|  |
| --- |
| Used Fuel disposition CampaignUsed Nuclear Fuel Loading and Structural Performance Under Normal Conditions of Transport – Final UQ Development, Modeling Approach and Resulting M&S Input |
|  |
|  |
|  |
|  |
| Prepared forU.S. Department of EnergyUsed Fuel Disposition CampaignJune 6, 2013FCRD-UFD-2013-000173 |

Disclaimer

This information was prepared as an account of work sponsored by an agency of the U.S. Government. Neither the U.S. Government nor any agency thereof, nor any of their employees, makes any warranty, expressed or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness, of any information, apparatus, product, or process disclosed or represents that its use would not infringe privately owned rights. References herein to any specific commercial product, process, or service by trade name, trade mark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the U.S. Government or any agency thereof. The views and opinions of the authors expressed herein do not necessarily state or reflect those of the U.S. Government or any agency thereof.

Disclaimer

This information was prepared as an account of work sponsored by an agency of the U.S. Government. Neither the U.S. Government nor any agency thereof, nor any of their employees, make any warranty, expressed or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness, of any information, apparatus, product, or process disclosed or represents that its use would not infringe privately owned rights. References herein to any specific commercial product, process, or service by trade name, trade mark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the U.S. Government or any agency thereof. The views and opinions of the authors expressed herein do not necessarily state or reflect those of the U.S. Government or any agency thereof.

Disclaimer

This information was prepared as an account of work sponsored by an agency of the U.S. Government. Neither the U.S. Government nor any agency thereof, nor any of their employees, make any warranty, expressed or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness, of any information, apparatus, product, or process disclosed or represents that its use would not infringe privately owned rights. References herein to any specific commercial product, process, or service by trade name, trade mark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the U.S. Government or any agency thereof. The views and opinions of the authors expressed herein do not necessarily state or reflect those of the U.S. Government or any agency thereof.

Disclaimer

This information was prepared as an account of work sponsored by an agency of the U.S. Government. Neither the U.S. Government nor any agency thereof, nor any of their employees, make any warranty, expressed or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness, of any information, apparatus, product, or process disclosed or represents that its use would not infringe privately owned rights. References herein to any specific commercial product, process, or service by trade name, trade mark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the U.S. Government or any agency thereof. The views and opinions of the authors expressed herein do not necessarily state or reflect those of the U.S. Government or any agency thereof.

Disclaimer

This information was prepared as an account of work sponsored by an agency of the U.S. Government. Neither the U.S. Government nor any agency thereof, nor any of their employees, make any warranty, expressed or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness, of any information, apparatus, product, or process disclosed or represents that its use would not infringe privately owned rights. References herein to any specific commercial product, process, or service by trade name, trade mark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the U.S. Government or any agency thereof. The views and opinions of the authors expressed herein do not necessarily state or reflect those of the U.S. Government or any agency thereof.

EXECUTIVE SUMMARY

This document addresses Oak Ridge National Laboratory milestone M4FT-13OR08220110 *Report documenting final UQ development, approach, and resulting M&S input.* This report provides an overview of the selected modeling and simulation (M&S) methodology documents the inputs, the failure analysis that will be conducted based on the M&S results, and the sensitivity analyses that will be performed.

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 to a geologic repository or other storage facility. 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 and/or can be generated to establish a preliminary determination of UNF structural performance under normal conditions of transport.

A recently prepared Research, Development, and Demonstration Plan (Adkins 2013b) describes a methodology, including development and use of analytical models, to evaluate loading and associated mechanical responses of UNF rods and key structural components during normal conditions of transport. Since the publication of the Research, Development, and Demonstration Plan, work has progressed in several key areas to implement this plan.

An approach has been developed to obtain transportation loading data that will be used by the modeling and simulation team to determine the effect of shock and vibration on the fuel rods and assemblies. The modeling and simulation team has been organized into three modeling teams that will model the UNF at the cask level, the assembly level, and the fuel pin level. Pathways for data communication between these three modeling teams have been developed. An initial fatigue failure criterion was presented in the *Material Properties Handbook* (Geelhood 2013). A large quantity of data was recently collected and used to evaluate the applicability of this failure criterion for UNF under normal conditions of transport. Additionally, surrogate tests from Oak Ridge National Laboratory with unirradiated cladding were compared to this failure criterion. Failure criteria for loading due to both shock and vibration have been selected, and an approach to determine failure based on these criteria and the results produced by the modeling and simulation team has been developed. Finally, various sensitivity analyses have been identified to evaluate the impact of various material property and modeling uncertainties on the resulting stress and strain predictions and failure prediction. These analyses will guide researchers to focus on areas where uncertainties have a significant impact on the performance predictions.

As mentioned in the Research, Development, and Demonstration Plan, the methodology demonstration is initially focused on structural performance evaluation of Westinghouse Electric 17×17 Optimized Fuel Assembly 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 S-2043 specification (AAR 2003).

This document will summarize the work performed to date on implementing this plan as well as finalize the inputs and outputs that are expected and the sensitivity analyses that will be performed.

ACKNOWLEDGMENTS

The authors, Harold E. Adkins and Ken Geelhood (Pacific Northwest National Laboratory), would like to thank Carl E. Beyer (Pacific Northwest National Laboratory) and Justin L. Coleman (Idaho National Laboratory) for performing the technical peer review.

The authors would also like to thank Steve Matsumoto and Susan Tackett, Pacific Northwest National Laboratory technical communications specialists, for editorial assistance.

CONTENTS

[EXECUTIVE SUMMARY iii](#_Toc358284036)

[ACKNOWLEDGMENTS v](#_Toc358284037)

[ACRONYMS ix](#_Toc358284038)

[1. Introduction 1](#_Toc358284039)

[2. Background 3](#_Toc358284040)

[3. Objectives 5](#_Toc358284041)

[4. Modeling Approach 7](#_Toc358284042)

[5. Modeling Inputs 9](#_Toc358284043)

[5.1 Material Properties 9](#_Toc358284044)

[5.2 Loading Histories 9](#_Toc358284045)

[5.3 Component Temperatures 10](#_Toc358284046)

[5.4 Fuel Post-Irradiation Conditions 10](#_Toc358284047)

[6. Failure Determination 17](#_Toc358284048)

[6.1 Interim Failure Criteria 17](#_Toc358284049)

[6.2 Failure Application Methodology 18](#_Toc358284050)

[6.3 Ongoing Experimental and Analytical Work 21](#_Toc358284051)

[7. Sensitivity Analyses 23](#_Toc358284052)

[7.1 Material Properties Sensitivity 24](#_Toc358284053)

[7.2 Initial Conditions Sensitivity 24](#_Toc358284054)

[7.3 Modeling Assumptions Sensitivity 26](#_Toc358284055)

[8. Conclusions 29](#_Toc358284056)

[9. References 31](#_Toc358284057)

FIGURES

[Figure 4.1. Overview of Modeling Approach with Inputs and Outputs 7](#_Toc358283469)

[Figure 6.1. Methodology to Analyze the Potential for Failure for the Shock and Vibrational Loads 20](#_Toc358283470)

[Figure 6.2. ORNL Fatigue Bend Test Apparatus Being Prepared for Installation in the Hot Cell 21](#_Toc358283471)

**TABLES**

[Table 5.1. Typical Conditions for Peak Fuel Rod from PWR Westinghouse 17x17 Fuel Assembly Discharged at Assembly Average Burnup of 30 GWd/MTU. 11](#_Toc357773357)

[Table 5.2. Typical Conditions for Peak Fuel Rod from PWR Westinghouse 17x17 Fuel Assembly Discharged at Assembly Average Burnup of 40 GWd/MTU. 12](#_Toc357773358)

[Table 5.3. Typical Conditions for Peak Fuel Rod from PWR Westinghouse 17x17 Fuel Assembly Discharged at Assembly Average Burnup of 50 GWd/MTU. 13](#_Toc357773359)

[Table 5.4. Typical Conditions for Peak Fuel Rod from PWR Westinghouse 17x17 Fuel Assembly Discharged at Assembly Average Burnup of 55 GWd/MTU. 14](#_Toc357773360)

[Table 7.1. Sensitivity analyses to be performed for this initiative 23](#_Toc357773361)

ACRONYMS

BWR boiling water reactor

DOE U.S. Department of Energy

GBC generic burnup cask

GWd gigawatt-days

MP material properties

M&S modeling and simulation

MTU metric tons (Tonnes) of uranium

NCS normal conditions of storage

NCT normal conditions of transport

NRC U.S. Nuclear Regulatory Commission

ORNL Oak Ridge National Laboratory

PNNL Pacific Northwest National Laboratory

PWR pressurized water reactor

RD&D research, development, and demonstration

TL transportation loading

TTCI Transportation Technology Center, Inc.

UFDC Used Fuel Disposition Campaign

UNF used nuclear fuel

Used Fuel disposition Campaign

Used Nuclear Fuel Loading and Structural Performance Under Normal Conditions of Transport – Final UQ Development, Modeling Approach, and Resulting M&S Input

# Introduction

The U.S. Department of Energy (DOE) Office of Nuclear Energy, 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. This understanding establishes part of 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.

# 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 had 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 gigawatt-days (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 publicly available for use 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 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 ductile-to-brittle-transition temperature 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 ductile-to-brittle-transition temperature, 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 single or multiple transports would have a negative impact on fuel integrity and its suitability to meet the regulations regarding retrievability after transport required by regulations. 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. This understanding 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 2013a) was developed to identify the decisions and proceedings of that meeting. 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. Following this meeting a Modeling, Simulation, and Experimental Integration RD&D Plan (Adkins 2013b) was developed. This document describes a methodology, including development and use of analytical models, to evaluate loading and associated mechanical responses of UNF rods and key structural components. Since the publication of the RD&D Plan, work has progressed in several key areas to implement this plan. On April 24–25, 2013 the modeling and simulation team met to review their progress and establish a path forward. Based on this discussion, a document was produced, describing the loading, M&S Methodology, Sensitivity, Failure, and Associated Development Strategies (Adkins 2013c).

This document will summarize the work performed to date on implementing this plan as well as finalize the inputs and outputs that are expected and the sensitivity analyses that will be performed.

# Objectives

The objective of the work as described in the RD&D Plan is to determine the mechanical loads 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. Uncertainties in these results will be quantified through sensitivity studies. 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 subject matter experts, developing and demonstrating a methodology, performing computational analyses, planning and executing experimental measurements, 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 in the RD&D Plan is the end of fiscal year 2013. The fundamental near-term objectives of this initiative are stated below, and a summary of progress to date in each area is provided:

* **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.** Most of the initial literature review has been performed to identify, assemble, and document applicable data and information and to identify information gaps. This information has been 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., have been consolidated into a *Material Properties Handbook* (Geelhood 2013), and sensitivity ranges are being selected for use in upcoming modeling and simulation (M&S) efforts. Uncertainty quantification will be limited to UNF mechanical properties and various modeling assumptions 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) are being constructed at three discrete levels; cask level, assembly level, and pin level. These models are being documented in a fashion that is conducive to future upgrades and modifications/alterations for performing alternate simulations, and they are able to accommodate new information as it becomes available. Fundamentally, the models are being 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 have been selected based on availability and pertinence to the deterministic predictions intended to be performed for this initiative.
* **Provide an initial demonstration of the developed models’ 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 to identify 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 and will 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 the following:

* Provide an analytical assessment of UNF integrity when subjected to NCT.
* Identify the type of ductility demands that would be required to achieve 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 subsequent transportation of UNF.

# Modeling Approach

The general modeling approach is shown schematically in Figure 4.1 along with the identified inputs and outputs. The modeling and simulation approach has been described in detail previously (Adkins 2013c). In general, the modeling and simulation approach consists of three levels of sub-models. As described in the Modeling, Simulation, and Experimental Integration RD&D Plan (Adkins 2013b), this methodology will utilize finite element analysis sub-modeling techniques to accurately model the complete spent nuclear fuel transport system on the railcar (cask restraint structure, cask, basket, and fuel). This sub-modeling approach will allow for more detailed finite element models of individual system components, faster analysis run times for the individual sub-models, and flexibility when updating or modifying the sub-models to incorporate better excitation data, initial material properties, or other pertinent information. The cask modeling effort interfaces with the transportation loading (TL) team by using excitations from the railcar bed. These excitation loads are 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 finally output to the material properties (MP) team for failure evaluation. All levels of sub-modeling will use input component temperatures, fuel post-irradiation conditions, and material properties from a common source, as appropriate.

Construction of model geometry, meshing, and implementation of material properties has been completed at each of the different levels of the modeling effort, and initial numerical results are being generated.



Figure 4.1. Overview of Modeling Approach with Inputs and Outputs

The key to moving forward with producing numeric results is to define all the inputs that will be used by the M&S team so that calculations can proceed. Section 5 describes the inputs that will be used for each area identified in Figure 4.1. Of equal importance is identifying how the outputs of the M&S effort will be used to determine whether fuel rod failure will occur. Adkins (2013c) identified several interim failure criteria and a counting methodology that will be used to determine whether failure is likely to occur, given the fuel rod stress and strain history provided by the M&S team. Section 6 summarizes these criteria and this methodology as well as describes the experimental and analytical work that is under way to improve these criteria for future analyses.

In the modeling of UNF under NCT, it is acknowledged that there are many sources of uncertainties in the material properties, the fuel assembly initial conditions, and the modeling assumptions. To quantify what the impact of these uncertainties is on the fuel rod stress and strain histories and ultimately on the failure prediction, a number of sensitivity studies have been previously identified (Adkins 2013c). Section 7 identifies which of these sensitivity studies will be performed under this initiative as well as the sub-modeling group that will perform the study. Sensitivity studies that have been identified but will not be performed are listed in this section and could be performed under a future initiative.

# Modeling Inputs

As previously identified, there are a number of modeling inputs that need to be finalized before the M&S work can proceed. These inputs are component material properties, loading histories of the rail car, component temperature, and fuel and assembly post-irradiation conditions. This section will identify what should be used for each of these items. In some cases, work is still under way to finalize various inputs. In these cases, interim values will be provided and a discussion on the anticipated completion for the final values will be provided.

## Material Properties

Material property correlations for irradiated, fuel, cladding, and other assembly components (e.g., grid spacers) have been provided in the *Material Properties Handbook* (Geelhood 2013). This document also includes material property correlations for the materials in the generic burnup cask (GBC)-32. Since the release of the *Material Properties Handbook*, the M&S team has requested elastic modulus and density for the neutron shield material used in the GBC-32.

Properties are given below based on values for NS-4-FR given in the NAC International Technical Data Brochure (Danner 1994).

* Elastic Modulus: 3.87 GPa
* Density: 1.68 g/cm³

The M&S team should exclusively use the material property correlations in the material properties handbook and the properties for the neutron shield material given above.

## Loading Histories

In the Loading, M&S Methodology, Sensitivity, and Failure strategy document (Adkins 2013c), an approach for obtaining shock and vibration data was developed. This data 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. The option chosen by the TL team used the extensive data archives and simulation capabilities of the TTCI in a three-step process:

1. Obtain existing data for a representative railcar.
2. Use the NUCARS® program (TTCI 2013) to simulate the representative railcar and establish a validation demonstration of the NUCARS® to accurately predict rail-related loading conditions.
3. Use the NUCARS® program to simulate a representative UNF railcar carrying a generic current-generation rail transportation cask such as the GBC-32.

The first activity on the above list has been completed, and this representative data has been delivered to the M&S team for initial use in their models. These data will initially be used to set up and debug the FEA models and perform sensitivity studies with the understanding that as the data from task 3 above becomes available, it can be fed into the model and the model can be re-run.

## Component Temperatures

An ANSYS model has been assembled to calculate the temperature of various components in a system similar to the GBC-32 as a function of decay heat within the fuel assemblies and ambient temperature outside a GBC-32-like system. Work is currently under way to provide temperatures at various locations within the cask for typical cask loading scenarios. These temperatures can be used by the M&S team in their modeling effort. Many of the material properties models are a function of temperature; therefore the results of the modeling team will likely be impacted by the temperature of the components.

The expected component temperatures will be available to the M&S team in late June 2013. In the meantime modelers should assume a constant temperature of 200°C throughout the cask.

## Fuel Post-Irradiation Conditions

In addition to providing material properties correlations for the components in the fuel assemblies and the GBC-32, the *Material Properties Handbook* (Geelhood 2013) also contains a section that provides representative conditions of fuel rods as a function of burnup and axial location. These conditions include burnup level, fast neutron fluence, corrosion layer thickness and hydrogen content. All of these are parameters are input values to various correlations described in the *Material Properties Handbook*. These tables are reproduced below and should be used as necessary to provide initial conditions for the fuel rods being modeled under this initiative.

The following tables (Table 5.1 through Table 5.4) show typical conditions at 1-ft intervals for the peak rod from assemblies discharged at 30, 40, 50, and 55 GWd/MTU assembly average burnup. These tables were developed with calculations from FRAPCON-3.4 (Geelhood et al. 2010a). No fuel cladding gap is predicted in moderate to high burnup fuel when cooled to room temperature. The contact pressure is removed, but there is no effective gap predicted beyond the sum of the fuel and cladding roughness (about 2.5 m). The conditions shown in these tables are for the peak rod in the assembly. These conditions may conservatively be used for all the rods in an assembly. The oxide layer predicted should be assumed to be in place on the rod but offer no load-carrying capability. The cladding should be thinned using the Pillings-Bedworth correlation as discussed in the *Material Properties Handbook* (Geelhood 2013).

Table 5.1. Typical Conditions for Peak Fuel Rod from PWR Westinghouse 17x17 Fuel Assembly Discharged at Assembly Average Burnup of 30 GWd/MTU



Table 5.2. Typical Conditions for Peak Fuel Rod from PWR Westinghouse 17x17 Fuel Assembly Discharged at Assembly Average Burnup of 40 GWd/MTU



Table 5.3. Typical Conditions for Peak Fuel Rod from PWR Westinghouse 17x17 Fuel Assembly Discharged at Assembly Average Burnup of 50 GWd/MTU



Table 5.4. Typical Conditions for Peak Fuel Rod from PWR Westinghouse 17x17 Fuel Assembly Discharged at Assembly Average Burnup of 55 GWd/MTU



It is acknowledged that the conditions described here are those of the fuel rod immediately upon discharge from the reactor. It is possible that the conditions of vacuum drying (up to 400°C) and extended dry cask storage (20°C to 400°C for 20 to 60 years) could change the material properties and initial conditions provided in the *Material Properties Handbook*. This is identified as a possible information gap that will be evaluated in future years to come.

One possibility includes hydride reorientation due to the vacuum drying operation. Typically the hydrides in PWR Zircaloy-4 cladding are circumferentially oriented and primarily located in a dense hydride rim along the outer edge of the cladding. If a significant number reorient to the radial direction, it can cause brittle failure to occur in the cladding by providing an easy path for a crack to propagate through the cladding thickness. When hydride reorientation is observed, brittle failure has been observed before yielding of the cladding occurs. Work is currently under way on this initiative and other initiatives to evaluate if hydride reorientation has occurred in fuel currently being stored in dry casks and, if so, what the impact on failure or ductility is. The possibility of hydride reorientation will not be examined in the modeling work under the current initiative, although the results of the modeling work for this initiative can be used to assist in assessing the impact of hydride reorientation if it is found to have occurred. As such, this will be considered an information gap until it can be evaluated further.

Another possibility includes low-temperature/long-term annealing of the irradiation damage in the cladding. Irradiation damage significantly increases the strength of Zircaloy-4 and slightly increases the elastic modulus of Zircaloy-4. If this damage is annealed out, the strength and modulus could be reduced and approach their unirradiated values. Sensitivity studies will be performed by the M&S team as described in Section 7 to address the impact of low-temperature/long-term annealing following extended dry cask storage.

# Failure Determination

The primary purpose for this initiative is to determine whether the loads that UNF is subjected to under rail-related NCT are sufficient to cause failure in the cladding and/or assembly structure. The modeling activities described in the previous Section 4 and in the Loading, M&S Methodology, Sensitivity, and Failure strategy document (Adkins 2013c) can provide computed stress and strain histories for the fuel rods based on input loading from the rail car, material properties given in Geelhood (2013), temperature distribution, and the initial conditions. Additionally, a set of interim failure criteria and a methodology to assess failure against these criteria have been established in order to determine whether these stress and strain histories are sufficient to cause cladding failure.

Various material properties that can be used as failure criteria (e.g., yield strength, uniform elongation, fracture toughness, fatigue strength) have been identified in the *Material Properties Handbook* (Geelhood 2013). The loading experienced by UNF under rail-related NCT will consist of shock and vibrational loads. A methodology has been developed to analyze the potential for failure for the shock loads separately from the vibrational loads. Also, with the vibrational loads it is anticipated that the loading will not consist of a constant-amplitude loading over the entire transportation time. Rather, it will likely consist of several different amplitude loadings occurring with different frequencies. To accommodate this, a methodology to calculate a cumulative fatigue damage fraction based on a number of difference cycles at a number of different stress or strain amplitudes must be developed. This cumulative damage fraction will be combined with the damage fraction from the shock loading and then be used to assess the potential for failure for a given stress and strain history.

## Interim Failure Criteria

Various material properties have been proposed in the *Material Properties Handbook* (Geelhood 2013) for use as failure criteria that provide a first-order estimate of failure of UNF under NCT. The potential failure modes where criteria could be established are as follows:

* **Fatigue failure**: Failure of the rods due to excessive strain cycling at low amplitude. The mechanism causing this strain cycling would be vibration normal to the axial direction of the fuel rods as they are transported by rail or road. This is the most likely failure mechanism for UNF under NCT. For this activity, the O’Donnell fatigue-design curve for irradiated Zircaloy (O’Donnell and Langer 1964) will be used to assess failure under vibrational loading.
* **Classic failure due to excessive stress or strain**: This type of failure could be caused by a large shock load that is due to a normal event such as rail car coupling but that does not cause impact on the fuel rods. Because these events are assumed to occur under NCT, this type of failure will be examined. For this activity, the uniform elongation (plastic strain at maximum load) will be used to assess failure under excessive stress or strain.

**Failure due to impact loading on the rods**: There is no anticipated impact loading for UNF under NCT. This type of loading is typically characterized by the fracture toughness of a material and is more applicable for accident and drop scenarios that are not considered in this work. The *Material Properties Handbook* (Geelhood 2013) provides a fracture toughness model that could be used if such a failure mechanism were considered. However, this information should be evaluated with respect to applicability to the accident scenario under consideration and considered an information gap until determined otherwise. For this activity, failure due to impact loading will not be considered.

## Failure Application Methodology

The cladding stress and strain history predicted by the detailed UNF pin model informed by the assembly and cask models under rail-related NCT conditions will be used to determine cladding integrity. The cladding stress and strain history will be broken down into shock events and vibrational loading.

For shock events, the maximum predicted cladding strain will be compared to predicted uniform elongation at the appropriate conditions. For vibrational loading, the strain history will be broken up into various cyclic strain magnitudes and the calculated number of occurrences. Because it is anticipated that there will be a different number of cyclic strains at different magnitudes, a damage fraction will be calculated that includes the cyclic loadings at various amplitudes. This approach known as the Rainflow-counting algorithm (Matsuishi and Endo 1968) is often used to calculate failure due to fatigue and is included in ASTM E 1049-85 (ASTM 2005).

To calculate the fatigue damage fraction, the vibrational loading history will be broken up into a number of strain amplitudes and the number of cycles at each of these amplitudes. For each strain amplitude, fatigue strength, S, will be calculated according to:

 

Where:

S = fatigue strength

E = elastic modulus

T = total strain change over cycle

For each fatigue strength calculated, the fatigue-design curve will be used to determine the allowable number of cycles at that strain amplitude and a damage fraction will be calculated by dividing the anticipated number of cycles at that strain amplitude by the number allowed at that amplitude. The cumulative damage fraction is calculated by adding the individual damage fractions for each strain amplitude to the damage fraction from each shock loading event according to:

 

Where:

Ni = number of anticipated cycles at a given strain amplitude

Nallow = number of allowable cycles at a given strain amplitude

plastic = plastic strain from a given shock loading event

UE = uniform elongation for given shock loading event

The methodology described above to analyze the potential for failure from shock loads and from the vibrational loads is shown graphically in Figure 6.1.

If the comparisons of the cladding stress and strain history to these failure criteria show that the fuel assembly performance during NCT is significantly below the established failure criteria, then this may provide adequate demonstration that the cladding will not experience gross failure. If the resulting finite element model stress/strain history indicate conditions that are close (e.g., within 80%) to the established failure criteria, then further experimental work may be required to refine the current failure curves.



Figure 6.1. Methodology to Analyze the Potential for Failure for the Shock and Vibrational Loads

## Ongoing Experimental and Analytical Work

As discussed elsewhere (Adkins 2013c), the proposed fatigue failure criterion (O’Donnell and Langer 1964) is not based on data completely prototypic to UNF under NCT. Although a significant body of other fatigue data provided some additional insight into the influence of various parameters on fatigue failure in Zircaloy (Wisner et al. 1994, Soniak et al. 1994, Mehan and Wiesinger 1961, Lin 1998, Pandarinathan and Vasudevan 1980), there was no data that was fully representative of the conditions of UNF.

Oak Ridge National Laboratory (ORNL) is beginning fatigue tests on irradiated cladding segments containing fuel. The fatigue bend test apparatus seen in Figure 6.2 is currently being installed in a hot cell at ORNL. These data should be available in the summer/fall of 2013 and will be the most representative of UNF under NCT data collected to date. When these data are available, they will be compared to the other fatigue data that have been collected, and if they deviate significantly from the other data, they can be used to adjust the current fatigue-design curve. Zircaloy fatigue strength was relatively insensitive to various other conditions tested in the data mentioned above. It is noted that the data obtained to date are not fully representative of the stress state and specimen condition of high burnup cladding during NCT.



Figure 6.2. ORNL Fatigue Bend Test Apparatus Being Prepared for Installation in the Hot Cell

The modeling performed under this initiative will be used to provide guidance for future testing on spent fuel and surrogate material at ORNL. For example, if it is determined that the anticipated strain amplitudes and number of cycles is close (e.g., within 80%) to the current fatigue-design curve, then more tests on spent fuel may be necessary to refine this curve. Otherwise, vibrational tests may be useful to validate the modeling results and the assumptions regarding damping, or testing may be performed on cladding from a different burnup range or different cladding type.

Additional work that is currently under way includes developing fundamental models to predict the mechanical behavior of Zircaloy cladding with hydrides in various distributions and orientations. A second model being developed will predict hydride nucleation, precipitation, and reorientation to predict hydride reorientation that could occur during vacuum drying operations.

The mechanics model is being tuned to predict the ring crush data taken at Argonne National Laboratory based on micrographs that provide hydride orientation (Billone et al. 2013). Once tuning is complete, the model will be validated against other data, including mechanical test data compiled by Pacific Northwest National Laboratory (PNNL) and upcoming static bend test data from ORNL.

The hydride nucleation, precipitation, and reorientation model could be used to set the initial condition of irradiated Zircaloy cladding upon discharge from the reactor. However, these conditions are fairly well known from post irradiation examination of fuel discharged from reactors. It would be more beneficial for this model to be able to predict, based on a given hydride orientation and distribution, what the hydride orientation and distribution is following various temperature cycles under various stresses as would be expected during vacuum drying operations. This model could be validated against the Argonne National Laboratory reorientation data (Billone et al. 2013) or other data available in the open literature. Once validated, this model could be used to provide initial conditions for the mechanics model.

If a significant portion of the hydrides in the cladding reorient to the radial direction, the uniform elongation and fracture toughness could be significantly lowered. This would have an impact on the failure due to shock loading.

# Sensitivity Analyses

As mentioned in the RD&D Plan (Adkins 2013b), uncertainty qualification will be limited to sensitivity analyses on UNF mechanical properties, initial conditions, and modeling assumptions during this initiative because of the volume of work scheduled to be conducted within the defined performance period. These sensitivity evaluations will be performed by varying parameters related to mechanical properties, initial conditions, and modeling assumption to determine their impact on the outputs of interest.

The M&S team and the MP team have identified a list of mechanical properties, initial conditions, and modeling assumptions that will be included in these sensitivity analyses. Other uncertainties have been identified but have been determined to be outside the scope of the current initiative. These are discussed in this section but will not be performed. Table 7.1 gives a summary of the sensitivity analyses and identifies which sub-modeling group will perform each analysis. The following sections discuss each of these analyses in more detail.

Table 7.1. Sensitivity Analyses to Be Performed for This Initiative

|  |  |
| --- | --- |
| **Analysis** | **Responsible Party** |
|  | **Fuel Rod****(INL)** | **Assembly****(PNNL)** | **Cask****(SNL)** | **Failure Team** |
| Cladding elastic modulus | X |  |  |  |
| Cladding yield stress |  |  |  | X |
| Spacer grid stiffness | X | X |  |  |
| Spacer grid location | X | X |  |  |
| Fuel assembly basket location |  | X | X |  |
| Fuel rod location in assembly | X | X |  |  |
| Temperature distribution |  | X | X |  |
| In-reactor fretting wear | X |  |  |  |
| Gaps between assembly and cask |  |  | X |  |
| Influence of control components |  | X | X |  |
| Fuel rod damping | X |  |  |  |
| Pellet-to-cladding bonding | X |  |  |  |
| Pin pressure influence | X |  |  |  |
| Total cyclic loading |  |  | X |  |

Sensitivity analyses will be performed using the preliminary rail loading data discussed in Section 5.2. This will allow the M&S team to perform these studies in parallel to the effort by the TL team developing the final rail loading data. Additionally, the sensitivity analyses will be performed for shorter shock events as appropriate and for a limited set of conditions.

The following sections discuss in more detail the sensitivity analyses that will be performed.

## Material Properties Sensitivity

These sensitivity evaluations will employ UNF MP and corresponding uncertainties that have had extensive uncertainty qualification performed on them to support Pacific Northwest National Laboratory/NRC fuel performance codes, FRAPCON-3 (Geelhood et al. 2010a), and FRAPTRAN (Geelhood et al. 2010b), respectively. The *Material Properties Handbook* (Geelhood 2013) summarizes the nominal model and associated uncertainty for each of the MP. It also includes discussions on the impact of long-term storage and vacuum drying on various MP. Using this information, sensitivity analyses may be readily performed on the MP.

The M&S team identified the following as important material properties to analyze the sensitivity of these items on the outputs of interest.

* **Cladding elastic modulus**: Examine the impact of uncertainty in modeling and the potential impact of annealing during storage. This sensitivity will be evaluated at the fuel rod level by examining the impact of using an upper bound, lower bound, and best-estimate elastic modulus on the resulting stress and strain history.
* **Cladding yield stress**: Examine the impact of uncertainty in yield strength as well as the potential for annealing during storage to reduce the yield strength. Changes in the yield strength may affect whether the model results show only elastic deformation or elastic plus plastic deformation. The failure team will examine the stress and strain history produced by the M&S team to determine whether a nominal yield stress or lower bound yield stress has been exceeded.
* **Spacer grid stiffness**: Examine the sensitivity of model outputs to spacer grid stiffness. This sensitivity will be evaluated at the assembly level and the fuel rod level by examining the impact of this parameter on the behavior of the assembly and the fuel rods.

## Initial Conditions Sensitivity

As discussed in Section 5.4, the *Material Properties Handbook* (Geelhood 2013) summarizes the nominal initial conditions that are anticipated for UNF before NCT. It is acknowledged that there will be variation in these conditions due to local variations within a rod, as well as differences due to different fuel assembly designs or in-reactor operation, vacuum drying, and pool storage differences. Sensitivity analyses will be performed on the initial conditions that have been identified as possibly impacting the results of the models.

The M&S team identified the following as important initial conditions to analyze the sensitivity of these items on the outputs of interest.

* **Spacer grid location**: The location of the grids is dependent on year of manufacture, and the spacer grid location provides a different distance for deflection. This sensitivity will be evaluated at the assembly level and the fuel rod level by examining the impact of spacer grid location on the behavior of the assembly and the fuel rods, including the natural frequency of these items.
* **Fuel assembly loading dependence on basket location**: This sensitivity will be evaluated at the cask level and the assembly level by examining the impact of basket location on the behavior of the assembly. A limited number of basket locations will be examined at the assembly level. The cask model will be used to identify the worst basket location.
* **Fuel rod loading dependence on location within assembly**: This sensitivity will be evaluated at the assembly level and the fuel rod level by examining the impact of rod location within the assembly on the behavior of the rod. A limited number of assembly locations will be examined at the fuel rod level. The assembly model will be used to identify the worst location within the assembly.
* **Temperature and temperature distribution within package**: These temperatures are expected to be dependent on an average decay heat generation rate, the assembly location within the basket, and the ambient temperature. Lower ambient temperatures will also influence material properties, increasing rail bed stiffness and thereby amplifying the loading imparted to the fuel assemblies. This sensitivity will be evaluated at the cask level and the assembly level by examining the impact of these temperatures on the behavior of the assembly.
* **Fretting from in-reactor**: The *Material Properties Handbook* provides a bounding value of 10% cladding thinning due to fretting. This sensitivity will be evaluated at the fuel rod level by examining the impact of thinner cladding in the vicinity of spacer grids on the resulting stress and strain history.

The M&S team identified the following as potentially important initial conditions to analyze the sensitivity of these items on the outputs of interest. However, given the short period of performance and available budget, these are beyond the scope of this study.

* **Rod bowing**: Examine the impact of previously deformed rods. This impact needs to be looked at eventually but for now is daunting enough to be considered to be beyond the scope of this study given the short period of performance and available budget.
* **Bambooing**: Minimal in modern fuel, bambooing has been observed in legacy fuel. This impact needs to be looked at eventually but for now is considered to be beyond the scope of this study given the short period of performance and available budget.

## Modeling Assumptions Sensitivity

The M&S team has identified a number of modeling assumptions that will need to be made to model the response of the fuel rods to NCT. Many of these assumptions rely on conditions or properties that have not been previously measured or are very difficult to measure. Therefore to determine whether resources should be spent in the future to better qualify these assumptions, sensitivity analyses will be performed using two bounding modeling assumption that are known to bound the actual condition of the UNF. By examining the impact of these bounding modeling assumptions, a determination may be made as to whether these assumptions have a significant impact on the predicted response of the fuel. Those modeling assumptions that are found to significantly impact the response of the fuel will be identified to guide future test programs.

The M&S team identified the following as important modeling assumptions to analyze the sensitivity of these items on the outputs of interest.

* **Influence of gaps between the package and its payload**: Examine how these gaps will influence assembly and rod loading and how these loads are imparted to the payload. This sensitivity will be evaluated at the cask level by examining the impact of gaps—between the cask and the canister, the canister and the basket, and the basket and the fuel—on the shock loads imparted to the assembly.
* **Influence of control components**: Examine the influence of including these components on the response of the rods. These items will influence the top head mass and guide tube stiffness. The guide tube stiffness will influence the dynamic response of the assembly as well as loading distribution and delivery to the rods. This sensitivity will be evaluated at the assembly level by examining the impact of the presence and absence of control rods and the associated components on the behavior of the assembly, including the natural frequency.
* **Fuel rod damping – equivalent stiffness**: Examine the influence on upward (out to the conveyance) and downward (down to the rod) model responses through sub-modeling. This sensitivity will be evaluated at the fuel rod level through the basic approach to the fuel rod evaluation described in the Loading, M&S Methodology, Sensitivity, and Failure strategy document (Adkins 2013c) where bounding properties are used in the time history analysis.
* **Pellet-to-cladding bonding**: Examine the influence of assuming different bonding or non-bonded scenarios. This could influence the static/dynamic response and affect fuel rod survivability. This sensitivity will be evaluated at the fuel rod level through the basic approach to the fuel rod evaluation described in the Loading, M&S Methodology, Sensitivity, and Failure strategy document (Adkins 2013c) where bounding properties are used in the time history analysis. Additionally the fuel rod level model will evaluate the stress concentrations that occur between the pellet-clad-pellet interface during NCT. Because the assembly-level model will use beam elements (with stiffness and damping properties provided by the fuel-level analysis), it will not capture the stress concentrations that will occur at the pellet-clad-pellet interface. The detailed fuel-rod-level model will investigate how much this interaction “magnifies” the stress in those areas.
* **Pin pressure influence**: Examine the response of the fuel rod with higher pressure and lower pressure. This sensitivity will be evaluated at the fuel rod level by examining the impact of a high and low rod internal pressure on the resulting stress and strain history.
* **Total cyclic loading**: Examine the influence of various modeling assumptions for accounting for the complete vibration and shock loading history accumulated during a specified realistic hypothetical transport campaign. Given current computing power, it is not possible to explicitly model the entire time frame of the transport campaign. Therefore some assumptions will have to be made about how to extend the results of the model over some smaller period of time to mimic the entire time frame. This sensitivity will be evaluated at the cask level, and guidance will be provided regarding a reasonable time frame to model and how to extend the results from this time frame to the entire time frame.

# Conclusions

Work on this initiative is progressing according to schedule. The M&S team has elected to divide the modeling into three levels; the cask level, the assembly level, and the fuel rod level. Communication pathways that go both ways (smaller to larger and larger to smaller) have been established, and the models have been created. Final input values for material properties and initial conditions have been established. Initial input values for rail loading history have been prepared and will be used in the sensitivity studies. Input values for component temperatures are being prepared. In the meantime modelers should assume constant temperature of 200°C throughout the cask and payload.

An interim failure criterion to be used to assess the potential for cladding failure due to shock and vibrational loading has been established. A methodology for assessing the potential for cladding failure based on the data provided by the M&S team has been developed and can be implemented once results are available from the modeling efforts. The MP team is aware of current test programs that are ongoing and will incorporate new data as they become available into the material properties models and failure criteria as justified by the data. Additionally modeling activities are under way to create validated models to provide improved failure criteria.

Sensitivity analyses and a plan for conducting these analyses have been identified to be performed to investigate the impact of material property uncertainties, uncertainties in the initial condition of the UNF, and modeling uncertainties on the outputs of interest.

# References

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).

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

Adkins HA. 2013b. *Used Nuclear Fuel Loading and Structural Performance Under Normal Conditions of Transport – Modeling, Simulation and Experimental Integration RD&D Plan*. FCRD-TIO-2013-000135, U.S. Department of Energy, Washington, D.C.

Adkins HA. 2013c. *Used Nuclear Fuel Loading and Structural Performance Under Normal Conditions of Transport – Loading, M&S Methodology, Sensitivity, Failure, and Associated Development Strategies*. FCRD-TIO-2013-000155, U.S. Department of Energy, Washington, D.C.

ASTM E 1049-85. 2005R. *Standard Practices for Cycle Counting in Fatigue Analysis*. ASTM International, West Conshohocken, Pennsylvania.

Billone MC, TA Burtseva, and RE Einziger. 2013. “Ductile-to-brittle transition temperature for high-burnup cladding alloys exposed to simulated drying-storage conditions.” *Journal of Nuclear Materials* (433):431–448.

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.

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.

Danner, TA. 1994. *NAC International GESC Shield Materials Technical Data and Material Safety Data Sheet: NS-4-FR*. NAC International, Norcross, Georgia.

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. Accessed June 4, 2013 at <http://energy.gov/downloads/strategy-management-and-disposal-used-nuclear-fuel-and-high-level-radioactive-waste>.

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.

Geelhood KJ. 2013. *Used Nuclear Fuel Loading and Structural Performance Under Normal Conditions of Transport – Supporting Material Properties and Modeling Inputs*. FCRD-UFD-2013-000123, U.S. Department of Energy, Washington, D.C.

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.

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.

Matsuishi M and T Endo. 1968. “Fatigue of metals subjected to varying stress.” *Japan Society of Mechanical Engineering*.

Mehan RL and FW Wiesinger. 1961. *Mechanical Properties of Zircaloy-2*. KAPL-2110, Knolls Atomic Power Laboratory, Schenectady, New York.

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

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, 2nd ed. Organisation for Economic Co-operation and Development, Nuclear Energy Agency. Accessed June 4, 2013 at <http://www.oecd-nea.org/nsd/reports/2012/nea7072-fuel-safety-criteria.pdf>.

Pandarinathan PR and P Vasudevan. 1980. “Low-Cycle Fatigue Studies on Nuclear Reactor Zircaloy-2 Fuel Tubes at Room Temperature, 300 and 350°C.” *Journal of Nuclear Materials* (91):47-58.

Soniak A, S Lansiart, J Royer, J-P Mardon, and N Maeckel. 1994. “Irradiation Effect on Fatigue Behavior of Zircaloy-4 Cladding Tubes.” *Zirconium in the Nuclear Industry: Tenth International Symposium*, STP 1245, pp. 549-558. American Society for Testing and Materials, now ASTM International, West Conshohocken, Pennsylvania.

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

Wisner SB, MB Reynolds, and RB Adamson. 1994. “Fatigue Behavior of Irradiated and Unirradiated Zircaloy and Zirconium.” In *Zirconium in the Nuclear Industry: Tenth International Symposium*, STP 1245, pp. 499-520. American Society for Testing and Materials, now ASTM International, West Conshohocken, Pennsylvania.