses (e.g., input loads at a transportation package surface), and data that could be used to verify analysis codes and validate predictions. Encompassing data and information, as well as proposed model inputs, are being documented. Also, specific recommendations based on the information available will be included, and guidance regarding implementation in future M&S evaluation work performed under this, as well as possible future programs, will be provided. Where possible, data gaps in supporting information and boundary conditions that may exist, will be identified. Recommendations for possible sources include but are not limited to:

- Normal conditions of transport (transportation loading, boundary conditions, and applicable regulations for rail and over the road)
  - 10 CFR 71
  - Possible Conveyance Candidate(s) – (special nuclear fuel conveyances will all need to meet AAR S-2043 [AAR 2003] – Performance Specification for Trains Used to Carry High Level Radioactive Materials)
  - Sandia National Laboratories (SNL) study of shock and vibration in truck transport (Magnuson 1978)
  - Possible data and information from TTCI
  - Transport loading time histories from fresh fuel transports
- Normal Storage Conditions
  - 10 CFR 72
  - Central Research Institute of Electric Power Industry (CRIEPI) Seismic Simulations
- Possible loading incurred during transfer
- Other available data to validate analysis models, methodologies, and predictions that may exist in the literature.

As previously stated, storage and over-the-road transportation information is being collected during the literature review portion of this initiative. However, rail transport remains the primary focus of this initiative at least until an order of magnitude deterministic assessment on ductility requirements of UNF under this mode of transport is fully realized.

4.1.2 MATERIAL PROPERTIES

A literature review is being performed to identify material properties relevant to moderate to high burnup UNF, as well as associated storage and transport components, to establish a credible database to feed into M&S tools for determining cladding and fuel assembly specific mechanical performance characteristics. Applicable data and information, as well as inputs yielded from this literature review will be documented in the weeks to come. Also, specific recommendations based on the information available will be documented, and guidance regarding implementation in M&S evaluation work performed under this, as well as possible future programs, will be provided in the documentation. Data gaps in supporting information and boundary conditions that exist will be identified. Possible sources of information include, but are not limited to:

- Documentation pertaining to storage and transport cask/packaging systems such as safety analysis reports and vendor data sheets and specifications
  - established preliminary focus on viable high-capacity PWR (≥32 assembly payload) transportation package candidate previously defined as the GBC-32
- Available low/moderate/high burnup UNF information

- primary focus on PWR WE 17×17 type assembly configuration with Zircaloy-4 cladding but should also include other smaller-rod diameter PWR fuel configurations (e.g., WE 15×15) and associated cladding candidates
- assembly average burnup values ranging from 30 to 58 GWd/MTU are targeted for data collection
- cladding test data and material information for degraded fuel assembly components such as spacer grids, intermediate fluid mixers, and control components
- realistic thermal profiles stemming from anticipated rail-related NCT and credible decay heat loading.

Understanding performance characteristics of UNF under loads stemming from NCS, transfer, and NCT and establishing cumulative effects are necessary to determine fuel condition for retrievability purposes. However, information on all possible loading influences is beyond the scope of this exercise. As such, limited information pertaining to any configurations other than normal transport will be collected. An initial fuel material state will be assumed for the pre-transport condition, based on the best information available to date.

4.1.3 MODELING AND SIMULATION

A literature review identifying modeling inputs and previously applied M&S methodologies is under way to establish a credible basis and relevant analytical approach for determining used fuel mechanical performance. Modeling methodologies considered and identified herein are based on loading of fully featured representations with highly resolved meshing of UNF assemblies and relevant assembly components. Applicable data, information, and previously implemented modeling methodologies as well as inputs yielded from this literature review are being documented. Also, specific recommendations based on the information available will be documented and guidance regarding implementation in future M&S evaluation work performed under this, as well as possible future programs, will be provided. Recommendations for sources of information include but are not limited to identifying previous

- modeling inputs and constitutive relations
- M&S methodologies and modeling approaches that are relevant to the current initiative/scope
- codes and computational packages that are relevant to the current initiative/scope (Note: code candidates must be commercially available [e.g., ANSYS, Abaqus, LS-DYNA, etc.])
- verification and validation, and benchmarking studies
- evaluations of codes and computational package capabilities that apply.

4.2 SOLICITATION OF INPUT FROM SUBJECT MATTER EXPERTS

An Implementation Group composed of national laboratory, DOE, NRC, and TTCI staff met in February 2013 to identify the specific inputs needed to develop a focused overall RD&D project implementation plan, based on input from SMEs selected for their expertise related to this effort. The Implementation Group participants are listed in Appendix B. SME input and recommendations were solicited during this meeting. Information stemming from this solicitation will be documented as part of the literature reviews and supported documentation developed during this initiative. It will also be used to provide supporting direction and interpretation of supporting input data and M&S definition. Members of the contributing SMEs, as well as other SMEs were selected to form a core implementation team that will conduct the execution of the implementation plan as documented herein. Near the conclusion of the working meeting, responsibilities were established and deliverable dates were communicated, as reflected in Appendix D.

4.3 METHODOLOGY DEVELOPMENT

The Implementation Group received preliminary briefings on the status of the ongoing literature reviews to aid them in decision making. Three task teams were formed from members of the Implementation Group and are referred to as the Transportation Loading (TL), Material Properties (MP), and M&S Teams, respectively. Preliminary literature findings, available information, and recommendations established by the SMEs that participated in the Implementation Group meeting were considered. A methodology for generation of the M&S evaluation tools was developed based on the interactions of the three task teams. The Implementation Team was established based on a refined set of TL, MP, and M&S Teams as well as other selected SMEs. Biographies of the Implementation Team members are presented within the laboratory-specific statements of work (SOWs) included in Appendix C. The methodology development path that will be employed by the Implementation Team for assembling the foundation for constructing models for performing adequately resolved deterministic structural evaluations is described in the subsections below. This foundation for model construction allows for future upgrades/modifications/alterations for performing additional simulations, and readily accommodates the incorporation of information as it becomes available.

4.3.1 APPROACH FOR OBTAINING TRANSPORTATION LOADING DATA

The objective of the TL Team is to evaluate options and develop an approach for obtaining transportation loading data that will be used by the M&S Team to determine the effects of the shock and vibration associated with the rail-related NCT on UNF assemblies and rods. Ideally, these shock and vibration data would be at the fuel assembly or fuel rod level. However, previous studies have typically collected shock and vibration data at the transportation cask or conveyance (i.e., railcar or truck bed) level, so it is likely that the M&S Team will have to use a transfer function to estimate effects at the UNF rod level.

The TL Team evaluated options for obtaining transportation loading data. The team received a briefing on the status of the TL literature review. As a first step in the literature review, NRC guidance on evaluating the vibration and shock normally incident to transport was evaluated. In Section 2.5.6.5 of NUREG-1609, *Standard Review Plan for Transportation Packages for Radioactive Material* (NRC 1999), three documents are cited:

- NUREG/CR-6007, *Stress Analysis of Closure Bolts for Shipping Casks* (Mok et al. 1993)
- NUREG/CR-2146, *Dynamic Analysis to Establish Normal Shock and Vibration of Radioactive Material Shipping Packages, Quarterly Progress Report, Final Summary Report* (Fields 1983)
- NUREG/CR-0128, *Shock and Vibration Environments for a Large Shipping Container During Truck Transport (Part II)* (Magnuson 1978).

In addition, transportation cask safety analysis reports (SARs) were reviewed to determine the industry approach for evaluating vibration and shock because of the NCT during transportation cask licensing. The transportation cask SARs reviewed were those that are licensed to transport UNF from shutdown sites, such as the MP187, NAC-STC, NAC-UMS UTC, TS125, and the HI-STAR 100 and HI-STAR HB. In these SARs, the applicants typically assumed that a 2g vertical acceleration was bounding (ANSI N14.23). In addition, for the MP197 transportation cask, which is not licensed for transporting UNF from the shutdown sites, the applicant used the shock and vibration data from NUREG-766510 (Magnuson and Wilson 1977).

The literature review consisted of searches conducted of various sources and databases and extracting papers and reports that were relevant to the shock and vibration associated with the NCT.

It should be noted that these searches concentrated on papers and reports related to the transport of UNF, radioactive waste, or other radioactive material, in part because of the weights associated with UNF transportation casks, which are typically in the range of 200,000 to 320,000 lb., and because the weight of the object (e.g., a transportation cask) upon a conveyance directly affects the magnitude of vibrations and shock imparted to the contents of the object.

The preliminary results of the literature review were as follows:

- 198 documents have been collected.
- There are few recent studies that relate to the shock and vibration associated with NCT.
- Most studies that are related to the shipment of UNF, radioactive waste, or radioactive material were published in the 1970s and 1980s.
- Sanders et al. (1992) contains a summary of tests published in the 1970s and 1980s.
- One study (Prulhiere and Israel 1980) was found where an assembly was instrumented inside a UNF transportation cask.

- No studies were found that evaluated current generation UNF rail transportation casks, which have weights in the range of 200,000 to 320,000 lb.
- The center of gravity for current generation UNF rail transportation casks will likely be higher than for transportation casks used in previous tests.
- No shock and vibration studies using railcars that meet AAR Standard S-2043 were found (AAR 2003).
- Over-the-rail tests typically involved regular trains (40-50 cars), as opposed to short dedicated trains as will be implemented for rail-related UNF transport.

Table 4.1 summarizes the key attributes of the most relevant studies. Figures 4.1, 4.2, and 4.3 illustrate bounding acceleration shock spectra previously determined for truck transport, rail transport, and rail coupling, respectively (Sanders et al. 1992). Based on the results of the literature review, the data currently used to characterize the shock and vibration associated with NCT by rail appear to overestimate the shock and vibration encountered during transport of a transportation cask on an AAR Standard S-2043 compliant railcar (AAR 2003), and would be unrealistic. In addition, the cask weights used to derive the current data are non-representative of current generation cask weights. For these reasons, the TL Team evaluated options for obtaining additional shock and vibration data that would then be used by the M&S Team to model the effects of rail-related NCT.

Table 4.1. Summary of Relevant Shock and Vibration Studies

Reference	**Rail/Truck**	**Cargo**	**Comments**
Foley 1966a	T	15-ton radioactive materials cask	Transport (unloaded) from Ft. Eustis, VA to Wilmington, DE. Transport (loaded) from Wilmington, DE to Albuquerque, NM. Data presented in multiple formats. See also Foley (1966b) and Bryan (1965).
Foley and Gens 1971a	R/T	15-ton used nuclear fuel cask	Evaluates light cargo, particularly for the rail-shock response spectrum. Dynamic environments were similar for truck and rail. Truck transport was from Oak Ridge, TN to Paducah, KY. Rail transport was from Paducah, KY to Oak Ridge, TN. See also Foley and Gens (1971b) and Gens (1970).
Magnuson and Wilson 1977	R/T	No load to 15-ton cargo, 5-ton cargo for rail coupling tests	Evaluates light cargo. Provides a summary of earlier test results, including seven different truck and tractor-trailer configurations. Observed that shock decreases as cargo weight increases. Provides shock response spectra, bounding single pulse representation, and rail-coupling data.
Magnuson 1977	T	22-ton used nuclear fuel cask	Provides shock response spectra and single pulse representation. Two axle 35-foot trailer with air suspension. Cask transported from Mercury, NV to Albuquerque, NM. Two sets of accelerometers on the container and four sets of accelerometers on the structure supporting the container.
Magnuson 1978	T	28-ton used nuclear fuel cask	Provides shock response spectra and single pulse representation. Three axle 40-foot trailer with spring suspension. Cask transported from Mercury, NV to Albuquerque, NM. Two sets of accelerometers on the container and four sets of accelerometers on the structure supporting the container.

Table 4.1. (contd.)

Reference	**Rail/Truck**	**Cargo**	**Comments**
Magnuson 1980	R	40-ton to 70-ton used nuclear fuel casks, rail coupling tests	Evaluates different railcar designs. Provides shock response spectra and single pulse representation. Evaluates heavy cargo. The railcars were equipped with either standard draft gear, hydraulic end-of-car draft gear, or a sliding center sill cushion underframe. See also Petry (1980).
Prulhiere and Israel 1980	R/T	TN 12 (100 MT)	Instrumented assembly inside a used nuclear fuel cask. Presents maximum acceleration values.
Magnuson 1982	R	50-ton used nuclear fuel cask	Provides shock response spectra. Includes no equivalent pulse. Evaluates heavy cargo. Cask transported from Denver, CO to Albuquerque, NM.
Dalziel et al. 1986	R/T	32-MT truck cask and 68-MT rail cask (simulated)	Canadian Shock and Vibration Program consisted of: (a) road transportation field measurements of normal vibration and transients; (b) rail transportation field measurements of normal vibration, transients, and railcar coupling impact; (c) modal testing of the fuel transport containers; (d) impact and fatigue testing on irradiated fuel and unirradiated fuel; (e) analytical modeling of the road and rail transportation modes.
Glass and Gwinn 1986	T	NuPac 7D-3.0 cask (15.7 tons) CNS 14-170 cask (18.5 tons)	Road simulator was used. Predictive analytical method was developed. Includes no shock response spectra or equivalent pulse.

Table 4.1. (contd.)

Reference	**Rail/Truck**	**Cargo**	**Comments**
Glass and Gwinn 1987	T	TRUPACT-I	Acceleration-time signals were reduced to power spectral densities. These give the vibrational energy as a function of shock response spectra. Includes no equivalent pulse. The Type B package was evaluated.
Pujet and Malesys 1989	R/T	NTL 8/3 (36 tons) NTL 11 (80 tons)	Maximum accelerations at the trunnions presented. NTL 8/3 truck cask shipped from La Hague, France to Tihange, Belgium, 1600 km. NTL 11 rail cask shipped from Valognes, France to Wurgassen, Germany, 2360 km. See also Cory (1991).
Gwinn et al. 1991	T	CNS 14-170 (24 tons) CNS 3-55 (33 tons)	Power spectral densities presented.

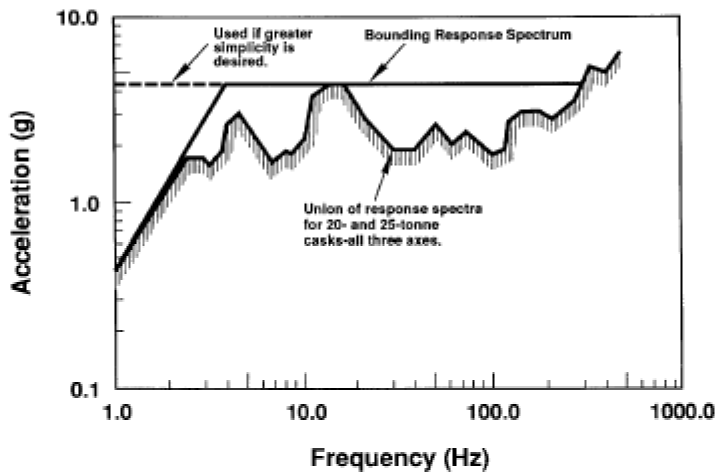


Figure 4.1. Truck Bounding Acceleration Shock Response Spectrum for 3% Damping on all Three Axes (Sanders et al. 1992)

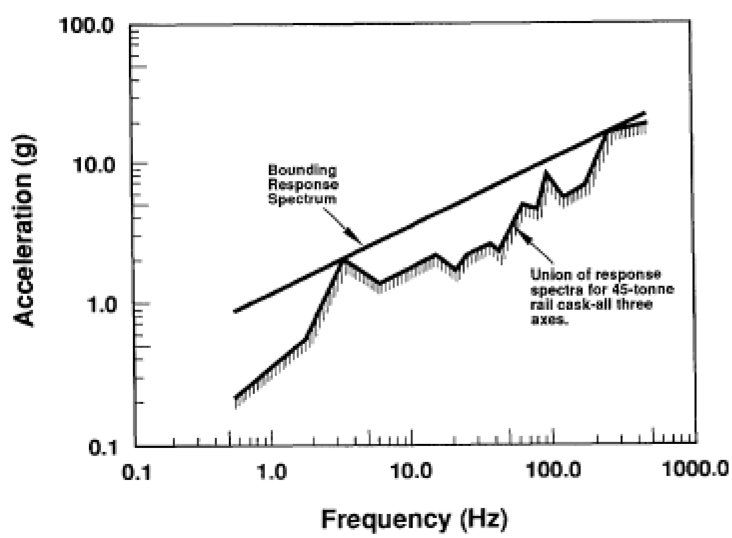


Figure 4.2. Rail Bounding Acceleration Shock Response Spectrum for 3% Damping on all Three Axes (Sanders et al. 1992)

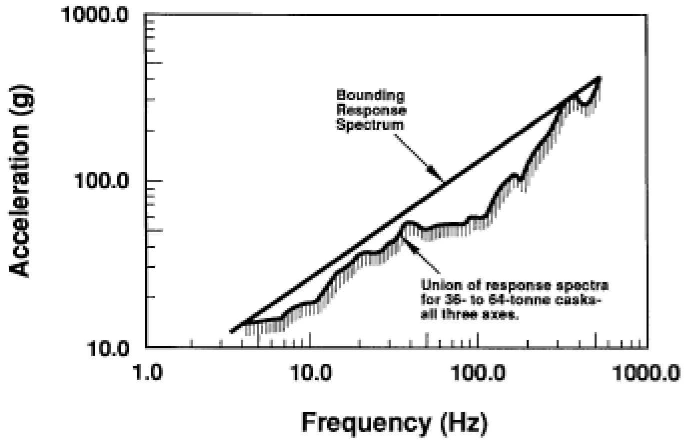


Figure 4.3. Rail Coupling Bounding Acceleration Shock Response Spectrum for 3% Damping on all Three Axes (Sanders et al. 1992)

The TL Team next received a briefing on the capabilities of TTCI, the requirements of AAR Standard S-2043 (AAR 2003), and the modeling that has been conducted using the NUCARS® computer code (TTCI 2013). One key result of this briefing was that TTCI has tested and modeled a large number of railcars for clients in the railroad industry, including the railcar that is designed to carry the U.S. Navy M-290 transportation cask. However, the results of testing and modeling are often proprietary, and the representatives of TTCI stressed the need to get client permission before these data could be used for other purposes.

The TL Team then discussed and evaluated options for obtaining shock and vibration data that would then be used by the M&S Team to model the effects of the NCT. The following options were discussed and evaluated:

- **Use existing shock and vibration data (see Table 4.1 or the performance criteria contained in AAR Standard S-2043, Tables 4.1 and 5.1 [AAR 2003]).** This option was downgraded because the cargo weights used in previous studies would not be representative of the weights of current used nuclear fuel casks (200,000 to 320,000 lb.). In addition, previous studies typically report maximum acceleration values, which would not be a realistic representation of the shock and vibration associated with the rail-related NCT. Previous tests also did not involve railcars that meet AAR Standard S-2043. The option of using the performance criteria contained in AAR Standard S-2043 was downgraded because these performance criteria represent maximum values, not expected values (AAR 2003).
- **Instrument the U.S. Navy M-290 railcar during the upcoming Pennsylvania-to-Colorado transit.** This option was downgraded because of the need to get permission from the U.S. Navy to instrument the railcar and because the option may not be practical, based on the project schedule.

- **Instrument the U.S. Navy M-290 railcar at TTCI during AAR Standard S-2043 testing.** This option was downgraded because of the need to get permission from the U.S. Navy to instrument the railcar and because the option may not be practical, based on the project schedule.
- **Use archived data available at TTCI for a representative railcar.** The degree that the railcar is representative of a railcar carrying a UNF transportation cask and support structures could be an issue for this option.
- **Use NUCARS® simulated railcar data and reference NUCARS® validation cases to establish the pedigree of these data.** This option was downgraded because it would use simulated, not actual, data. However, it remains a viable option for obtaining shock and vibration data.
- **Instrument the Private Fuel Storage mass currently at TTCI, and test at TTCI. The Private Fuel Storage mass would not be representative because of weight.** Also, a railcar would need to be obtained to use in testing, and consequently this option may not be practical, based on the project schedule.
- **Instrument a heavy duty railcar that is loaded with a mass that would be similar to a used nuclear fuel cask and support structures.** This option was not viewed as practical, based on the project schedule.

Based on this evaluation, the TL Team developed the following approach for obtaining shock and vibration data to be used in modeling the effects associated with NCT:

- Use archived data available from TTCI for a representative railcar. TTCI will choose the railcar based on the likely characteristics of railcars used to carry used nuclear fuel casks. The TL Team will develop the likely characteristics of the railcars.
- Compare the TTCI data to the data from the studies listed in Table 4.1 to verify that it is representative of data associated with rail transport of relatively light loads on a railcar that did not comply with AAR Standard S-2043 (AAR 2003).
- Conduct NUCARS® simulations to match the archived data. TTCI will perform the NUCARS® simulations.
- Simulate a railcar carrying a generic used nuclear fuel cask (such as the GBC-32 identified above) and support structures using NUCARS®. The railcar will be a railcar that would likely meet AAR Standard S-2043 (AAR 2003). TTCI will perform the NUCARS® simulations. The TL Team will develop the characteristics of the UNF cask, support structures, and railcar with assistance from the M&S Team to determine the interface for the acceleration data.
- Provide the results to the M&S Team.
- If possible, conduct the shaker table tests outlined in McConnell (2012) using spectra derived from the NUCARS® simulations. Sandia National Laboratories will perform the shaker table tests.

- In parallel, pursue instrumenting the U.S. Navy M-290 railcar. This could be done for either the Pennsylvania-to-Colorado transit or during testing of the M-290 railcar at TTCI.

4.3.2 APPROACH FOR DEFINING APPLICABLE MATERIAL PROPERTIES

A document containing the material properties to be used for this project's M&S tasks for each of the materials used in a UNF assembly and the storage and transportation container is being assembled by members of the MP Team. These material properties will represent the current best-estimate values for the needed material. Material properties for the surrogate assembly used in the shaker table test will also be included in the provided data sets. The conditions experienced by the fuel assemblies in-reactor are very challenging and the combination of fuel burnup, damage due to fast neutron flux, and long service operation at high temperature (280°C-320°C) and in high pressure (15.5 MPa) conditions while immersed in water, lead to large substantial changes in the properties of the fuel, cladding, and other assembly components.

Material properties correlations will include best-estimate fits to both unirradiated and irradiated data. The cladding properties in particular have been used successfully in the Pacific Northwest National Laboratory (PNNL)/NRC fuel performance codes, FRAPCON-3 (Geelhood et al. 2010a) and FRAPTRAN (Geelhood et al. 2010b) to model in-reactor behavior and transient performance, respectively.

There is variation in the available data, particularly irradiated data. Additionally, the impact of long-term dry storage at 100°C-400°C on these materials is uncertain. In order to address these uncertainties, standard deviations and uncertainty distributions will be included with the appropriate material properties data sets.

The material properties document will consist of the following sections:

- Reference assembly
- Condition of irradiated fuel
- Cladding material properties
- Fuel material properties
- Grid material properties
- Cask material properties
- Material properties for surrogate assembly to support the shaker table work.

The following sections will briefly describe the material properties, their uncertainties, and their applicability to NCT of UNF.

4.3.2.1 REFERENCE ASSEMBLY

The reference assembly for this study is a generic 17×17 fuel assembly for a Westinghouse PWR. This assembly can be seen in Figure 4.4.

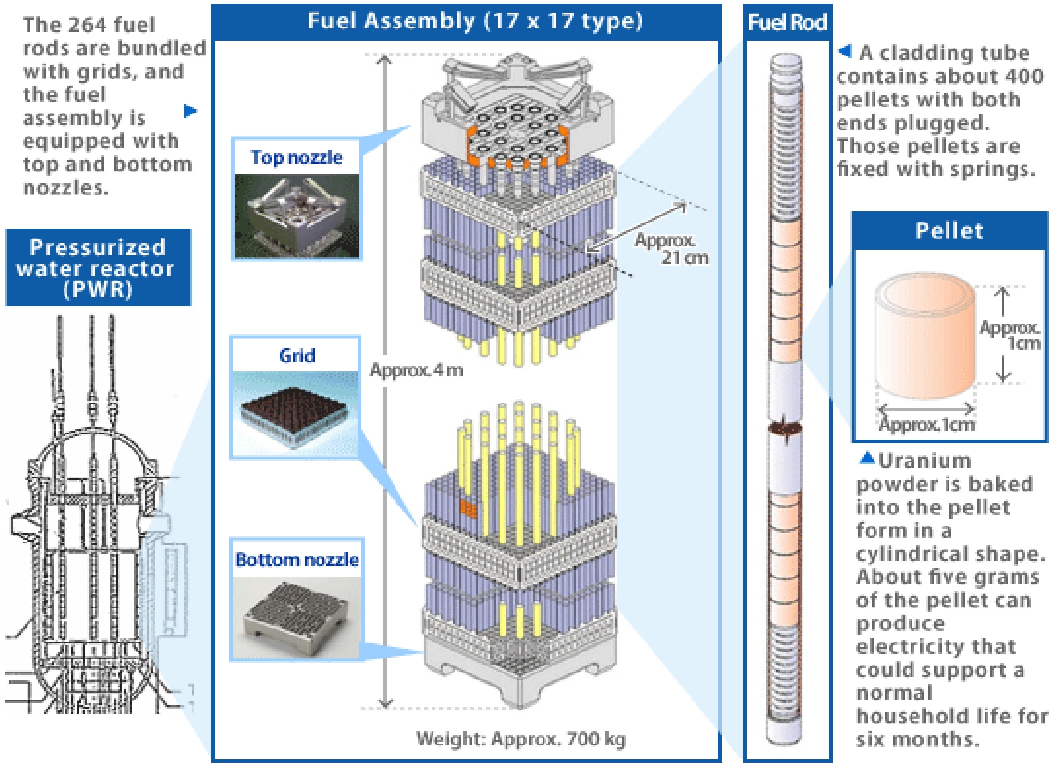


Figure 4.4. Basic Structure of PWR Fuel Assembly (Nuclear Fuel Industries 2013 http://www.nfi.co.jp/e/product/prod02.html#b) Reprinted with permission

Table 4.2 shows the parameters and materials for this fuel assembly.

Table 4.2. Parameters and Materials for Reference Fuel Assembly. Data taken from ENUSA Industrias Avanzadas, S.A. (2013)

Fuel Assembly Parameters	
Array	17×17
Number of Fuel Rods	264
Number of Guide Rods	24
Number of Instrument Tubes	1
Number of Grids (standard and IFM)	12
Material Properties: Fuel Rod	
Fuel Pellet	UO_2
Cladding	Zircaloy-4
Spring	Stainless Steel
End Plugs	Zircaloy-4
Material Properties: Assembly	
Grids	Zircaloy-4 or Inconel-718
Upper and Lower Nozzles	Stainless Steel
Spring	Inconel-718
Thimble Tubes	Zircaloy-4
Instrument Tube	Zircaloy-4

4.3.2.2 CONDITION OF IRRADIATED FUEL

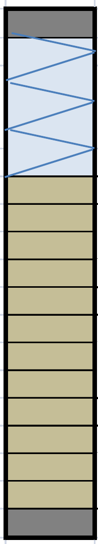
In this section the various parameters that are used to define the damage to the fuel and the cladding are briefly described. Burnup is the parameter used for the damage in the fuel. Various burnup units and the conversion between these units will be given and formally documented. Additionally, fuel and assembly peaking factors will be described to show how assembly average burnup, rod average burnup, and pellet burnup relate to one another. Fast neutron fluence is the parameter used for the damage in the cladding. These units will be described and a typical conversion from fast neutron fluence to displacements per atom (DPA) will be provided. The time in-reactor causes corrosion on the outer surface of the cladding and hydrogen pickup in the cladding. This section will describe the general trends in corrosion and hydriding as well as the impact of these phenomena on cladding properties.

Finally, to assist the M&S Team in defining the condition of fuel rods that have been discharged at various burnups, tables of typical values of damage for rods at various burnup levels that have been produced for previous initiatives will be provided in this section. The expected axial variation in these parameters is shown at one-foot intervals. Table 4.3 gives an example of one of these tables. In addition to these parameters, the effective fuel cladding to fuel pellet gap as a function of temperature will also be included. Although a fuel/clad bonding layer exists at high burnup, an effective gap from opening of cracks in the pellet is useful for modelers. These parameters, along with the calculated component temperatures will be used as inputs to the material property models given in the document.

Table 4.3. Expected Initial Conditions for Reference Fuel Assembly at 50 GWd/MTU

PWR Westinghouse 17x17 Fuel Assembly

Assembly Average Burnup	50 GWd/MTU
Peak Rod Average Burnup	53.7 GWd/MTU
Cladding material:	Zircaloy-4
Peak Rod Fast Neutron Fluence	8.99E+25 n/m²



Gas Composition 88.2% He, 10.0% Xe, 1.8% Kr — Total void volume = 10.81 cm³ — Total Gas = 1.9e-2 Moles

	Local Burnup GWd/MTU	Local Fast Neutron Fluence n/m²	Corrosion layer Thickness µm	Metal Consumed µm	Hydrogen Concentration ppm	Excess Hydrogen @ 20°C ppm	Excess Hydrogen @ 100°C ppm	Excess Hydrogen @ 200°C ppm	Excess Hydrogen @ 300°C ppm	Excess Hydrogen @ 400°C ppm
Upper Plenum										
node 12	41.2	6.89E+25	56.8	36.4	472	472	471	459	407	272
node 11	52.1	8.73E+25	71.2	45.6	593	593	592	580	528	393
node 10	57.3	9.58E+25	72.9	46.7	608	608	607	595	543	408
node 9	58.2	9.74E+25	59.8	38.3	497	497	496	484	432	297
node 8	58.2	9.75E+25	50.9	32.6	423	423	422	410	358	223
node 7	58.1	9.73E+25	41	26.3	341	341	340	328	276	141
node 6	58.1	9.72E+25	35.1	22.5	292	292	291	279	227	92
node 5	58	9.70E+25	28.6	18.3	239	239	238	226	174	39
node 4	57.9	9.69E+25	23.5	15.1	197	197	196	184	132	0
node 3	56.6	9.47E+25	17.9	11.5	152	152	151	139	87	0
node 2	49.9	8.36E+25	12.4	7.9	109	109	108	96	44	0
node 1	39.3	6.58E+25	7.1	4.6	66	66	65	53	1	0

4.3.2.3 CLADDING MATERIAL PROPERTIES

Cladding properties that will be necessary for M&S of the fuel assemblies will predominantly come from models developed for FRAPCON-3 (Geelhood et al. 2010a) or MATPRO (Siefken et al. 2001). Comparisons of model predictions to data will be provided with irradiated data shown separately from unirradiated data. As appropriate, the calculated uncertainty and the distribution on that uncertainty will also be provided.

It has been noted that the data these models are based on were taken from cladding that was recently (approximately 6 months) removed from the reactor spent fuel pool after some period of cooling. It is not currently known if the vacuum drying operation at temperatures up to 400°C or the 20 or more years spent in dry cask storage at 100°C-400°C will result in annealing of irradiation damage or in grain growth. Some experimental data show the microhardness of irradiated Zircaloy will decrease about 7 percent after being held at 330°C-420°C (Ito et al. 2004). The primary cause of increase in yield strength with irradiation is the formation of dislocations. These dislocations may be annealed our after some period of time at elevated temperature. For the parameters of interest, it is not expected that extended time at this temperature will greatly affect the properties. For example, the elastic modulus is not a strong function of fluence and the increase with fluence saturates at low values, such that some for irradiation induced dislocations annealing would not result in a large change in modulus. Although the yield stress changes considerably with irradiation, the effect of irradiation quickly saturates at low fluence levels. Finally, there is not much evidence that would support the annealing of dislocations and grain growth in metals at such low temperatures.

Table 4.4 shows the cladding properties that will be provided for M&S Team application as a function of temperature and fast neutron fluence. Also shown is the source for the data, the uncertainty that is provided, and the number of available data points that were used to develop the model.

Table 4.4. Cladding Properties to be Provided

Property	**Source**	**Uncertainty**	**Number of Data**
Elastic Modulus	FRAPCON/MATPRO	σ calculated from data	125
Yield Stress	FRAPCON	σ calculated from data	432
Ultimate Tensile Strength	FRAPCON	σ calculated from data	419
Uniform Elongation	FRAPTRAN	σ calculated from irradiated data (biaxial and uniaxial tests)	236
Total Elongation	Discussion Only	NA	NA
Density	FRAPCON/MATPRO	Physical property	NA
Fracture Toughness	PNNL Model	σ calculated from data	519
Wear and Fretting	Analysis Limit Provided	NA, design dependent	NA
Fatigue	O'Donnell and Langer (1964)	Lower bound failure curve	100+

The fatigue curve presented by O'Donnell and Langer (1964) represents the best available failure criterion for fuel rods under NCT because of the vibration normally incident to transport. This is the curve that is most often used by the fuel vendors to perform their fatigue analysis for their in-reactor safety analyses. It is noted that although the data used to develop these fatigue limits are taken on irradiated samples under several different stress conditions, the samples are from plates rather than tubes, and there was no hydrogen in the samples, as would be expected in cladding tubes. However, it is expected that this fatigue curve will be generally representative of the behavior of cladding from spent fuel. As initial stress intensities and expected stress cycles are calculated from the M&S task, this fatigue curve should be used to assess the potential for failure. If this curve shows operation far from the failure limit, then that will provide adequate demonstration that the rods will not fail because of cyclic loading. If the M&S results show operation near this failure curve, then further experimental work should be performed to produce a more prototypic cyclic loading failure curve.

4.3.2.4 FUEL MATERIAL PROPERTIES

This section describes the fuel properties that will be necessary for the M&S portion relating to the UNF assemblies. Most of these data sets come from MATPRO as FRAPCON and FRAPTRAN do not consider the impact of the strength of the pellet since it is much greater than that of the cladding. Table 4.5 shows the cladding properties that will be provided for the M&S work as a function of temperature and burnup. Comparisons of model predictions to data will be provided with irradiated data shown separately from unirradiated data. As appropriate, the calculated uncertainty and the distribution on that uncertainty will also be reported.

Table 4.5. Properties Provided for Cladding

Property	**Source**	**Uncertainty**	**Number of Data**
Elastic Modulus	MATPRO - unirrad. Laux (2012) – irrad.	σ calculated from data	78 unirradiated 41 irradiated
Fracture Strength	MATPRO - unirrad.	σ calculated from data	~100
Density	FRAPCON/MATPRO	Physical Property	NA
Fuel/Clad Bonding Layer	No model. Bonding layer is 10-20 μm thick and a discussion of strength of this layer is provided	Range	Small database, no strength data

One of the largest gaps in the properties above is the strength of the fuel/clad bonding layer. This parameter is a very difficult quantity to measure directly. However, sensitivity studies may be performed under the M&S effort to determine the impact, if any, of the fuel clad bonding layer strength on the response of the fuel to rail-related NCT.

4.3.2.5 GRID MATERIAL PROPERTIES

As seen in Table 4.2, the grids may be made of either Zircaloy-4 or Inconel-718. This section will provide a discussion on how the Zircaloy-4 cladding models may be used for Zircaloy-4 grids. For example, the expected fast neutron fluence, corrosion thickness, and hydrogen content for Zircaloy-4 grids will be specified. These parameters may then be used as input to the Zircaloy-4 cladding models.

For Inconel-718, the basic mechanical properties as a function of temperature will be provided for unirradiated sheet. Table 4.6 shows the Inconel-718 properties that will be provided as a function of temperature. The properties of Inconel-718 will likely change with irradiation, but no data are currently available for irradiated Inconel-718.

Table 4.6. Properties Provided for Unirradiated Inconel-718 Sheet

Property	Source	Uncertainty
Elastic Modulus	Product Reference Guide, Special Metals Corporation	Not Calculated
Yield Stress		
Ultimate Tensile Strength		
Total Elongation		
Density		

The biggest issue with the grids is the force with which the spring in the grid contacts the UNF rod. The geometry of this situation is shown in Figure 4.5. The design of these springs and dimples varies significantly from assembly to assembly between vendors and even for a single vendor from year to year as the design evolves. Also, it is not known how much relaxation of the spring may occur during in-reactor operations. In order to assess the importance of these springs on the response of the assembly, each spring will be modeled as a simple spring and sensitivity studies will be performed to evaluate the importance of the properties of these springs. This study will reveal if further testing is required on the irradiated grids in order to better model the response of the assembly to conditions of normal transportation.

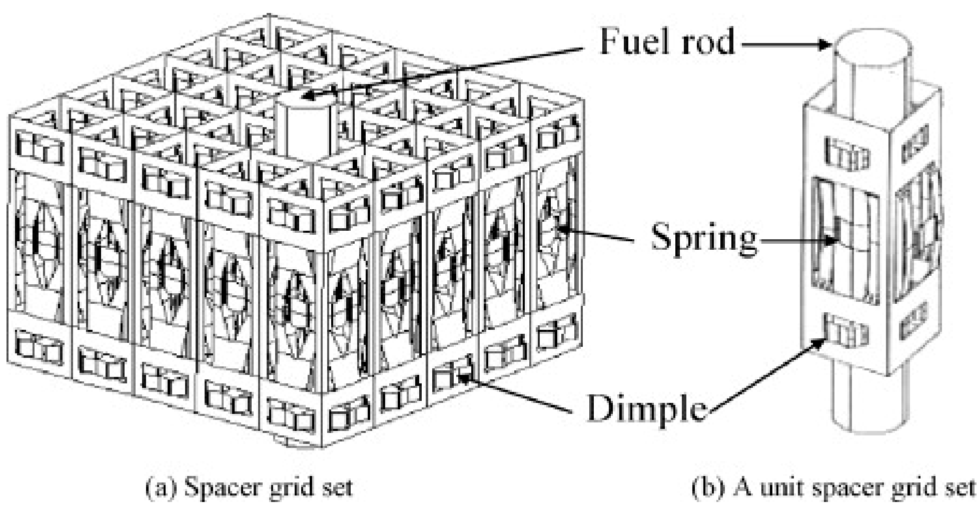


Figure 4.5. Typical Grid with Springs and Dimples to Retain the UNF Rods (Shin et al. 2008) Reprinted with permission

4.3.2.6 CASK MATERIAL PROPERTIES

A generic cask, canister, and basket system that holds 32 assemblies has been selected as the reference packaging for this study (GBC-32). The canister is shown in Figure 4.6. It is assumed that the basket, basket support structures, and the multi-purpose canister (MPC) shell are all made of 300 series stainless steel. Many canisters have neutron poison plates consisting of aluminum alloy, Al-1100 and Boral™. Table 4.7 shows the GBC-32 component properties that will be provided as a function of temperature for this initiative.

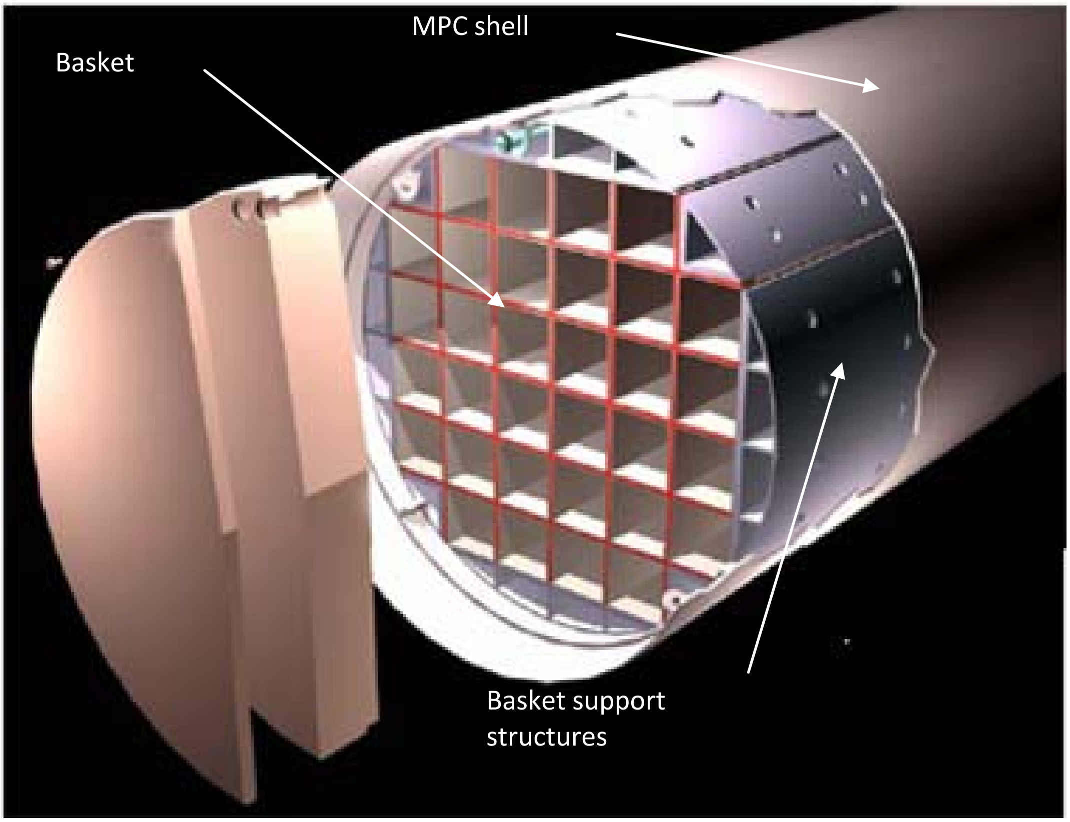


Figure 4.6. Basic Structure of the Generic Basket and Canister for Initial M&S Initiative

Table 4.7. Properties for the Generic Basket and Canister

Property	300 Series Stainless Steel	Aluminum alloy 1100	Boral™	Uncertainty
Elastic Modulus Yield Stress Ultimate Tensile Strength Total Elongation Density	Carpenter Technology, product data sheet and Allegheny Ludlum Corp., product data sheet	Properties of Aluminum Alloys, ASM and the Aluminum Association of America (1999)	Ceradyne Boral Composite Specification Sheet	Very low compared to fuel rod data. Material will be unirradiated

4.3.2.7 MATERIAL PROPERTIES FOR SURROGATE ASSEMBLY

Plans are under way to assemble a surrogate fuel assembly, replacing a majority of the Zircaloy-4 cladding tubes with copper tubing (Alloy 12200) and replacing the UO_2 fuel with lead bar stock. This surrogate assembly will be instrumented with accelerometers and tested under simulated NCT on a shaker table as previously discussed. This work is being performed under a separate WBS within UFDC (WBS 1.02.08.13.04). M&S of this experiment will be performed in an attempt to provide some validation of the modeling methodology by comparing the results to the data from the instrumented surrogate assembly. To assist in this modeling activity, properties for the materials are given. Table 4.8 shows the surrogate assembly properties that will be provided at room temperature.

Table 4.8. Properties for Materials in the Surrogate Assembly

<table>
<tr><th>Property</th><th>Copper alloy 12200</th><th>Lead</th><th>Uncertainty</th></tr>
<tr><td>Elastic Modulus</td><td rowspan="5">Mechanical Properties of Copper and Copper Alloys at Low Temperatures, #144/8 Rev. - 11/74 from Copper Development Assoc., Greenwich, CT</td><td rowspan="5">P.J. Adams, Massachusetts Institute of Technology, MS Thesis, Drapier Lab (1986) (MPDB 2013)</td><td rowspan="5">Very low compared to fuel rod data.

Material will be unirradiated</td></tr>
<tr><td>Yield Stress</td></tr>
<tr><td>Ultimate Tensile Strength</td></tr>
<tr><td>Total Elongation</td></tr>
<tr><td>Density</td></tr>
</table>

In summary, the material properties that will be documented as part of this initiative will contain the relevant material properties necessary to perform M&S of the transportation of a GBC-32 package containing 32 PWR 17×17 fuel assemblies under NCT. The largest uncertainty exists on the material properties for the irradiated components, particularly the fuel and the cladding.

Best-estimate models for the fuel and the cladding material properties will be provided. These models have been successfully used as part of the PNNL/NRC fuel performance codes. Areas of particular uncertainty have been identified and these areas will be targeted for sensitivity studies in order to determine if the uncertainties have significant impact on the response of the GBC-32 system and UNF assembly contents under NCT. If any area is found to have significant impact, it will identify a need for further research (testing, M&S, or a combination of both of these). However, it is expected to the properties provided will give reasonable value to the M&S Team for their initial modeling activities.

4.3.2.8 UNCERTAINTY QUANTIFICATION OF MATERIAL PROPERTY INPUTS

As mentioned previously, UQ will be limited to UNF mechanical properties during this initiative because of the volume of work scheduled to be conducted within the defined performance period. It will primarily involve performing sensitivity evaluations on the outputs of interest by varying the available fuel and ancillary fuel structure properties within their range of identified uncertainty. These sensitivity evaluations will employ UNF material properties and corresponding uncertainties that have had extensive UQ performed on them to support PNNL/NRC fuel performance codes, FRAPCON-3 (Geelhood et al. 2010a) and FRAPTRAN (Geelhood et al. 2010b), respectively. Uncertainties for each model are calculated based on the comparison between model predictions and measured data for the data used to develop and validate the model. In some cases, a large enough data set is not available to calculate an accurate uncertainty value and in these cases bounding uncertainty values are provided and sensitivity to this parameter can be performed to determine if these bounding uncertainties have any impact on the outputs of interest. The final material properties documented to support this initiative will summarize the nominal model and associated uncertainty for each of the material properties such that UQ analyses may readily be performed in the future if deemed necessary.

4.3.3 APPROACH FOR APPLICATION OF MODELING AND SIMULATION TECHNIQUES

4.3.3.1 MODELING OBJECTIVE

The objective of the M&S Team is to determine the dynamic loads and deformations (i.e., stresses and strains) experienced by a PWR UNF assembly placed in a representative transportation system (GBC-32), from anticipated loads during NCT. The modeling approach includes both short-term shock loads (e.g., railcar coupling) and longer-term vibration loads (e.g., random vibration during rail transportation). This effort will establish a methodology for determining PWR fuel structural performance during NCT. This methodology will be used to obtain a best estimate for the mechanical state and physical configuration of PWR fuel assemblies (including fuel pins) after a rail transportation campaign. Based on input from the Implementation Group working meeting, this methodology will use sub-modeling techniques to accurately model the complete system (railcar, cask, basket, and fuel) behavior. This will allow for more detailed finite element models of individual assemblies and faster analysis run times. The separate finite element models will also allow flexibility when updating or modifying the models to incorporate better vibration data, initial material property estimates, or other pertinent information.

4.3.3.2 TECHNICAL APPROACH

A literature review of modeling approaches for spent fuel under normal conditions of transport was initiated previously. The literature review focused on methodologies, model inputs, constitutive relations, software packages, and validation/benchmarking. The results of the literature search to-date were summarized and presented at the Implementation Group working meeting in Las Vegas, Nevada. For the M&S of the system dynamics, the finite element method (FEM) has generally replaced simpler approaches using lumped parameter models to provide more detailed response of structures. Different FEM approaches used include quasi-static modeling of inertial loads scaled by an amplification factor, time domain approaches (implicit and explicit) to capture the full transient response, modal analysis to identify natural frequencies, and spectral analysis to capture the maximum response based on the frequency content of the excitation. For analysis of cask dynamics, the majority of the literature was found to focus on accident analyses and containment. Several numerical studies with comparisons to experiments have validated the FEM method to adequately predict the response of casks using 3D models (Snow et al. 2000, Wilt et al. 2010, Ove Arup 1998, Kim et al. 2010a, Kim et al. 2010b). However, very few M&S studies were related to rail cask NCT since hypothetical accident condition (HAC) simulation loads to PWR UNF assemblies are typically assumed to be bounding. The most recent work on cask dynamics has included detailed fuel assembly geometries to evaluate the response during hypothetical accident conditions and the associated stresses and/or strains of the fuel cladding using explicit finite elements codes (Machiels 2005, NRC 2012). Despite the lack of published literature directly relevant to validation of cask assembly dynamics during NCT, the HAC papers demonstrate finite element codes accurately capture the extreme dynamic behavior associated with HAC. Therefore, the FEM approach validated for HAC simulations are entirely applicable to the study of NCT which result in far less extreme dynamic loads when compared to HAC.

An overview of the proposed technical approach is shown schematically in Figure 4.7. While this approach uses existing M&S tools and analysis methods for cask dynamic analysis, it should be emphasized that a methodology to accurately simulate PWR UNF assembly behavior during NCT has not been developed. The modeling and simulation activity captures the dynamics of the complete system (railcar, cask, basket, fuel) using several linked finite element analysis runs (sub-modeling). The cask modeling effort interfaces with the TL Team by using excitations from the railcar bed. These excitation loads are then used to define the boundary conditions for the fully loaded cask model. The system dynamics are then sequentially evaluated at the cask level, fuel assembly level, and fuel rod level. The predicted loads and deformations on the fuel rods are then output to the MP Team for failure analysis and evaluation. Further discussion for each of the modeling levels is presented below.

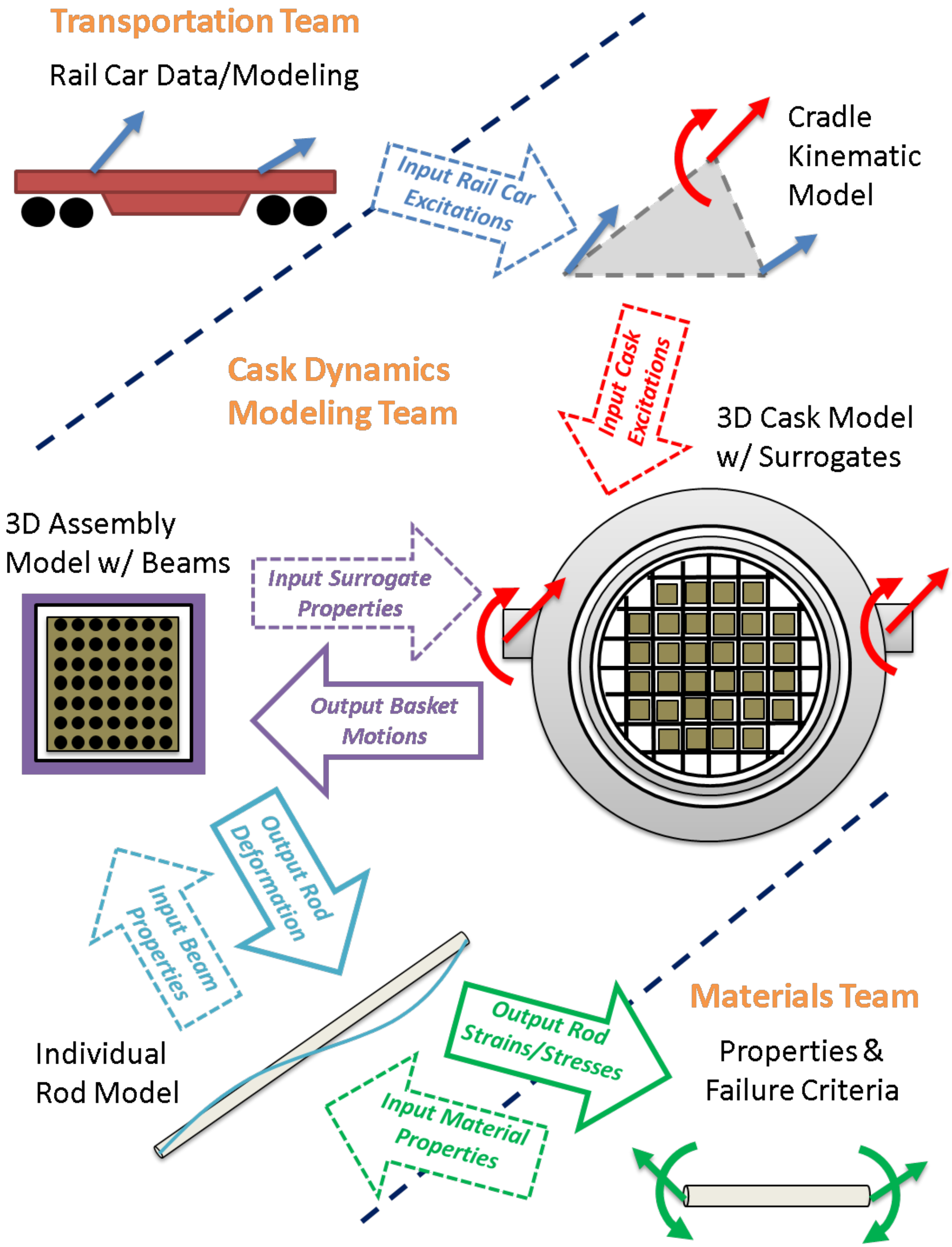


Figure 4.7. Schematic of Proposed Compartmentalized Modeling and Simulation Activities

CRADLE KINEMATIC MODEL

The excitation data from the TL Team are expected to be specified at the bed of the railcar (pending any experimental measurements at the cask restraint location that may become available during the course of the project). Since the railcar and the cradle design are presently undefined, the cradle itself will not be modeled with a detailed FEM model. Rather it will be assumed to be rigid, so a kinematics model will be developed to compute the excitations at the cask trunnions (restraint location) based on the bed excitations and actual locations of measurement or simulation. This will be completed using a spreadsheet calculation. The data will then be used as dynamic boundary conditions to feed in to the Generic Cask Model (GBC-32).

3D CASK MODEL WITH SURROGATES

The highest assembly level FEM model will consist of the generic cask, canister, 32-compartment fuel basket, neutron poison plates, and surrogate assemblies per the GBC-32 definition (3D Cask Model with Surrogates in Figure 4.7). Because of the expected low amplitude of loading and the relative stiffness of the cask to the basket (cask is much stiffer), the cask will be assumed to be rigid (or nearly rigid for numerical purposes). This model will be constructed with solid and/or shell elements. Gaps at the interfaces between the cask and canister, canister and basket, and basket and surrogate assemblies will be included in the model to capture motion and contact within each cavity as friction beneficially dissipates kinetic energy. The surrogate assembly definition-stiffness, mass, and mass moments of inertia, will be defined using a detailed PWR fuel assembly model. Based on dynamic boundary conditions applied at the trunnions or alternative restraint configuration, the response of any fuel assembly and its basket compartment can be calculated. The time history motion of the individual compartment can then be used as boundary conditions for the detailed assembly model.

ASSEMBLY MODEL WITH BEAMS

The next sub-assembly level model will consist of a basket position (fuel compartment) containing an individual WE 17×17 type PWR UNF assembly with representation of the fuel rods, spacer grids, and other structures that affect the stiffness of the assembly (e.g., thimbles, nozzle, etc.). The structures will be simulated with beam, shell, and discrete spring finite elements to represent the mass and stiffness of each component. The fuel rods are simulated as beam elements with effective stiffness properties. Motion and contact of the rods with their neighbors and the basket compartment will be captured. Time history boundary conditions developed in the Generic Cask Model of the GBC-32 will be applied at carefully selected basket position boundaries. This model will compute the time histories of deformations (curvature) and loads (moment, shear, and axial) on the effective cross-section along each fuel rod. These results will then be provided to the MP Team for failure assessments of the cladding.

INDIVIDUAL ROD MODEL

This sub-model is anticipated to consist of a continuum representation of a section of a single fuel pin. This model will include a 3D representation of the fuel pin, cladding, and any possible gap between the pin and cladding. This model will be used to obtain the effective properties of the beam elements in the detailed UNF assembly model. The model also aims to approximate the damping properties of the beam because of frictional sliding between the cladding and pellet (to be evaluated as a sensitivity).

FAILURE CRITERIA

The cladding stresses and strains predicted by the detailed UNF assembly model under rail-related NCT conditions will then be used for limited performance assessments of cladding integrity. Initially, the failure criteria to be used will rely primarily on existing data from the material properties database established by the MP Team. For shock events, the maximum predicted cladding strain will be compared to the maximum allowable elongation (true strain limit). For vibration, the cyclic stress magnitude will be used with the cladding fatigue limits to estimate the number of cycles to failure. If this comparison shows fuel assembly performance during NCT far from the established failure criteria, then this may provide adequate demonstration that the cladding will not experience gross failure. If the FEM produced stress/strains indicate conditions that are close to the established failure criteria, then further experimental work may be required to produce a more prototypic failure curve, and further failure analyses should be performed on an actual fatigue life analysis. More detailed analyses for different hypothesized failure mechanisms in irradiated cladding may also be needed and will be proposed as appropriate.

4.3.3.3 MODEL INPUTS AND OUTPUTS

The baseline case is desired to be characteristic of prototypic rail-related NCT exclusively.

GEOMETRY

The geometry will be based on the generic cask/canister/basket definition of the GBC-32 capable of holding 32 WE 17×17 type PWR UNF assemblies per the Steering Team's recommendations and Implementation Group's concurrence.

MATERIALS

The material properties used in the model will be obtained from the material property database established and documented by the MP Team. Data for density, elastic modulus, Poisson's ratio (or shear modulus), yield strength, ultimate tensile strength, and elongation will be used to create isotropic bilinear elastic-plastic material models for the applicable components. Proposed material assignments from the MP Team include UO_2 for the fuel pellet, Zircaloy-4 for the cladding, Inconel-718 and Zircaloy-4 for spacer grids, Boral and Al-1100 for neutron poison plates, and 300 series stainless steel for the canister and basket components. Material properties for the irradiated fuel and cladding will be used from this materials database. The spacer grid spring stiffness will be reduced to account for relaxation and appropriate sensitivity studies will be performed for this parameter to determine influence.

LOADING

The primary loading input for the rail-related NCT evaluations is time-history motion provided by the TL Team at the railcar level as previously defined. Since the cask and railcar are treated as rigid, this will resolve to just three translational and three rotational components for the cask. The desired format of data from the TL Team are actual acceleration-time histories. The acceleration time history will be applied as boundary conditions in the time domain model. For shock loading, these will be data for given short-term events representative of the largest g-loads imparted to the railcar (e.g., railcar coupling or humping). For shock (short-duration event), the explicit dynamics model will simulate the entire event. For vibration loading (long-duration event), these will be long-term data representative of the random vibrations during travel of the railcar. For this portion of the M&S analysis, a time domain implicit or explicit solver will be used to determine fuel assembly performance under characteristic vibration conditions. Prior to NCT evaluations, modal analyses of the fuel rod and assembly will be performed to identify the system frequencies of interest. Spacer grid spring pressure sensitivity will be evaluated using this approach. Gravitational acceleration will also be included in the model.

The cask and its contents have an elevated temperature profile because of fuel decay heat and coupled rejection capability. The thermal profiles will be based on prototypic decay heating established from reactor burn, pool cooling, and dry storage durations with the addition of solar insolation and anticipated ambient conditions. Material properties typically depend on temperature, so these properties will be included in the M&S platform. A single storage duration and associated temperature distribution will be assumed based on existing thermal modeling performed within the UFDC program. Property variations based on the temperature range will be evaluated and included in the material model definitions. Material properties will be assigned to each component in the model using a reasonable number of zones to approximate the material property variations for different temperature domains.

RESULTS

The M&S Team will input acceleration time history data into the finite element models to obtain stress and strain results of the fuel cladding during rail NCT. The stress and strain results will then be reviewed by the MP Team to determine if the fuel cladding is near the established failure criteria. This effort establishes a methodology for determining WE 17×17 type PWR UNF structural performance during NCT.

KEY ASSUMPTIONS

The major assumptions made for the M&S development effort are detailed below. These assumptions are required to make the developed model tractable.

- The cask can be assumed rigid for the expected load levels of rail-related NCT.
- The cradle is assumed to be rigid.
- The basket is assumed to have nominal geometry with no initial deformations.
- The fuel rods are assumed to have nominal geometry with no initial bowing.
- Fuel pellet geometry is assumed to be nominal.
- The spacer grid springs are assumed to be relaxed.
- Material properties are assumed to be representative of moderate to high burnup fuel.

Additionally, thermal steady state is assumed during structural loading. A representative NCT temperature field from an existing transport cask model similar to the GBC-32, with representative decay heat load will be used to establish the temperature-dependent structural material properties.

4.3.3.4 CONTINGENCIES

While this approach uses existing M&S tools and analysis methods for cask dynamic analysis, no methodology to simulate PWR UNF assembly behavior during NCT has been established. With that understanding, the M&S Team identified possible difficulties in the proposed approach that could arise during the model development period. This could require altering or simplifying the proposed approach in order to ensure project success. These contingencies are:

- The planned sub-modeling approach adds complexity during exchange of information between models. The high-level models must transfer time-history boundary condition data to the lower level models (“handshaking”) and entails a large number of data points simulated at a significant number of boundary locations for each event. Successful handshaking may require accommodations of different time step requirements for the models. Slightly different responses between the surrogate and detailed assemblies could introduce local artificial loading that is not representative. If such difficulties are observed, then careful expansion of the lower-level modeling boundary or combination of the proposed modeling domains may be pursued to eliminate any such artifacts.
- The selected modeling approach using transient dynamics calculations was necessary to capture the nonlinear effects of contact between components, friction at interfaces, and potential inelastic deformation of the materials. Depending on the magnitude and frequency content of the time histories provided by the TL Team for the actual rail conveyance (the magnitudes of which are expected to be lower than existing guidance), the model could behave nearly elastically with localized areas of nonlinearities. If so, then an implicit evaluation using time domain or spectral approaches may be more appropriate and numerically efficient for the vibration analysis.

4.3.3.5 POSSIBLE RESOURCES

Pacific Northwest National Laboratory has developed models for application review/independent confirmation that should be considered for use as a starting point:

- Structural
  - Casks (Holtec HI-STAR 100, Transnuclear TN-68 & TN-40)
  - UNF (WE 17×17, GE 9×9)
- Thermal
  - Casks (majority of transportation, transfer, and storage cask industry leaders modeled)
  - UNF (numerous PWR & BWR assemblies modeled)

Models that should also be considered as a starting point are those constructed by Sandia National Laboratories (NRC 2012). The Implementation Team will evaluate the applicability of models available and recommend the initial model construction basis. Final approval will be at the discretion of the Initiative Lead.

4.4 COMPUTER CODE(S), METHODS AND APPLICATION DEVELOPMENT

A number of analytical codes and computational methods are generally used only in the national laboratory environment, and specific modeling approaches strongly depend on the subject matter under evaluation as well as the subject's behavior. As previously identified, all code candidates selected for application to support this initiative must be commercially available (e.g., ANSYS, Abaqus, LS-DYNA, etc.). The Implementation Team will evaluate the applicability of codes and methods available and make a determination regarding integration of codes, as required. Final selections will be approved by the Initiative Lead prior to application.

4.5 MODELING AND SIMUALTION VALIDATION

After fuel is irradiated to high burnup levels, the fuel and clad may bond together to form an integrated system. It is recognized that understanding the separate effects of pellet/clad interface dynamics in a transportation environment is critical when evaluating integrated assembly properties. Oak Ridge National Laboratory (ORNL) has developed testing equipment to evaluate the rod structure's integrity, and collect data on the fuel/clad interface and fuel/clad interactions (described in Section 4.6). Prior to conducting the physical experimentation on high burnup fuel, ORNL will conduct an FEM evaluation of UNF vibration integrity. This UNF fuel-clad interaction modeling will use a "contact element" approach as well as the embedded/prescribed boundary conditions, such as internal pressure and residual stress, etc. Additionally, FEM analyses for investigating the synergistic effects of UNF rod–to-rod coupling/interactions through the spacer grids and anchor springs during vibration events will be evaluated. The output from these studies should provide missing information on the effects of pellet-clad interaction, as well as the impact of pellet fracture and pellet-clad bonding effects to the cladding fatigue strength.

Large-scale M&S, and associated methodology, validation will most likely come from studies and information yielded by the literature reviews being conducted for this initiative. This belief is based on the absence of physical modeling and coupled testing of UNF in the DOE and NRC sectors. It is highly desirable to validate model predictions against experimental data to ensure that the modeling approach, assumptions, geometry simplifications, and material properties are appropriate. One of the more promising approaches for validation is to use the shaker table testing being performed at SNL under UFDC. The Shaker Table Simulations/Evaluations work being performed under UFDC S&T (WBS# 1.02.08.13.03 and .04) could potentially serve to establish a viable benchmarking and validation path. The advantage of this experiment is that it will permit the M&S Team to have complete information regarding the experimental geometry and loading conditions to confidently construct a representative model. Comparison of predicted results for rod accelerations or grid forces at specified locations in the fuel assembly M&S representation against measured experimental data can then provide validation of the modeling methodology. The validation could be completed by using the detailed assembly model to simulate the shaker table test performed (this includes appropriate boundary conditions and material properties) at SNL and comparing the results with actual measured data.

Three other potential validation sources could be derived from papers that contain dynamic characteristics similar to NCT as identified preliminarily during the ongoing literature review (Queval et al. 2001, Shirai et al. 2007, Prulhiere and Israel 1980). These papers could provide additional validation information if both measured dynamic data and quality description of the test conditions and components are available for accurate model construction.

The Implementation Team will be responsible for developing and recommending a credible validation path. Final selection will be approved by the Initiative Lead prior to application.

4.6 EXPERIMENTAL MEASUREMENT PLANNING AND POSSIBLE EXECUTION

A lack of data and supporting information, or “gaps,” may be revealed during refined M&S methodology development and application or within the preliminary calculation phases under this initiative. The Implementation Team will evaluate the significance of any such identified gap and form recommendations regarding resolution paths. If a particular gap can be resolved via experimental measurement and will fit within the budget constraints and performance period of this initiative, real-time planning and possible execution will be proposed. An example of this M&S validation gap analyses will be the performance of control simulation experiments to generate calibration data to describe material properties at the fuel rod structural level (flexural rigidity) or at interfaces (cohesion bonding) for FEM material input. This information will then be fed back in to the finalized models and reflected in the analyses results prior to conclusion of the initiative. Otherwise, planning and execution will be recommended for possible future consideration, and will be communicated within UFDC.

A possible example of this might apply to performing additional bend testing of a WE 17×17 rod using the reversal bending fatigue tester (RBT) at ORNL to generate the fatigue strength data for the materials of interest. This may provide valuable information regarding the influence of pellet-clad interactions at frequency and load levels of interest as well as fuel rod flexural rigidity evolution under cyclic loading; the current testing at ORNL will be conducted on an H.B. Robinson WE 15×15 high burnup UNF rod. Testing for this program will be focused on resolving the more complex boundary conditions potentially occurring in a UNF assembly under rail-related NCT. The proposed experiments would initially occur in an out-of-hot cell environment to provide proof-of-principle information, identify critical failure modes, such as bonding effects, surface effects, and coupling effects of the various components in the fuel assembly, and allow multiple tests to be run to evaluate and identify key testing parameters of interest. Using existing UNF at ORNL, the testing may then migrate into the hot cell to support data collection on irradiated fuel of interest. Goals of the testing include gathering information on the fuel-to-fuel coupling effect through the spacer grid/anchor spring, the effect of surface stress concentrations on UNF reliability, and the associated effective cladding lifetime, and investigating bending fatigue performance. The information gathered in these tests may also complement the shaker table testing work being conducted at Sandia National Laboratories.

The Implementation Team will be responsible for developing the basis and recommending such testing for future consideration and campaign management approval.

4.7 METHODOLOGY DEMONSTRATION

The methodology demonstration portion of this initiative will consist of exercising the models developed as identified in the M&S methodology development phase, which incorporates all retrieved and integrated bases and supporting information, to evaluate loading and associated mechanical responses of UNF rods and key structural components. These models will be used to evaluate performance characteristics of UNF cladding and ancillary components under loading stemming from rail-related NCT. In doing so, a preliminary assessment of the regulatory basis will have been performed. This assessment will provide insight into the post-transport integrity of the UNF rods and effects on subsequent retrievability.

An encompassing Project Summary Report will document the overall results of the project, including methodology, analytical and experimental results, findings and associated interpretations (activity M2FT-13OR0822015-Demonstration of Approach and Results on Used Fuel Performance Characterization identified in the Gantt chart in Appendix D.). Recommendations for further work will then be developed and documented by the Implementation Team members. Additionally, all inputs and analyses will be completely documented in a traceable and defendable manner, and analysis models will be made available to DOE's UFDC community for future assessment purposes.

5. REFERENCES

10 CFR 71. Packaging and Transportation of Radioactive Material. U.S. Nuclear Regulatory Commission, Washington, D.C.

10 CFR 72. Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste. U.S. Nuclear Regulatory Commission, Washington, D.C.

AAR (Association of American Railroads). 2003. *Performance Specification for Trains Used to Carry High-Level Radioactive Material*. Standard S-2043, Association of American Railroads, Washington, D.C. (DIRS 166338) 2003.

Adkins HA. 2013. Used Nuclear Fuel Loading and Structural Performance Under Normal Conditions of Transport – Method and Approach. FCRD-TIO-2011-00050. U. S. Department of Energy, Washington D.C.

ANSI N14.23. 1980. *Draft American National Standard Design Basis for Resistance to Shock and Vibration of Radioactive Material Packages Greater than one Ton in Truck Transport*. American National Standards Institute, Inc.

BRC (Blue Ribbon Commission). 2012. *Blue Ribbon Commission on America's Nuclear Future, Report to the Secretary of Energy.* Prepared by the Blue Ribbon Commission on America's Nuclear Future for the U.S. Department of Energy, Washington, D.C.

Bryan ME. 1965. *Transportability Study Covering Highway Movement of Atomic Energy Commission 15-Ton Nuclear Cask from Wilmington, Delaware to Albuquerque, New Mexico*. USATEA Report 65-9. U.S. Army Transportation Engineering Agency, Fort Eustis, Virginia.

Carter JT, AJ Luptak, J Gastelum, C Stockman, and A Miller. 2012. *Fuel Cycle Potential Waste Inventory for Disposition*. FCR&D-USED-2010-000031 Rev. 5, U.S. Department of Energy, Washington, D.C.

Cory AR. 1991. "Flask Tiedown Design and Experience of Monitoring Devices." *RAMTRANS* 2(1/3):15-22.

Dalziel BP, MA Elbestawi, and JW Forest. 1986. "CANDU Irradiated Fuel Transportation: The Shock and Vibration Program." In *International Studies on Certain Aspects of the Safe Transport of Radioactive Materials, 1980-1985*. IAEA-TECDOC-375. pp. 35-45. International Atomic Energy Agency, Vienna, Austria.

Dideon CG and GM Bain. 1983. *Fuel Performance Under Extended-Burnup Operation B&W 15×15 Design.* DOE/ET/34212-38, BAW-1716, Babcock & Wilcox, Lynchburg, Virginia.

DOE (U.S. Department of Energy). 2009. *National Transportation Plan.* DOE/RW-0603 Revision 0, U.S. Department of Energy, Washington, D.C.

DOE (U.S. Department of Energy). 2013. *Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste*. U.S. Department of Energy, Washington, D.C. Available at http://energy.gov/downloads/strategy-management-and-disposal-used-nuclear-fuel-and-high-level-radioactive-waste.

Energy Information Administration. 1995. *Spent Nuclear Fuel Discharges from U.S. Reactors 1993*. SR/CNEAF/95-01, Energy Information Administration, Washington, D.C.

ENUSA Industrias Avanzadas, S.A. 2013. "Latest Generation Integral Solutions." Accessed March 30, 2013 at http://www.enusa.es/eng/actividad/pwr.html.

EPRI. 2010. *Impacts Associated with Transfer of Spent Nuclear Fuel from Spent Fuel Storage Pools to Dry Storage After Five Years of Cooling.* TR-1021049, Electric Power Research Institute, Palo Alto, California.

Fields SR. 1983. *Dynamic Analysis to Establish Normal Shock and Vibration of Radioactive Material Shipping Packages, Quarterly Progress Report, Final Summary Report.* NUREG/CR-2146, Volume 3, HEDL-TME 83-18. Hanford Engineering Development Laboratory, Richland, Washington.

Foley JT. 1966a. "Preliminary Analysis of Data Obtained in the Joint Army/AEC/Sandia Test of Truck Transport Environment." *The Shock and Vibration Bulletin*, Bulletin 35, Part 5, pp. 57-70, U.S. Naval Research Laboratory, Washington, D.C.

Foley JT. 1966b. *The Environment Experienced By Cargo on a Flatbed Tractor-Trailer Combination.* Report No. SC-RR-66-677, Sandia National Laboratory, Albuquerque, New Mexico.

Foley JT and MB Gens. 1971a. "Shock and Vibration Measurements During Normal Rail and Truck Transport." In *Proceedings of the Third International Symposium Packaging and Transportation of Radioactive Materials Richland, Washington, August 16-20, 1971*. pp. 905-933, U.S. Atomic Energy Commission, Oak Ridge Tennessee.

Foley JT and MB Gens. 1971b. *Environment Experienced By Cargo During Normal Rail and Truck Transport – Complete Data.* Report No. SC-M-71-0241. Sandia National Laboratories, Albuquerque, New Mexico.

Geelhood, KJ, WG Luscher, and CE Beyer. 2010a. *FRAPCON-3.4: A Computer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup*, NUREG/CR-7022, Vol. 1, PNNL-19418, Vol. 1, Pacific Northwest National Laboratory, Richland, Washington.

Geelhood, KJ, WG Luscher, CE Beyer, and JM Cuta. 2010b. *FRAPTRAN 1.4: A Computer Code for the Transient Analysis of Oxide Fuel Rods*, NUREG/CR-7023, Vol. 1, PNNL-19400, Vol. 1, Pacific Northwest National Laboratory, Richland, Washington.

Gens MB. 1970. "The Rail Transport Environment." *The Journal of Environmental Sciences*, XIII(4):14-20.

Glass RE and KW Gwinn. 1986. "Shock and vibration environments for truck-transported nuclear waste: Test and analysis." INMM 1986 Annual Meeting, June 22-25, 1986. pp. 319-323.

Glass RE and KW Gwinn. 1987. *TRUPACT-I Over-the-Road Test*. SAND87-0513, Sandia National Laboratories, Albuquerque, New Mexico.

Gwinn KW, RE Glass, and KR Edwards. 1991. *Over-the-Road Tests of Nuclear Materials Package Response to Normal Environments.* SAND91-0079, Sandia National Laboratories, Albuquerque, New Mexico.

Hanson B, H Alsaed, C Stockman, D Enos, R Meyer, and K Sorenson. 2012. *Gap Analysis to Support Extended Storage of Used Nuclear Fuel*, FCRD-USED-2011-000136, Rev. 0, U.S. Department of Energy, Washington, D.C.

Ito K, K Kamimura, and Y Tsukuda. 2004. "Evaluation of Irradiation Effect on Spent Fuel Cladding Creep Properties." In *Proceedings of the 2004 International Meeting on LWR Fuel Performance Orlando, Florida, September 19-22, 2004*, Paper 1117.

Kim K, J Kim, K Choi, T Shin, and H Yun. 2010a. "Dynamic Impact Characteristics of Kn-18 SNF Transport Cask – Part 1: An Advanced Numerical Simulation and Validation Technique." *Annals of Nuclear Energy* 37(4):546-559.

Kim K, J Kim, K Choi, T Shin, and H Yun. 2010b. "Dynamic Impact Characteristics of Kn-18 SNF Transport Cask – Part 2: Sensitivity Analysis of Modeling and Design Parameters." *Annals of Nuclear Energy* 37(4):560-571.

Laux D, D Baron, G Despaux, AI Kellerbauer, and M Kinoshita. 2012. "Determination of high burn-up nuclear fuel elastic properties with acoustic microscopy." *Journal of Nuclear Materials*, 420(1–3):94–100.

Leduc DR. 2012. *Dry Storage of Used Fuel Transition to Transport*, FCRD-UFD-2012-000253, Savannah River National Laboratory, Aiken, South Carolina.

Livermore Software Technology. 2007. *LS-DYNA Keyword User's Manual Volume I*, Version 971. Livermore Software Technology Corporation.

Machiels A. 2005. *Spent Fuel Transportation Applications: Global Forces Acting on Spent Fuel Rods and Deformation Patterns Resulting from Transportation Accidents.* EPRI-1011817. Electric Power Research Institute, Palo Alto, California.

Machiels AJ and AH Wells. 2009. "Direct Disposal of Dual-Purpose Canisters—Options for Assuring Criticality Control." Presentation at the Winter 2009 NWTRB Meeting Accessed March 30, 2013 at http://www.nwtrb.gov/meetings/2009/jan/machiels.pdf.

Magnuson CF. 1977. *Shock and Vibration Environments for Large Shipping Container During Truck Transport (Part I).* SAND77-1110, Sandia National Laboratories, Albuquerque, New Mexico.

Magnuson CF. 1978. *Shock and Vibration Environments for a Large Shipping Container During Truck Transport (Part II).* NUREG/CR-0128/SAND78-033, Sandia Laboratories, Albuquerque, New Mexico.

Magnuson CF. 1980. *Shock Environments for Large Shipping Containers During Rail Coupling Operations.* NUREG/CR-1277, SAND79-2168, Sandia National Laboratories, Albuquerque, New Mexico.

Magnuson CF. 1982. *Shock and Vibration Environments Encountered During Normal Rail Transportation of Heavy Cargo*. SAND82-0819, Sandia National Laboratories, Albuquerque, New Mexico.

Magnuson CF and LT Wilson. 1977. *Shock and Vibration Environments for Large Shipping Containers on Rail Cars and Trucks.* SAND76-0427, NUREG-766510, Sandia National Laboratories, Albuquerque, New Mexico.

McConnell P. 2012. *Fuel-Assembly Shaker Test Plan, Tests for Determining Loads on Used Nuclear Fuel under Normal Conditions of Transport.* FCRD-UFD-2012-000341, Sandia National Laboratories, Albuquerque, New Mexico.

Michener TE, DR Rector, JM Cuta, RE Dodge, and CW Enderlin. 1995. *COBRA-SFS: A Thermal-Hydraulic Code for Spent Fuel Storage and Transportation Casks*. PNL-10782, Pacific Northwest National Laboratory, Richland, Washington.

Miller A, R Rechard, E Hardin, and R Howard. 2012. *Assumptions for Evaluating Feasibility of Direct Geologic Disposal of Existing Dual-Purpose Canisters*. FCRD-UFD-2012-000352,U.S. Department of Energy, Washington, D.C.

Mok GC, LE Fischer, and ST Hsu. 1993. *Stress Analysis of Closure Bolts for Shipping Casks*. NUREG/CR-6007, UCRL-ID-110637, Lawrence Livermore National Laboratory, Livermore, California.

MPDB. 2013. "Temperature Dependent Elastic & Thermal Properties Database." Accessed on March 30, 2013 at http://www.jahm.com/.

NRC (U.S. Nuclear Regulatory Commission). 1999. *Standard Review Plan for Transportation Packages for Radioactive Material*. NUREG-1609, U.S. Nuclear Regulatory Commission, Washington, DC.

NRC (U.S. Nuclear Regulatory Commission). 2003. *Cladding Considerations for the Transportation and Storage of Spent Fuel*, Interim Staff Guidance Memorandum 11, Revision 3, November 17, 2003, U.S. Nuclear Regulatory Commission, Washington, D.C.

NRC (U.S. Nuclear Regulatory Commission). 2012. *Spent Fuel Transportation Risk Assessment*. NUREG-2125, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Fuel Industries. 2013. "Basic Structure of PWR Fuel." Accessed March 30, 2013 at http://www.nfi.co.jp/e/product/prod02.html#b.

O'Donnell WJ, and BF Langer. 1964. "Fatigue Design Basis for Zircaloy Components," *Nuclear Science and Engineering* (20):1-12.

OECD (Organisation for Economic Co-operation and Development). 2012 *Nuclear Fuel Safety Criteria Technical Review*. OECD NEA No. 7072. Organisation for Economic Co-operation and Development, Nuclear Energy Agency. Available at http://www.oecd-nea.org/nsd/reports/2012/nea7072-fuel-safety-criteria.pdf.

ORNL (Oak Ridge National Laboratory). 2011. *SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design, Version 6.1*. ORNL/TM-2005/39, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Ove Arup (Ove Arup & Partners International Ltd and Gesellschaft für Nucklear-Behälter, mbH). 1998. *Evaluation of Codes for Analysing the Drop Test Performance of Radioactive Material Transport Containers, Phase 1 Final Report*. 52376/02, Ove Arup & Partners International Ltd, London, United Kingdom and Gesellschaft für Nucklear-Behälter, mbH, Essen, Germany

Petry SF. 1980. *Rail Tiedown Tests with Heavy Casks for Radioactive Shipments*. DP-1536. Savannah River Laboratory, Aiken, South Carolina.

Prulhiere JP and F Israel. 1980. "Measurement of Vibrational Effects on an Assembly of PWR Fuel Elements During Handling and Transport By Road and Rail." In *Proceedings of 6th International Symposium on the Packaging and Transportation of Radioactive Materials, November 10-14, 1980, Berlin, West Germany*, pp. 1292-1301.

Pujet D and P Malesys. 1989. "Measurement of the Acceleration Undergone by the Trunnions of Irradiated Fuel Transport Flasks During Normal Use." In *Proceedings of 9th International Symposium on the Packaging and Transportation of Radioactive Materials, June 11-16, 1989, Washington, D.C*. pp. 932-939.

Queval, JC, J Rigaudeau, and E Viallet. 2001. "Seismic Tests of Interacting Full-Scale Fuel Assemblies on Shaking Table," SMiRT 16 Conference, August 12-17, 2001, Washington D.C.

Sanders TL, KD Seager, YR Rashid, PR Barrett, AP Malinauskas, RE Einziger, H Jordan, TA Duffey, SH Sutherland, and PC Reardon. 1992. *A Method for Determining the Spent-Fuel Contribution to Transport Cask Containment Requirements*. SAND90-2406, Sandia National Laboratories, Albuquerque, New Mexico.

Shin MK, HA Lee, JJ Lee, KN Song, and GJ Park. 2008. "Optimization of a nuclear fuel spacer grid spring using homology constraints", *Nuclear Engineering and Design*, 238(10).

Shirai, K, T Saegusa, and M Wataru. 2007. "Experimental Studies of Free-Standing Spent Fuel Storage Cask Subjected to Strong Earthquake," In *Proceedings of the 15th International Symposium on the Packaging and Transportation of Radioactive Materials PATRAM 2007, October 21-26, 2007, Miami, Florida, USA.*

Siefken LJ, EW Coryell, EA Harvego, and JK Hohorst. 2001. *SCDAP/RELAP5/MOD 3.3 Code Manual MATPRO-A Library of Materials Properties for Light-Water-Reactor-Accident Analysis.* NUREG/CR-6150, Vol. 4, Rev. 2, INEL-96/0422, Idaho National Engineering and Environmental Laboratory, Idaho Falls, Idaho.

Smith GP. 1983. *The Evaluation and Demonstration of Methods for Improved Fuel Utilization*. DOE/ET/34010-10, CEND-414, C-E- Power Systems, Windsor, Connecticut.

Smith GP, EJ Ruzauskas, RC Pirek, and M Griffiths. 1993. *Hot Cell Examination of Extended Burnup Fuel from Calvert Cliffs-1*. TR-1033-2, Electric Power Research Institute, Palo Alto, California.

SNL (Sandia National Laboratories). 2011. *Presto 4.20 User's Guide*, SAND2011-1824, Sandia National Laboratories, Albuquerque, New Mexico.

Snow SD, DK Morton, TE Rahl, AG Ware, and NL Smith. 2000. "Analytical Evaluation of Drop Tests Performed on Nine 18-Inch Diameter Standardized DOE Spent Nuclear Fuel Canisters." INEEL/CON-99-01072, Idaho National Engineering & Environmental Laboratory, Idaho Falls, Idaho..

TTCI (Transportation Technology Center, Inc.). 2013. *NUCARS® Help Manual, Version 2013.1*. Transportation Technology Center, Inc., Pueblo, Colorado.

Wagner JC. 2001. *Computational Benchmark for Estimation of Reactivity Margin from Fission Products and Minor Actinides in PWR Burnup Credit*. NUREG/CR-6747, ORNL/TM-2000/306, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Weihermiller WB and GS Allison. 1979. *LWR Nuclear Fuel Bundle Data for Use in Fuel Bundle Handling*. PNL-2575, UC-85, Battelle Pacific Northwest Laboratory, Richland, Washington.

Wilt T; A Chowdhury, and T Ahn. 2010. *A Literature Review of the Structural Integrity of Analog Canisters*. Center for Nuclear Waste Regulatory Analyses, San Antonio, Texas. 20.14002.01.441, Q201010220001, ML112650310

Appendix A

Steering Working Group Meeting Participants List

Appendix A

Steering Working Group Meeting Participants List

Used Nuclear Fuel Loading and Structural Performance under Normal Conditions of Transport & Storage

Attendance List

Name	Organization	Email	Phone
Rob Howard	ORNL	howardrl1@ornl.gov	(865) 241-5750
John Orchard	DOE-NE	John.orchard@doe.gov	(702) 295-2395
Gordon Bjorkman	NRC	gordon.bjorkman@nrc.gov	(301) 492-3298
Peter Swift	SNL	pnswift@sandia.gov	(505) 284-4817
Harold Adkins	PNNL	Harold.adkins@pnnl.gov	(509) 372-6692
Jorge Monroe-Rammsy	DOE	jorge.monroe-rammsy@doe.gov	(702) 295-2369
Siegfried Stockinger	DOE-NE	Siegfried.stockinger@doe.gov	(702) 295-2389
Mark Nutt	ANL	wnutt@anl.gov	(630) 252-8387
Bob Clark	DOE-NE	Robert.Clark@doe.gov	(702) 295-2376
Ned Larson	DOE-NE	ned.larson@doe.gov	(702) 295-2378
Steve Marschman	INL	Steve.marschman@inl.gov	(208) 526-2335
Ram Murthy	DOE/NE-53	ram.murthy@doe.gov	(702) 295-2361
John Wagner	ORNL	wagnerjc@ornl.gov	(865) 241-3570

Appendix B

Implementation Plan Working Group Meeting Participants List

Appendix B

Implementation Plan Working Group Meeting Participants List

Used Nuclear Fuel Loading and Structural Performance under Normal Conditions of Transport & Storage Attendance List			
Name	Organization	Email	Phone
Rob Howard	ORNL	howardrl1@ornl.gov	(865) 241-5750
John Orchard	DOE-NE	John.orchard@doe.gov	(702) 295-2395
Gordon Bjorkman	NRC	gordon.bjorkman@nrc.gov	(301) 492-3298
Harold Adkins	PNNL	harold.adkins@pnnl.gov	(509) 372-6692
Ned Larson	DOE-NE	ned.larson@doe.gov	(702) 295-2378
Steve Marschman	INL	steve.marschman@inl.gov	(208) 526-2335
John Wagner	ORNL	wagnerjc@ornl.gov	(865) 241-3570
Paul McConnell	SNL	pemccon@sandia.gov	(505) 845-8361
Steve Maheras	PNNL	steven.maheras@pnnl.gov	(614) 486-5350
Remi Dingreville	SNL	rdingre@sandia.gov	(505) 844-9083
Brady Hanson	PNNL	brady.hanson@pnnl.gov	(509) 375-5051
Carl Beyer	PNNL	carl.beyer@pnnl.gov	(509) 372-4605
Brian Koeppel	PNNL	brian.koeppel@pnnl.gov	(509) 372-6816
Jy-An Wang	ORNL	wangja@ornl.gov	(865) 574-2274
Ruben Pena	TTCI	ruben_pena@aar.com	(719) 584-0543
Nicholas Wilson	TTCI	nicholas_wilson@aar.com	(719) 584-0573
Dennis Vinson	SRNL	dennis.vinson@srnl.doe.gov	(803) 725-1961
Ken Sorenson	SNL	kbsoren@sandia.gov	(505) 844-0074
Doug Ammerman	SNL	djammer@sandia.gov	(505) 845-8158
Kenneth Geelhood	PNNL	kenneth.geelhood@pnnl.gov	(509) 372-4556
Bruce Bevard	ORNL	bevardbb@ornl.gov	(865) 574-0279
Justin Coleman	INL	justin.coleman@inl.gov	(208) 526-4741
Yung Liu	ANL	yyliu@anl.gov	(630) 252-5127
John Scaglione	ORNL	scaglionejm@ornl.gov	(865) 574-9284

Appendix C

Laboratory-Specific Statements of Work

Appendix C

Laboratory-Specific Statements of Work

C-1. Pacific Northwest National Laboratory

C-1.1 Statement of work to provide technical services supporting the Department of Energy's Office of Nuclear Energy Used Fuel Disposition Campaign

Provide as directed by ORNL, technical services to establish the methodology and approach for coupling M&S and experimental efforts to quantify the mechanical performance of cladding and relevant fuel assembly components, with quantified uncertainties, during normal conditions of storage and transportation. To the maximum extent possible, scoping M&S activities determined to be of high priority will be initiated. This works cope will support development and integration of used nuclear fuel (UNF) data and analysis capabilities, as well as support the Used Fuel Disposition Campaign (UFDC) mission regarding scientific research and technology development to strength the basis for and understanding of storage and transportation of used nuclear fuel. The technical services support work includes, but is not limited to collecting information via literature search, establishing a comprehensive foundation to be used in multi-physics evaluations, performing analyses, and preparing a variety of supporting documentation.

C-1.2 The following tasks are included in this work scope:

1. Perform the subtasks that this task is comprised of (listed below) as well as provide project management, task leadership, and technical contributions within the scopes defined in all following tasks (2-6).
    a. Conduct a SME steering team meeting for RD&D plan development that identifies all inputs and leads into the development of a top level RD&D implementation plan. Produce a Finalized Method and Approach document with input received from the SME core/steering team meeting. Submit document for program review.
    b. Prepare for and conduct a SME core implementation team meeting leading to the development of an implementation path for RD&D plan developed under previous subtask, initiate execution, establish responsibilities, and communicate deliverable dates.
    c. Develop and submit draft and final documentation identifying the Execution RD&D Plan and Demonstration of Approach on Selected Technical Issue. This milestone will be the blueprint for how to move forward for the next several years to resolve the numerous technical challenges related to storage

and transport of UNF that will require M&S, measured data, and UQ. In short, developing a used fuel performance modeling, simulation and experimental integration plan.

Transportation Loading and Supporting Information:

2. Perform a literature review for the purpose of identifying all loading and governing regulations applicable to the determination of establishing preliminary performance characteristics of used fuel during normal conditions of storage and transport in the US. Document encompassing data and information as well as inputs yielded from this literature search. Also document specific recommendations based on this information to be implemented in future multi-physics M&S evaluation work performed under this as well as possible future programs.

Material Property Development and Documentation:

3. Perform a literature review identifying material properties relevant to moderate to high burnup used nuclear fuel as well as associated storage and transport components to establish a credible database to feed into state-of-the-art M&S tools for determining cladding and fuel assembly specific mechanical performance characteristics. Document encompassing data and information as well as inputs yielded from this literature search. Also document specific recommendations based on this information to be implemented in future multi-physics M&S evaluation work performed under this as well as possible future programs.

M&S Methodology Development & Application:

4. Perform a literature review identifying modeling inputs and previously applied M&S methodologies to establish a credible basis and state-of-the-art analytical approach for determining used fuel mechanical performance based on loading of used fuel cladding and relevant assembly components. The purpose of this work is to preliminarily establish performance characteristics of used fuel during normal conditions of storage and transport. Document encompassing data, information, and previously implemented modeling methodologies as well as inputs yielded from this literature search. Also document specific recommendations based on this information to be implemented in future multi-physics M&S evaluation work performed under this as well as possible future programs. This work is to be performed collaboratively with the SNL M&S Technical Contributor(s).
5. Perform multi-physics M&S construction and preliminary evaluations based on information yielded from Tasks 2, 3, 4, and 6. Document results and findings initially in the form of a letter report. Assemble and submit formal documentation, including all modeling assumptions, incorporated input information, and associated results, to be incorporated in to an encompassing demonstration report. This work is to be performed collaboratively with the SNL M&S Technical Contributor(s).

6. Document encompassing data, information, material properties, and modeling inputs yielded from literature searches outlined previously. This information will form the foundation of the multi-physics evaluations. All of the above literature review results and associated recommendations will be assembled to form an all-encompassing document that graded UQ can be performed on and fed into the RD&D.

C-1.3 Milestones:

1. Deliverables for subtasks making up Task 1 are as follows:
   a. Conduct the SME steering team meeting on or prior to 12/21/2012. Submit report entitled "Proposed RD&D Plan for Used Fuel Performance Modeling, Simulation and Experimental Integration" documenting proposed approach – 1/19/2013 PICS-NE Level 4 deliverable QRL-N/A.
   b. Conduct SME core implementation team meeting on or prior to 2/18/2013.
   c. Submit draft report entitled "Draft Used Fuel Performance Modeling, Simulation and Experimental Integration RD&D plan" documenting proposed approach – 3/2/2013 PICS-NE Level 3 deliverable QRL-N/A. Submit final report entitled "Used Fuel Performance Modeling, Simulation and Experimental Integration RD&D plan" documenting proposed approach – 4/1/2013 PICS-NE Level 2 deliverable QRL-N/A.
2. Submit whitepaper identifying transportation literature review findings as well as path forward recommendations for review, comment, and comment incorporation – no later than 3/1/2013. Submit Final whitepaper to be formally incorporated into report featured in Task 6 - no later than 3/8/2013.
3. Submit whitepaper identifying material literature review findings as well as path forward recommendations for review, comment, and comment incorporation – no later than 3/1/2013. Submit Final whitepaper to be formally incorporated into report featured in Task 6 - no later than 3/8/2013.
4. Submit whitepaper identifying M&S methodology literature review findings as well as path forward recommendations for review, comment, and comment incorporation – no later than 3/1/2013. Submit Final whitepaper to be formally incorporated into report featured in Task 6 - no later than 3/8/2013.
5. Submit whitepaper establishing M&S development and evaluation results status – 6/14/2013. Submit preliminary reporting of results and contribution to development of report "Used Fuel Transportation & Storage M&S Methodology, Basis, and Results" to be submitted for – 8/16/2013 PICS-NE Level 4 deliverable QRL-N/A. Submit Preliminary reporting contribution – no later than 7/26/2013.
6. Submit final encompassing report derived from Tasks 2, 3, and 4, with entire content combined entitled "Supporting Data, Material Properties, and Modeling Input Information for Used Fuel Transportation & Storage Evaluations" – 3/16/2013. PICS-NE Level 4 deliverable QRL-N/A.
7. Additional Optional Subtasks Under Task 5 for Remainder of Contract

Continue to support and contribute to activities identified in Task 5 as funding and schedule permit for model development of fuel, system, and loading variants. Other additional specific tasks may be requested in writing via task order at a later time.

C-1.4 ENVIRONMENTAL, SAFETY AND HEALTH (ES & H)

Both ORNL and PNNL are committed to the goal of zero accidents through continuous improvement processes and to conduct work safely and responsibly; ensure a safe and healthful working environment for workers, contractors, visitors, and other on-site personnel; and protect the health, safety, and welfare of the general public. Safety will never be compromised for personal, programmatic, operational, or any other reasons.

There are no special ES&H hazards, controls, and requirements identified in this statement of work. PNNL staff shall work under and comply with PNNL ES&H requirements.

The technical point of contact at PNNL is Harold Adkins (509-372-6629)

The Technical Project Officer (TPO) at ORNL is John Wagner (865-241-3570)

C-1.5 Distribution Breakdown

Harold Adkins - PM/Task Lead/Technical Contributor

- $160k - 40% FTE for duration of 10 months

Steve Maheras - Transportation Lead

- $52k - 5.0 man-weeks for transportation literature search and documentation of findings
- $12k - 1.0 man-week for contribution to development of RD&D Implementation Plan

Brian Koeppel - M&S Lead/Technical Contributor

- $45k - 5.0 man-weeks for M&S methodology literature search and documentation of findings
- $150k - 17.0 man-weeks of support for M&S development and assessments (could cover up to two analysts short term but is more so intended to initiate a strong start and continuation of core analytical work)
- $20k - 2.0 man-week for contribution to development of RD&D Implementation Plan

Ken Geelhood - Material Properties Lead/Technical Contributor

- $40k - 4.5 man-weeks for material property literature search and documentation of findings
- $20k - 2.0 man-week for contribution to development of RD&D Implementation Plan

Carl Beyer - Material Properties Technical Contributor

- $45k - 5.0 man-weeks for material property literature search and documentation of findings

Total funding required to be allocated for this MPO $544k.

C-1.6 Breakdown of M&S Work Scope

This work scope documents the objectives, tasks, schedule, and budget for PNNL's modeling work in support of the technical activities described in Section 4.3.3 of the draft plan "Used Nuclear Fuel Loading and Structural Performance Under Normal Conditions of Transport- Draft Modeling, Simulation and Experimental Integration Plan."

C-1.7 Objectives

The objectives of the Modeling and Simulation Team's activity is to develop a modeling methodology for analysis of a spent fuel package under loads characteristic of normal conditions of transport (NCT) with rail cars. The modeling approach utilizes submodels to evaluate the dynamic fuel response at the cask/basket, fuel assembly, and individual fuel pin levels. PNNL is proposing to perform 3D model construction at the fuel assembly level and analyses of NCT loading conditions.

C-1.8 Tasks

<u>Task 1: Detailed Fuel Assembly Modeling</u>: This task will develop a detailed finite element submodel of a single 17x17 PWR fuel assembly for evaluation of NCT shock and vibration loads. The modeling approach proposes to use a 3D beam element representation for all of the fuel pins in a single assembly based on input from the materials and single rod-level modeling teams. This detailed assembly model will be used to i) develop surrogate assembly definitions for use in the cask-level model and ii) analyze the assembly response under NCT based on loads/boundary conditions derived from the cask-level model. Modeling information will be passed to other team members as developed to facilitate successful progress for model development and analysis at each level.

- Develop an LS-DYNA finite element model of a single fuel assembly that includes fuel pins, spacer grids, guide tubes, and end plates within a single basket compartment.
- Work collectively with the materials team to define model material properties characteristic of low, mid, and high levels of burn-up.
- Work collectively with the INL modeling team to develop surrogate beam properties from the detailed single fuel rod model for use in the detailed fuel assembly model.

- Work collectively with the SNL modeling team to develop surrogate assembly definitions from the detailed assembly model for use in the cask level models.
- Work collectively with the model validation team simulating the SNL shaker table test to ensure that the assembly modeling approach is sufficient to capture the fuel rod and assembly dynamics.
- Utilize predicted basket compartment motions from characteristic NCT shock and vibration loading from the cask model as boundary conditions for the single fuel assembly model.
- Evaluate the assembly response under NCT shock and vibration loading.
- Evaluate the model sensitivity to different burn-up properties, spacer grid spring stiffness values, or other key parameters as indicated by the UQ effort.
- Work collectively with the materials team to provide fuel rod loads and deformations for subsequent cladding evaluation and failure analyses.

C-1.9 Schedule

<u>Deliverable: Modeling approach report</u> 5/1/2013

Prepare a report describing the fuel assembly modeling approach and development path to support the 5/7/2013 Level 4 milestone.

<u>Deliverable: Fuel assembly analysis report</u> 8/1/2013

Prepare a report describing the fuel assembly model and results to support the 8/31/2013 Level 3 milestone.

C-1.10 Staff Biographies

Harold E. Adkins, Jr. – Senior Research Engineer

Pacific Northwest National Laboratory – Radiological and Nuclear Science & Technology Division

(509) 372-6629; Email: harold.adkins@pnnl.gov

<u>Education and Training</u>

- B.S. in Mechanical Engineering, University of Wyoming, Laramie, WY, 1991
- M.S. in Mechanical Engineering, University of Wyoming, Laramie, WY, 1997

<u>Research and Professional Experience</u>

Harold Adkins currently provides technical and program management on US Nuclear Regulatory Commission and Department of Energy projects for a multi-discipline team of analysts responsible for performing advanced thermal-hydraulic and nonlinear structural evaluations of commercial spent nuclear fuel transport, transfer, and storage systems. Harold has over 20 years of thermal hydraulic and structural analytical experience, is serving as a key computational/analytical contributor in both fields, and provides task leadership for a multi-discipline, multi-directorate team of computational analysts responsible for performing extra regulatory/vulnerability evaluations. Harold also serves as PNNL's liquid metal Magnetohydrodynamic (MHD) based Annular Linear Induction Pump technology expert with over twenty years of experience in this field and has provided support to NASA/JPL regarding the Jupiter Icy Moons Orbiter program (AKA, Project Prometheus). Harold began his career at PNNL in 2000 when he joined International Security and Nonproliferation providing engineering support to development programs for large caliber concept munitions, die casting technologies, and non-lethal acoustic technologies. Prior to coming to the Lab, Harold provided consulting services in the field of thermal hydraulics for Q-Metrics, Inc. out of Kirkland Washington, and Westinghouse Hanford in Richland.

Kenneth Geelhood – Senior Research Engineer

Pacific Northwest National Laboratory – Nuclear Systems Design, Engineering, and Analysis

(509) 372-4556; Email: kenneth.geelhood@pnnl.gov

Education and Training

- B.S. in Mechanical Engineering, Calvin College, Grand Rapids, MI, 1999
- M.S. in Materials Science and Engineering, Michigan State University, East Lansing, MI, 2002

Research and Professional Experience

Mr. Geelhood has worked on the development and maintenance of the Nuclear Regulatory Commission (NRC) steady state fuel analysis code, FRAPCON-3, and the transient fuel analysis code, FRAPTRAN. This work includes the development of models to predict the thermal and mechanical responses of nuclear fuel rods (UO_2 and MOX) throughout the life of the rods. For example, fission gas release, pellet/cladding mechanical interaction and rod failure models have recently been developed. This work also involves the analysis of various accident scenarios such as loss-of-coolant accident and reactivity initiated accident.

Other activities that Mr. Geelhood is involved in include providing independent assessment of vendor fuel codes and other NRC-Nuclear Reactor Regulation (NRR) licensing submittals and the development of models to predict the mechanical properties of zirconium alloy cladding both for in-reactor and spent fuel storage applications.

Brian Koeppel, Ph.D. – Senior Research Scientist

Pacific Northwest National Laboratory – Radiological and Nuclear Science & Technology Division

(509) 372-6816; Email: brian.koeppel@pnnl.gov

Education and Training

- B.S. in Mechanical Engineering, Michigan Technological University, Houghton, MI, 1993
- Ph.D. in Mechanical Engineering-Engineering Mechanics, Michigan Technological University, Houghton, MI, 1997

Research and Professional Experience

Dr. Koeppel has over 15 years of experience in implicit and explicit finite element tools (e.g. ANSYS, MARC, LS-DYNA), nonlinear finite element analysis, structural modeling, thermal modeling, and thermal stress analysis. In support of the U.S. Nuclear Regulatory Commission's Spent Fuel Project Office (SFPO), he has performed advanced thermal analyses including mass, convective, and radiation transfer mechanisms for safety verification of commercial spent nuclear fuel systems and packaging. He has also performed advanced thermal and structural modeling in support of hypothetical and beyond-hypothetical accident conditions. He also led model development for analysis of cladding integrity under end impact during handling. These activities aid risk-informed decisions that affect public safety regarding potential accidents or terrorist threats to SNF transport and storage systems.

Dr. Koeppel has also led diverse engineering modeling and simulations for other systems including design of solid oxide fuel cell structures, design of fuel cell sealing systems, multi-physics flow-electrochemical-thermal analysis of solid oxide fuel stacks, development of software for fuel cell analysis, vibration of fuel-cell based auxiliary power units, transient mechanical reliability analyses of fuel cell systems during start-up and operations, structural-reliability analyses of fuel cell, solar power, and sealant systems, flow-electrochemical-thermal modeling of flow batteries, and structural-dynamic modeling of kinetic energy penetrators.

Nick Klymyshyn – Senior Research Engineer

Pacific Northwest National Laboratory – Radiological and Nuclear Science & Technology Division

(509) 375-2772; Email: nicholas.klymyshyn@pnnl.gov

Education and Training

- B.S. in Mechanical Engineering, Michigan Technological University, Houghton, MI, 1995
- M.S. in Mechanical Engineering, Michigan Technological University, Houghton, MI, 1996

Research and Professional Experience

Mr. Klymyshyn is an experienced research engineer with core capabilities in mechanical design, analysis, and numerical modeling. He has more than 15 years of experience in application of finite element methods to solve challenging engineering problems, including 10 years of experience supporting the U.S. Nuclear Regulatory Commission's Spent Fuel Project Office (SFPO), Office of New Reactors (NRO), and Office of Nuclear Reactor Regulation (NRR) as both a researcher and technical reviewer.

He has extensive experience in simulation of dynamic impacts events such as drop of spent fuel rods during handling, advanced techniques for analysis of cladding integrity during drop of SNF casks under regulatory hypothetical accident conditions, evaluation of SNF casks during beyond-hypothetical accident conditions (e.g., tunnel fires, bridge collapse), fracture analysis of SNF casks, vibration analyses or fuel rods and assemblies in the reactor core or in storage, and design of robust sealed sources. He has been sought as an independent technical reviewer of NRC applications regarding cask impact, control rod design, and Mitsubishi's FINDS code for seismic or LOCA events. He has also participated in cask thermal analyses for both hypothetical and beyond-hypothetical conditions.

In other projects, his numerical analyses have led to greater understanding of complex material behaviors in advanced manufacturing technologies such as induction melting/drawing of titanium wire, die casting of aluminum, forming of woven thermoplastic composites, and electromagnetic forming. He also has specialized FEA experience in electromagnetic analyses to support improvements to performance of radiation detector arrays and field enhancement for atomic force microscopy. Mr. Klymyshyn has also led analysis and design efforts for pressure vessels, tanks, composite structures, fuel cells, microchannel heat exchangers, and kinetic energy penetrator/sabot design.

Scott Sanborn, Ph.D. – Research Engineer

Pacific Northwest National Laboratory – Radiological and Nuclear Science & Technology Division

(509) 375-2809; Email: scott.sanborn@pnnl.gov

Education and Training

- B.E. in Civil and Environmental Engineering, Stevens Institute of Technology, Hoboken, NJ, 2004
- M.A. in Civil and Environmental Engineering, Princeton University, Princeton, NJ, 2008
- Ph.D. in Civil and Environmental Engineering, Princeton University, Princeton, NJ, 2010

Research and Professional Experience

2009 – present, Research Engineer at PNNL. Since joining PNNL Dr. Sanborn's research has focused on modeling of materials and structures for nuclear power plants and nuclear waste storage. At PNNL he has been involved in the structural integrity modeling of the Hanford nuclear waste storage tanks. In this work, the continuum mechanical properties of tank concrete (including the effects of cracking, crushing, thermal creep and degradation) are accurately modeled to determine the structural performance of the tanks subject to the thermal cycles of the nuclear waste as well as operating and seismic loads. Additionally, Dr. Sanborn was part of the institutional computational team that developed the USNRC's Extremely Low Probability of Rupture (xLPR) software to predict failure probabilities in nuclear power plant piping welds. The xLPR software predicts probabilities of crack initiation, crack growth and coalescence, and pipe weld failure by embedding physics based models for each stage of weld damage within a Monte Carlo routine. He has also worked on predictive modeling of crushing failure in liquid crystal polymer foams and design and analysis of connections for the tritium production program.

2005-2009, Graduate Research Assistant at Princeton University. Prior to joining PNNL, Dr. Sanborn was a graduate student at Princeton University in the Mechanics, Materials, and Structures program of the Civil Engineering Department. While there he completed his doctoral work on modeling failure in inelastic materials. Specifically his research focused on implementing shear band and slip plane localization of plastic materials into continuum level finite element programs. Additionally, his research also focused on modeling microstructure level material failure mechanisms to predict continuum level behavior of functionalized graphene sheet nanocomposites using finite element models and lattice models.

Naveen Karri – Research Scientist

Pacific Northwest National Laboratory – Radiological and Nuclear Science & Technology Division

(509) 372-4442; Email: naveen.karri@pnnl.gov

Education and Training

- B.T. in Mechanical Engineering, JNT University College of Engineering, Hyderabad, India, 2001
- M.S. in Mechanical Engineering, University of Alaska Fairbanks, Fairbanks, AK, 2004

Research and Professional Experience

Naveen Karri is a key researcher in the fields of computational mechanics and thermal systems design. He has extensive experience in finite element methods (FEM) applied to multidisciplinary mechanical, thermal and structural engineering fields. He is adept at applying first principles and solving engineering problems according to established standards. Mr. Karri also has significant experience relevant to nuclear power plant license renewal reviews and experiments related to Hanford nuclear waste treatment plant. He recently served as an independent technical reviewer for the spray leak testing project to detect the amount of aerosol release in the event of a leak from waste transfer pipes. In the past Mr. Karri served as a co-principal investigator, technical lead and key contributor to several government, laboratory and private industry funded projects. Mr. Karri is a competent engineer with a proven track record success and received several outstanding performance awards (Seven in the last 5 years) while working on multiple research projects. The program that Mr. Karri developed during his LDRD project studying stochastic effects on thermoelectric system performance served multiyear projects from U.S Department of Defense, U.S Army Logistics Innovation Agency and NASA Jet Propulsion Laboratory. He also holds an invention disclosure on thermoelectric systems design optimization.

Mr. Karri's core competencies and research interests include: Linear and nonlinear finite element methods applied to diverse fields, multi-physics simulations, thermoelectric systems design and modeling, nuclear waste treatment (vitrification) plant design related experiments, computational modeling of material and energy systems, numerical methods, scientific programming tools usage and spreadsheet simulations.

C-2. Sandia National Laboratories

ModSim RD&D Implementation Plan Work Scope for Sandia National Laboratories

C-2.1 Overview

This work scope documents objectives, tasks, schedule, and budget information for the work that Sandia National Laboratories will conduct to support the Used Nuclear Fuel Loading and Structural Performance Under Normal Conditions of Transport –Draft Modeling, Simulation and Experimental Integration RD&D Plan, April 1, 2013. The identified tasks flow directly from Sections 4.1 – 4.3 of this plan.

C-2.2 SNL Objectives

The objectives of the SNL work scope will be to support the Modeling and Simulation development of a spent fuel rail package subjected to the normal conditions of transport. The result of this work will support the assessment of the used fuel assembly's capability to withstand the Normal Conditions of Transport (NCT) for rail.

C-2.3 SNL Tasks

Section 4.1 of RD&D Implementation Plan:

- Literature Review: This work is largely completed with the preparation for and conduct of the SME meeting held in Las Vegas, Nevada in February.
    - Transportation Loadings
      personnel: Paul McConnell
    - Materials Properties
      personnel: Remi Dingreville

- Validation Information - Retrieve and establish viability of using published UNF transport campaign study results (Prulhiere and Israel 1980) as a possible ModSim validation case
  personnel: Remi Dingreville

Section 4.2:

- TTCI Collaboration: This work will advance the literature review work to assess existing TTCI rail shock and vibration data for use for the RD&D Implementation Plan scenario. Collaboration with TTCI will result in the selection of shock/vibration response spectra and loading time history data that will be used as loadings to the cask. This will include the loading as well as the point of interface/transfer of the load to the cask body. This will require close collaboration with the analysis team.
  - Compare data provided by TTCI to literature review data such as Magnuson and document findings
    personnel: Paul McConnell
  - Collectively work with supporting staff at other participating labs to determine TTCI /ModSim loading interfaces and specification of simulation configuration
    personnel: Paul McConnell

Section 4.3:

- Modeling Development: This task will focus on the modeling and analysis of the cask body, canister, and basket responses to the loading functions provided by the TTCI collaboration effort. The results of this analysis will then be provided as input to the assembly modeling and analysis team. In addition, this task will provide support to the definition of a failure criterion for this specific problem. Associated with this is a failure criterion analysis that will help establish quantitative estimates of the safety margin related to the integrity of high burnup fuel subjected to the NCT input loadings.
  - Cask/Canister/Basket response analysis
    personnel: John Bignell, Gregg Flores, Doug Ammerman
    - Work collectively with PNNL ModSim staff to convert TTCI data into applicable shock & vibration as well as loading time history forms
    - Model creation/application
      - Geometry will consist of the GBC-32 definition
    - Analyses and results processing
    - Identification and reporting of worst case fuel compartment loading to surrogate for sensitivity loading spans identified by multi-lab ModSim team

        - Surrogate information/definition will be collectively developed with key PNNL ModSim staff
    - Final review
- Failure criteria analysis
  personnel: Remi Dingreville, Doug Ammerman
    - Analysis of driving input parameters for failure criteria based on a tensile ductility model and fracture toughness model – incorporate consideration of burnup span as defined in cited RD&D plan (30 to 58 GWd/MTU assembly average burnup)
    - Response analysis of a fuel pin to determine the steady-state behavior when considering applicable frequency and amplitude ranges
    - Sensitivity analysis will be performed on key materials properties
    - Validation with Argonne data (Billone), Oak Ridge data (Howard, Wang) and PNNL materials group data (Geelhood, Bayer).
    - Develop/analyze failure criteria appropriate for fuels with values ranging from 30 to 58 GWd/MTU assembly average burnup
    - Final review

C-2.4 SNL Budget and Schedule

Task:	**Initial $100K MPO**	**New Funding**
Task 1: Literature Review	$40K	

- Transportation Loadings
- Material Properties
- Validation Information

Task 2: TTCI Collaboration	$40K	

- Compare TTCI data to lit review data
- Collectively work with supporting staff at other participating labs to determine TTCI/ModSim loading interfaces and specification of simulation configuration
- Perform TTCI loading case evaluation on shaker/mockup

Deliverable: Level 3 milestone. Letter report detailing SNL support of literature review and TTCI collaborations due June 30, 2013

Task 3: Modeling Development

o Cask/Canister/Basket response analysis /surrogate assembly definition(s)[w/PNNL]	$10K	$205K
o Failure criteria/single rod harmonics		$75K

Deliverable: Level 3 milestone. Analysis report detailing the results of the cask/canister/basket analysis due August 1, 2013

TRAVEL/MEETINGS	$10K	$20K
Total Funding	**$100K**	**$300K**

Note: Deliverable deadlines represent final reports. It is expected that information will be passed to other team members as it is developed and as it is needed to integrate into the entire modeling effort.

C-2.5 SNL Biographies

Doug Ammerman: Doug Ammerman has BS, MS, and PhD degrees in Civil Engineering from the University of Minnesota. His MS and PhD work was on the behavior of semi-rigid steel-concrete composite connections, funded by the American Institute of Steel Construction. For the past 24 years he has worked at Sandia National Laboratries within the general area of radioactive material transportation. Within this area he is considered internationally as a subject matter expert, with emphasis on determining the structural response of packages to accident environments via analysis and testing. He is the leader for the transportation package testing program and the structural analysis programs at Sandia. He has over 100 publications in areas related to structural response of structures (including transportation and storage packages, ships, and buildings) under extreme accident conditions. His expertise is utilized in nearly every program development/program planning activity within the transportation group. Doug is usually chosen to represent the transportation program when high-ranking officials from the US or foreign governments visit Sandia. Past examples include President Bush, NRC commissioners, DOE officials, French regulators, the NAS, a delegation from Singapore, and Korean utility representatives. Doug's expertise is also frequently sought by other national labs, such as INL, LANL, ANL, SRNL,Bettis Atomic Power Lab, Knolls Atomic Power Lab, and DOE sites such as Y-12, Hanford, and the former RFETS. He is a member of two ASME Boiler and Pressure Vessel Code committees providing advice and guidance on the use of inelastic analysis for radioactive material transporation packages and was a founding member of an ASME committee on use of finite element analysis methods for structures under large deformations.

Doug was the Sandia program manager and lead author for NUREG-2125, Spent Fuel Transportation Risk Assessment. This work utilized explict dynamic finite element analyses to determine the threshold against failure for NRC certified spent fuel transportation casks. He was one of the lead authors on the investigation of aircraft impacts into transportation and storage casks following the 9/11 terrorist attacks.

Doug is internationally renown for his expertise in the area of radioactive material transportation. He has been one of the leaders of the paper selection meeting for the PATRAM conference (held every three years) the past three times and a member of the paper selection committee twice before that. For PATRAM 2013 he is the Techincal Program Chariperson. He has been session organizer and/or session chair at PATRAM, the INMM Annual Meeting, the ASME Pressure Vesels and Piping conference, and the ASME Annual Meeting. His work has received best paper awards at four of the past five PATRAM meetings. He has taught tailored seminars on radioactive material transportation as part of an IAEA training course and to a delegation of Korean utility engineers.

John Bignell: John Bignell is a Principal Member of the Technical Staff at Sandia National Laboratories (SNL), Albuquerque, New Mexico, USA and is currently in the Structural and Thermal Analysis Department within the Nuclear Energy and Fuel Cycle Programs Center. He holds a BS in Civil and Environmental Engineering from the University of Utah (2000), and an MS and PhD in Civil and Environmental Engineering, with a certificate in Computational Science and Engineering, from the University of Illinois at Urban-Champaign, USA (2001 and 2006, respectively). As a research assistant at the University of Illinois Dr. Bignell developed analytical structural fragility curves for wall pier supported highway bridges typical of those found in southern Illinois for use in seismic risk assessment analyses. Upon his graduation, Dr. Bignell joined the Spacecraft Structures and Dynamics Group at NASA's Jet Propulsion Laboratory (JPL) in Pasadena, California, USA where he was the lead structural engineer for the rover mobility system on the 2.5 billion dollar Mars Science Laboratory (MSL) mission which recently successfully landed on the planet Mars. His responsibilities at JPL included the design, structural analysis, and test verification of components of the rover chassis and mobility systems. In 2010 Dr. Bignell joined SNL, where he specializes in the application of numerical and analytical methods in the design and assessment of structures subject to extreme load events. While Dr. Bignell has been involved in a number of projects, his primary role since joining SNL has been team lead for the Radioisotope Power System Launch Safety (RPSLS) program's Blast and Impact (B&I) team which is responsible for characterizing the response of various radioisotope power systems to blast and impact events resulting from possible launch accident scenarios.

Rémi Dingreville: Rémi Dingreville is a Principal Member of the Technical Staff at Sandia National Laboratories. Rémi currently is the lead on the xLPR (Extremely Low Probability of Rupture) project at Sandia National Laboratories for the Nuclear Regulatory Commission. He has relevant experience in physically-based constitutive modeling for non-linear and time-dependent behavior of structural materials and multiscale modeling. Rémi has worked on materials modeling for high-strain rate behavior of composite materials, nuclear fuel swelling, irradiated materials, ratcheting and fatigue of structural materials, biological materials, microstructured materials, and other projects related to constitutive modeling of materials for various applications and environments. Rémi received his Ph.D. degree in mechanical engineering from the Georgia Institute of Technology with a specialty in theoretical mechanics and computational materials science and served as a postdoctoral appointee at Sandia from 2007 to 2009. He then served on the Mechanical Engineering faculty as an assistant professor at the Polytechnic Institute of New York University from 2009 to 2011, at which time he was hired permanently as a staff member at Sandia. Rémi was recently awarded the 2013 Young Leader by the Minerals, Metals and Materials Society (TMS), a professional society that promotes global science and engineering in the field of materials science.

Gregg Flores: Gregg Flores began his career at Sandia National Laboratories as a technical year-round student intern while still in high school. He supported various projects within the Severe Accident Analyses groups throughout his undergraduate degree both in experimental and analytical aspects. Upon completion of his undergraduate degree in Civil Engineering at the University of New Mexico, Gregg was awarded the Masters Fellowship by Sandia National Laboratories and completed the Structural Engineering and Structural Mechanics master's program at the University of Colorado at Boulder in December 2011. Gregg's main focus is in the field of computational mechanics and is currently supporting the safety analysis of the Advanced Sterling Radioisotope Generator (ASRG) which powers the Mars Science Laboratory and potentially other space-bound systems. Gregg also supports modeling of nuclear material safeguards, nuclear reprocessing modeling, and transportation surety design and analysis for sensitive assets.

Paul McConnell: Paul McConnell is currently Transportation Team Leader for the Used Fuel Disposition Campaign, at Sandia National Laboratories. He has relevant experience in the transportation of nuclear waste, as Principal Investigator, Project Manager, and individual contributor to many projects. Paul has worked on probabilistic risk assessment for dry cask storage, aircraft impact assessment, transportation cask specification and design, transportation system testing, neutron absorption materials, reactor vulnerability assessment, repository licensing, cobalt contamination and non-fixed contamination of waste packages, hydrogen getters and sensors, radioactive material shipment by sea, and other projects related to transportation and the back-end of the nuclear fuel cycle. Paul holds degrees in metallurgy and metallurgical engineering with a specialty in mechanical metallurgy and fracture mechanics.

C-3. Idaho National Laboratory

C-3.1 Purpose

The tasks proposed in this work scope support the technical activities described in Section 4.3.3 of the draft plan "Used Nuclear Fuel Loading and Structural Performance Under Normal Conditions of Transport- Draft Modeling, Simulation and Experimental Integration RD&D Plan." The purpose of this work scope is to provide the Pacific Northwest National Laboratory (PNNL) modeling and simulation team lead with a modal analysis (natural frequency evaluation) of a single fuel pin and a PWR assembly (this analysis provides important information on what frequencies need to be checked in the PWR assembly model), and recommendations on model optimization (i.e. what frequencies are important in the model, how long the vibration model should be run, and can the vibration run be performed in an implicit solver or explicit solver). INL will also provide estimated beam stiffness and damping parameters that can be used in the assembly model. This will be done by performing an explicit analysis of the 3D fuel pin by tracking the free vibration decay after an initial perturbation (this is mainly due to frictional dissipation at the pellet/cladding interface when a gap exists). This effort will identify the frequency ranges of interest and informs the assembly modeling approach for evaluation used fuel under normal conditions of transport (NCT).

C-3.2 Scope

TASK 1: These analyses focus on modal development and application primarily for the detailed single rod model and the assembly model. This identifies the frequency ranges of interest and informs the PNNL shock and vibration modeling approach. The modeling activities will be performed by Justin Coleman and Bob Spears. INL will also maintain participation in the project steering committee and prepare and review project deliverables (Steve Marschman).

The following tasks are included in the scope:

- The primary focus is construction and development of the detailed 3D pin model and satisfactory reduction to beam element representation that captures the dynamic response.
- INL staff may utilize and adapt the existing PNNL beam assembly model for modal analysis.
- Determination of damping parameters for the frequency range of interest will be performed using free vibration analysis.
- The effect of pellet/cladding interaction on the rod natural frequency will be evaluated. This will be done by performing an explicit analysis of the 3D fuel pin and tracking the free vibration decay after an initial displacement perturbation (this is mainly due to frictional dissipation). The coefficient of friction between the pellet and cladding will be varied and Raleigh damping parameters will be developed based on the amplitude decay for each model. These parameters will be used in the modal analysis to determine the influence of friction on the fuel rod natural frequencies. They will also be used in the assembly level model for NCT evaluations.
- The 3D pin model will be made available for use to evaluate stress distribution through the fuel pin cladding cross section for failure criteria evaluations.

C-3.3 Modeling Team/Contributors

Justin Coleman

Civil/Structural Engineer, Risk Assessment and Management Services, INL

Justin Coleman is a licensed professional civil engineer. He performs seismic and structural engineering at the Idaho National Laboratory (INL) and has a master's degree in engineering structures and mechanics. Mr. Coleman's background and expertise is seismic analysis and structural analysis of spent fuel casks and canister impact analysis. Much of the impact analysis work is performed using explicit finite element analysis codes. He has performed linear and nonlinear seismic soil-structure interaction (SSI) analysis for safety-related nuclear structures. His research interests include nonlinear SSI analysis, seismic protective systems, spend fuel transportation and storage, and beyond design basis threats to nuclear structures. He serves on the ASCE 4 where he is the lead author of Chapter 3, "Modeling of Structures," and led the effort to write Appendix B, "Nonlinear Time Domain Soil-Structure Interaction Analysis," in the forthcoming edition of ASCE 4. Mr. Coleman has authored numerous reports on impact analysis of spent fuel casks and nuclear fuel, and seismic analysis; three relevant reports are provided below.

Coleman, J. C., "RWMC seismic stability analysis for three-high and five-high drum stacks," CH2M-WG, 2009.

Coleman, J. C., "Conceptual soil structure interaction analysis for the Calcine Disposition Project (SEN 9.3.2) (Conceptual Design)," CH2M-WG, 2012

Coleman, J. C., "LCC drop analysis in unloading pool #2, against the north wall, and in the cask receiving area with the rigid three-legged bridle assembly attached," CH2M-WG 2012.

Bob Spears

Mechanical/Structural Engineer, Structural Analysis, INL

Bob Spears performs seismic and structural engineering at the Idaho National Laboratory (INL) and has a Ph.D. in Mechanical Engineering. Mr. Spears` background and expertise is in seismic analysis of structures and piping and structural analysis of spent fuel casks and canister impact analysis. Much of the seismic analysis work is performed using linear and nonlinear implicit finite element analysis codes. Much of the impact analysis work is performed using nonlinear explicit finite element analysis codes. He has performed linear and nonlinear seismic analysis (structural and piping) for safety-related nuclear structures. His research interests include nonlinear SSI analysis, seismic protective systems, spend fuel transportation and storage, and beyond design basis threats to nuclear structures. He serves on the ASCE 4 where he has contributed by coauthoring a paper on an approach for selection of Rayleigh damping parameters used for time history analysis. The ideas in this paper are being incorporated into ASCE 4, Section C3.5.1.1. Mr. Spears has authored and coauthored numerous reports and papers on impact analysis and seismic analysis, and; eleven relevant papers are provided below.

Spears, R. E., "Elastic/Plastic Drop Analysis Using Finite Element Techniques," PVP-Vol. 390, Transportation, Storage, and Disposal of Radioactive Materials, ASME 1999.

Spears, R. E., "Finite Element Mesh Sensitivity Study of Low Velocity Elastic/Plastic Impact Using ABAQUS/EXPLICIT," Masters Thesis, Mechanical Engineering, University of Idaho, April 1001.

Russell, M. J., Spears, R. E., and Kobbe, R. G., "Seismic Evaluation of Atypical Special Plate Shear Walls," PVP2007-26369, Proceedings of 2007 ASME Pressure Vessels and Piping Division Conference, July 22-26, 2007, San Antonio, Texas.

Spears, R. E., "Unique Method for Generating Design Earthquake Time Histories," PVP2008-61243, Proceedings of 2008 ASME Pressure Vessels and Piping Division Conference, July 27-31, 2008, Chicago, Illinois.

Spears, R. E., "Unique Method for Generating Design Earthquake Time History Seeds," PVP2008-61244, Proceedings of 2008 ASME Pressure Vessels and Piping Division Conference, July 27-31, 2008, Chicago, Illinois.

Spears, R. E., and Jensen, S. R., "Approach for Selection of Rayleigh Damping Parameters Used for Time History Analysis," PVP2009-77257, Proceedings of 2009 ASME Pressure Vessels and Piping Division Conference, July 26-30, 2009, Prague, Czech Republic.

Crawford, A. L., Spears, R. E., and Russell, M. J., "Eliminating Conservatism in the Piping System Analysis Process Through Application of a Suite of Locally Appropriate Seismic Input Motions," PVP2009-77814, Proceedings of 2009 ASME Pressure Vessels and Piping Division Conference, July 26-30, 2009, Prague, Czech Republic.

Clark, D. T., Russell, M. J., Spears, R. E., and Jensen, S. R., "Adaption of Nonstandard Piping Components into Present Day Seismic Codes," PVP2009-77916, Proceedings of 2009 ASME Pressure Vessels and Piping Division Conference, July 26-30, 2009, Prague, Czech Republic.

Spears, R. E., "Method for Generating Design Earthquake Time Histories with an Emphasis on Maintaining Phasing," PVP2010-25229, Proceedings of the ASME 2010 Pressure Vessels & Piping Division / K-PVP Conference, July 18-22, 2010, Bellevue, Washington.

Spears, R. E. and Wilkins, J. K., "Comparison of Nonlinear Model Results Using Modified Recorded and Synthetic Ground Motions," Div-IV: Paper ID# 161, 21st International Conference on Structural Mechanics in Reactor Technology (SMiRT 21), 6-11 November, 2011, New Delhi, India.

Spears, R. E., and Jensen, S. R., "Approach for Selection of Rayleigh Damping Parameters Used for Time History Analysis," Journal of Pressure Vessel Technol. Volume 134, Issue 6, October, 2012.

Steven C. Marschman

Relationship Manager

Dr. Marschman has a Ph.D. in Materials Science and Engineering. He has over 28 years' experience that spans three national laboratories. His expertise spans nuclear waste issues including spent nuclear fuel and nuclear waste glass, heavy isotope production, strategic planning, nuclear facilities operations, and quality assurance, and project management. Dr. Marschman has authored or coauthored over 100 publications and presentations, and has earned three U.S. and eight foreign patents.

C-3.4 Deliverables

TASK 1: A report will be provided that documents the detailed information as discussed in the scope above (this report could be an appendix in the PNNL report). At a minimum, weekly discussions will be held between members of the analysis teams (PNNL, INL). Also, INL derived results required to feed in to PNNL evaluations will be communicated as they become available to expedite modeling and simulation development.

C-3.5 Schedule

Task 1:

- Preliminary beam properties/surrogate definition (feeding in to assembly modeling): May 1st, 2013
- Preliminary results: May 30th , 2013
- Final document: July 26th, 2013

C-3.6 Funding Required

Task 1: $205,000

C-4. Oak Ridge National Laboratory

Workscope for Single Rod Model Analyses

C-4.1 Background

After fuel is irradiated to high burnup levels, the fuel and clad bond together to form an integrated system. It is recognized that understanding the pellet/clad interface dynamics in a transportation environment is critical when evaluating assembly properties. Currently, a methodology to accurately simulate PWR UNF assembly behavior during NCT has not been developed. Prior to conducting any physical experimentation on high burnup fuel, ORNL will conduct a finite element evaluation of SNF vibration integrity to investigate and document the expected behavior of the fuel/clad system.

C-4.2 Workscope

This sub-model will consist of a continuum representation of a section of a single fuel pin. This model will include a 3D representation of the fuel pin, cladding, and any possible gaps between pellets or pellets to cladding. The fuel rod will be simulated as beam elements with effective stiffness properties. This model will be used to obtain the effective properties of the beam elements that can then be used in the detailed UNF assembly model. The model also aims to estimate the damping properties of the beam approximation due to frictional resistance between the cladding and pellet (to be evaluated as a sensitivity). This SNF fuel-clad interaction modeling will use a “contact element” approach as well as the embedded/prescribed boundary conditions, such as internal pressure and residual stress, etc.

The output from these studies will provide essential information on the effects of pellet-clad (PC) interaction, as well as the impact of pellet fracture and pellet-clad bonding effects to the cladding fatigue strength.

C-4.3 Model Description

Either Abaqus or LS-DYNA will be the code used for this modeling activity. This work will include quasi-static modeling of inertial loads scaled by an amplification factor, a time domain approach (implicit and explicit) to capture the full transient response, modal analysis to identify natural frequencies, and spectral analysis to capture the maximum response based on the frequency content of the excitation.

The material properties used in the model will be obtained from the material property database established and documented by the MP Team. Material assignments will be UO2 for the fuel pellet, and Zircaloy-4 for the cladding.

The load time history will be decoupled into frequency domains using a FFT algorithm. The associated amplitude of each target frequency will then be used as the input loading to perform the FEM analyses. Upon completion of the FEM for the targeted frequency range, all the frequency domain data will be integrated into a time domain output,

C-4.4 Description and Duration of Work

This work will consist of the following significant activities:

1. Develop the detailed rod model to evaluate pellet/clad interactions and associated interface bonding conditions under vibration loading. This work will include working cooperatively with other teams to utilize their existing information where available to select/prioritize the target frequencies and the associated amplitudes from the load-time history data and establish the boundary condition loads experienced by an individual rod. (15 weeks)
2. Evaluate resultant simulation data, and benchmark it with the fatigue aging data obtained from the reversal bending fatigue tests that have already been run using surrogate rods; determine if other frequencies or amplitudes should be modeled. Model additional frequencies and amplitudes as indicated by results. (5 weeks)
3. Write report documenting the model and results. (2 weeks)

C-4.5 Deliverables

This modeling effort will provide the stress and strain aging evolution of the fuel/clad during simulated rail NCT. Output data from the model for elastic modulus, Poisson's ratio (or shear modulus), yield strength, ultimate tensile, elongation, and flexural rigidity will be compared with information in the existing database.

There are two deliverables/milestones as described below:

Milestone:

1. Identify the loading conditions for an individual rod; complete a localized rod FEM model to support further investigation on the fuel rod aging evolution.

   Deliverable: Report on information collected and assumptions on boundary conditions to be used in establishing the individual rod loading conditions. (May 10, 2013)

Milestone:

2. Complete the FEM simulation on an individual fuel rod aging evolution and benchmark the FEM analysis with the vibration fatigue data obtained from surrogated rod.

   Deliverable: Final report with all results from the FEM simulations. (August 16, 2013) However, preliminary results will be communicated as they become available. Near-final (undocumented) results will be made available/communicated no later than August 1, 2013.

Properties that will be modeled for fuel/cladding system

Property	Model Output	Sensitivity	Existing Information
Elastic Modulus			
Yield Stress			
Ultimate Tensile Strength			
Uniform Elongation			
Total Elongation			
Aging Index			
Flexural Rigidity			
Interfacial Bonding			
Damage Index			

C-4.6 Schedule

This work will be completed by August 16, 2013 utilizing the duration of activities noted above. However, preliminary results will be communicated as they become available. Near-final (undocumented) results will be made available/communicated no later than August 1, 2013.

Funding

The modeling team will be led by Dr. Jy-An Wang. Other team members will be Dr. Hong Wang, and Dr. Hao Jiang with support from Bruce Bevard.

Estimated labor costs will be $150K.

C-4.7 Attachment 1 (Bios of modeling team)

(Full resumes available upon request)

Jy-An John Wang, Ph.D., P.E.

Distinguished R&D Staff

Materials Science and Technology Division

Oak Ridge National Laboratory, One Bethel Valley Rd, Oak Ridge, TN 37831

Phone: 865-574-2274; Fax: 865-574-6098; Email: wangh@ornl.gov

Education:

University of California, Berkeley: Division of Structural Mechanics and Materials and Structural Engineering of Civil Engineering Department; Major: Mechanics of Materials, Minors: Structural Dynamics and Numerical Analysis; Ph.D.: 1988.

Professional Experience:

1984-1988 - University of California, Berkeley, Research Assistance and Research Engineer.

1989 - Present: Oak Ridge National Laboratory, Distinguished R&D Staff, Group Leader.

HONG WANG Ph.D.

Materials Science and Technology Division

Oak Ridge National Laboratory, One Bethel Valley Rd, Oak Ridge, TN 37831

Phone: 865-574-5601; Fax: 865-574-6098; Email: wangh@ornl.gov

Citizenship: P. R. China

Immigrant Status: U.S. Permanent Resident since 11/20/2007

Education and Training

Michigan Tech Univ.	Mech. Eng.-Eng. Mech., Ph.D. Aug. 2001
WUHEE	Geotechnical Engineering, D.E. Nov. 1991
WUHEE	Geotechnical Engineering, M.E. June 1986
Wuhan Univ. of Hydraulic & Electric Engineering (WUHEE)	Hydraulic & Hydroelectric Engineering, B.E. July 1983

Professional Experience

April 2011 to present	R & D Staff, Oak Ridge National Laboratory
April 2008 - April 2011	Research Assist. Prof., Univ. of Tennessee
April 2004 - April 2008	Research Assoc., Oak Ridge National Laboratory
Nov 2001 - Mar 2004	Research Fellow, The Johns Hopkins University
Dec 1992 - Nov 1996	Assoc. Prof., WUHEE
June 1986 - Dec 1992	Lecturer, WUHEE

Hao Jiang Ph.D.

Materials Science and Technology Division

Oak Ridge National Laboratory, One Bethel Valley Rd, Oak Ridge, TN 37831

Phone: 865-574-5712; Fax: 865-574-6098; Email: jiangh@ornl.gov

Citizenship: P. R. China

Immigrant Status: U.S. Permanent Resident since 04/27/2012

Education and Training

Purdue University, West Lafayette, IN	Mech. Eng., Ph.D. Dec. 2008
Tongji University, Shanghai, China	Mech. Eng., M.S. March 1998
Tongji University, Shanghai, China	Mech. Eng., B.S. July 1995

Professional Experience

April 2011 to present	Research Associate, Oak Ridge National Laboratory
April 1009 – Jan. 2010	Vibration & Mechanics Engineer, the Trane company, Ingersoll Rand, WI.
Aug. 2004 – Dec. 2008	Research Assistant, Purdue University, IN.
March 1998 - May 2002	Research Faculty, Tongji University Shanghai, China

ORNL Transportation Technologies Group Support to *Used Nuclear Fuel Loading and Structural Performance Under Normal Conditions of Transport* Project

C-4.8 Introduction

The ORNL Transportation Technologies Group will provide support to the UFD Structural Performance Under Normal Conditions of Transport project. This support will be in an oversight, review and confirmation role. The project is a short-term project, to be completed by the end of fiscal year 2013, so any support provided will need to be timely and efficient. Task descriptions, resource requirement estimates and schedule information are provided below.

Task 1 – Review of applicable literature

As described in the Method and Approach document for this project, a literature search is currently ongoing in three specific areas – applicable boundary, loading and regulatory conditions, material properties, and modeling and simulation methodologies. To provide effective review of studies performed and subsequent documentation of such, the literature that is found through this search will need to be absorbed by the prospective reviewer. It is estimated that 40 hours would be used in the review of papers that are found through the literature search. PI: Oscar Martinez. Schedule – Completed by 4/30/13.

Task 2 – Review of Proposed Methodologies

As described in the Method and Approach document for this project, M&S methodologies will be developed for structural and thermal analyses based on both the results of the literature search and input from subject matter experts. It is anticipated that interim documents will be developed that will detail M&S methodologies to be used. ORNL TTG will perform a technical review of documents produced during this stage of the project and will provide comments back to the project lead. It is estimated that such review will take 40 hours of effort. PIs: Matt Feldman and Oscar Martinez. Schedule – as available – comments issued within 10 calendar days after receipt of documents for review. It is anticipated this task will be completed by the end of June 2013.

Task 3 – Review of Experimental Planning

As described in the Method and Approach document for this project, experiments will be planned and to the extent possible in the timeframe given, however, it is doubtful that enough time will exist to carry them out. ORNL TTG will brainstorm concepts for possible experiments that would be of use to the outcome of this project. It is estimated that this task will take up to 20 hours of effort. PIs: Matt Feldman and Oscar Martinez. Schedule – Initial brainstorming of experimental ideas will be completed and documented by 6/30/13.

Task 4 – Review of Methodology Demonstration

As described in the Method and Approach document for this project, the primary output (level 2 milestone) for this project is the delivery of documentation that provide a demonstration of the successful use of the methodologies developed during this project. ORNL TTG would provide a detailed review of this document and will provide comments back to the project lead. It is estimated that such review will take 60 hours of effort. PIs: Matt Feldman and Oscar Martinez. Schedule – Activities associated with this task will be completed to support milestones M4FT-13OR0822017 on 8/16/13, M4FT-13OR0822018 on 8/16/13, and M3FT-13OR0822016 on 8/31/13. It is anticipated the bulk of this work will be performed in July and August 2013.

Task 5 – Confirmatory Analysis

Similar to the fashion in which confirmatory analyses are performed during the certification process for proposed radioactive material shipping packages, TTG will perform confirmatory analyses for the M&S portions of this project. This task will entail using the various models developed by the core team, independently merging them together as a single model and then performing confirmatory analysis to show that the project's end product can be used by an autonomous analyst. With this approach, the project's benefit to the UFD community can be confirmed. This task would require 200 hours of effort. PI: Oscar Martinez. Schedule – this task will be performed as soon as the various FEA structural models are available from the developers; however, due to the short project duration it is considered doubtful this task would be complete prior to the level 2 milestone (M2FT-13OR0822015) at the end of September 2013. Projected completion would be as soon as possible and not later than 10/31/13.

C-4.9 Tabular Summary of Proposed TTG Activities

Task	Description	Effort (hrs)	Estimated Cost ($K)
1	Literature Review	40	8
2	Methods Review	40	8
3	Experiment Review	20	4
4	Results Review	60	12
5	Confirmatory Analyses	200	40
	Total	**360 hrs**	**$72K**

C-4.10 PI Bios

Matt Feldman has 25 years of experience working in the field of transportation and packaging of radioactive materials. He is currently the Chair of ANSI N14 – *Transportation and Packaging of Radioactive and Other Hazardous Materials* and the Co-Convener (Vice Chair) of the ISO/TC85/SC5/WG4 – *Transport of Radioactive Materials*. Feldman has experience with thermal modeling of Type B packages as well as development and certification of Type B Shipping Packages. He is the ORNL Package Testing Program Manager and also leads ORNL support of the DOE Packaging Certification Program, the NNSA Office of Packaging and Transportation. Feldman also works closely with the NRC and IAEA in matters related to transport of radioactive materials and is currently the Editor-in-Chief of the international journal *Packaging, Transport, Storage and Security of Radioactive Material*.

Oscar Martinez Ph.D. is an experienced mechanical analyst of metallic and composite systems. He was the technical lead in the advanced space suit project in which LS Dyna impact models were generated to determine protection capabilities of the suit from a fall during moon and Mars missions. Martinez also has experience as a stress analyst for NASA ground, test, and flight hardware, advanced structures technology, and advanced light-weight structural concept projects. Martinez has experience using FEMAP, ABAQUS, NASTRAN as well as LS Dyna and has 1.5 years of experience working in the transport of radioactive materials field.

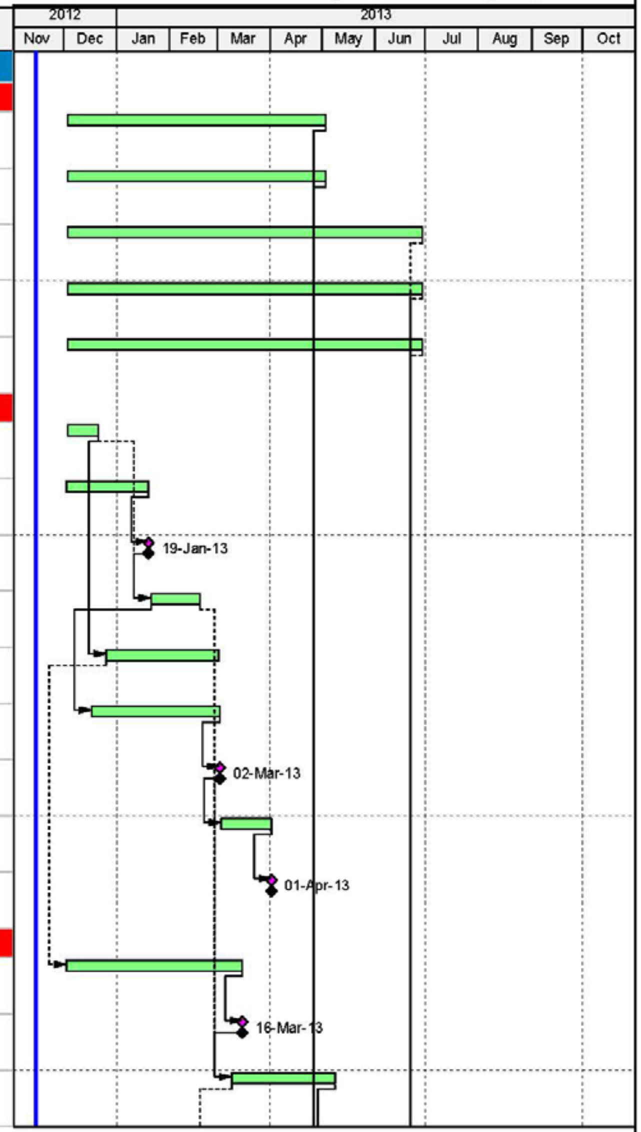
Appendix D

Program Gantt Chart

High Burnup Fuel Performance Modeling Plan Development

Updated: January 31, 2013

Activity ID	Activity Name	Start	Finish
HBUF	**High Burnup Fuel Perfomance Modeling Plan Development**	03-Dec-12	30-Sep-13
HBUF.1	**Complementary UFDC Program Information/Data**	03-Dec-12	28-Jun-13
0085	High burnup UNF bend testing is being conducted on cladding with fuel in place (WBS# 1.02.08.05.04)	03-Dec-12*	03-May-13
240	UNF Data and M&S Toolset Development and Demonstration (WBS# 1.02.09.14.06)	03-Dec-12*	03-May-13
250	Storage and Transportation Engineering Analysis (WBS# 1.02.08.10.02)	03-Dec-12*	28-Jun-13
260	Storage and Transportation Study (WBS# 1.02.08.13.03 & .04)	03-Dec-12*	28-Jun-13
270	Tool for Prediction of Hydride Precipitation and Reorientation during Long-term Dry-Storage (WBS# 1.02.08.10.03)	03-Dec-12*	28-Jun-13
HBUF.2	**HBUF Integration Plan**	03-Dec-12	01-Apr-13
0080	ActivityFT-13OR0822014 Conduct Core Team Meeting	03-Dec-12*	21-Dec-12
0060	Develop Final Method and Approach Document	03-Dec-12*	19-Jan-13
0070	M4FT-13OR0822019 Finalize Method and Approach Document		19-Jan-13
0050	ActivityFT-13OR0822012 Conduct Workshop on RD&D Plan with Implimentation Team	21-Jan-13*	18-Feb-13
0240	Perform Support Literature Reviews	26-Dec-12	01-Mar-13
0030	Develop Draft Modeling, Simulation and Experimental Integration RD&D Plan	17-Dec-12*	02-Mar-13
0040	M3FT-13OR0822013 Draft Modeling, Simulation and Experimental Integration RD&D Plan		02-Mar-13
0010	Develop Used Fuel Perfomance Modeling, Simulation and Experimental Integration Plan	03-Mar-13	01-Apr-13
0020	M2FT-13OR0822011 Used Fuel Performance Modeling, Simulation and Experimental Integration Plan		01-Apr-13
HBUF.3	**HBUF Finalize/Document Method & Approach**	03-Dec-12	08-May-13
220	Develop Documentation of data, material properties, modeling inputs and proposed analytical approaches	03-Dec-12*	16-Mar-13
230	M4FT-13OR08220111 Documentation of data, material properties, modeling inputs and proposed analytical approaches		16-Mar-13
200	Develop Doc of selected M&S methodology, proposed application of UQ established data and info., and selected dev. path	09-Mar-13	08-May-13



Actual Work ◆ Baseline Milestone
Remaining Work ◇ Milestone

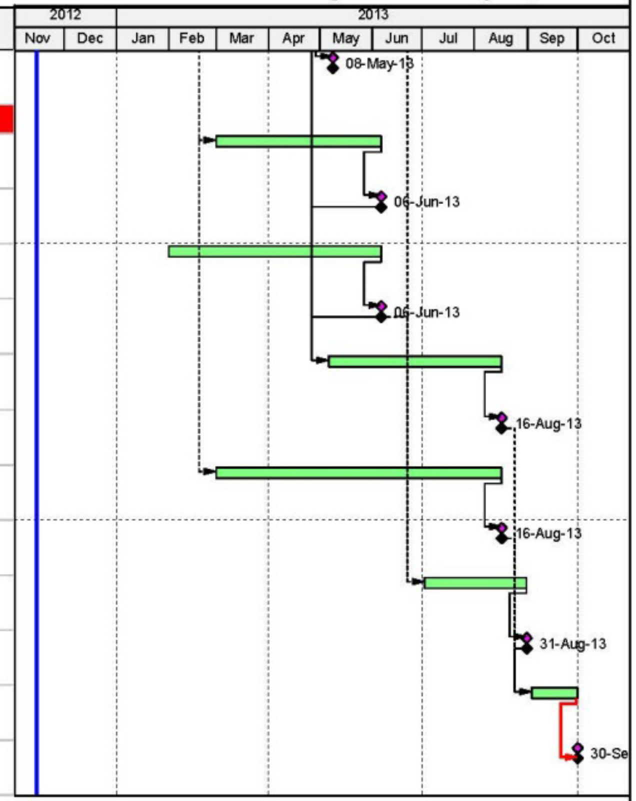
Page 1 of 2

TASK filter: All Activities

High Burnup Fuel Performance Modeling Plan Development

Updated: January 31, 2013

Activity ID	Activity Name	Start	Finish
210	M4T-13OR08220113 Doc of selected M&S meth, proposed application of UQ established data and info., and selected dev. path		08-May-13
HBUF.4	**HBUF Approach Demonstration**	**01-Feb-13**	**30-Sep-13**
0170	Develop Report documenting final UQ development, approach, and resultsulting M&S input	01-Mar-13*	06-Jun-13
0180	M4FT-13OR08220110 Report documenting final UQ development, approach, and resultsulting M&S input		06-Jun-13
0190	Develop Report documenting experimental activities	01-Feb-13*	06-Jun-13
0200	M4FT-13OR08220112 Report documenting experimental activities		06-Jun-13
0130	Develop Report documenting final data, material property, and M&S input information	06-May-13*	16-Aug-13
0140	M4FT-13OR0822017 Report documenting final data, material property, and M&S input information		16-Aug-13
0150	Develop Report documenting final M&S methodology, basis, approach, and results	01-Mar-13*	16-Aug-13
0160	M4FT-13OR0822018 Report documenting final M&S methodology, basis, approach, and results		16-Aug-13
0110	Develop Draft Demonstration of Approach on Used Fuel Performance Characterization	02-Jul-13*	31-Aug-13
0120	M3FT-13OR0822016 Draft Demonstration of Approach on Used Fuel Performance Characterization		31-Aug-13
0090	Develop Approach and Results on Used Fuel Performance Characterization	03-Sep-13*	30-Sep-13
0100	M2FT-13OR0822015 Demonstration of Approach and Results on Used Fuel Performance Characterization		30-Sep-13



Actual Work	Baseline Milestone	Page 2 of 2	TASK filter: All Activities
Remaining Work	Milestone		(c) Primavera Systems, Inc