ELECTRIC POWER RESEARCH INSTITUTE

Phenomena Identification and Ranking Table (PIRT) Exercise for Used Fuel Cladding Performance

2020 TECHNICAL REPORT

Phenomena Identification and Ranking Table (PIRT) Exercise for Used Fuel Cladding Performance

All or a portion of the requirements of the EPRI Nuclear Quality Assurance Program apply to this product.

YES NO

EPRI Project Manager
K. Waldrop

ELECTRIC POWER RESEARCH INSTITUTE

3420 Hillview Avenue
Palo Alto, CA 94304-1338
USA

PO Box 10412
Palo Alto, CA 94303-0813
USA

800.313.3774
650.855.2121

askepri@epri.com

www.epri.com

3002018439
Final Report, June 2020

DISCLAIMER OF WARRANTIES AND LIMITATION OF LIABILITIES

THE FOLLOWING ORGANIZATION, UNDER CONTRACT TO EPRI, PREPARED THIS REPORT:

MPR Associates, Inc.

THE TECHNICAL CONTENTS OF THIS PRODUCT WERE **NOT** PREPARED IN ACCORDANCE WITH THE EPRI QUALITY PROGRAM MANUAL THAT FULFILLS THE REQUIREMENTS OF 10 CFR 50, APPENDIX B. THIS PRODUCT IS **NOT** SUBJECT TO THE REQUIREMENTS OF 10 CFR PART 21.

NOTE

For further information about EPRI, call the EPRI Customer Assistance Center at 800.313.3774 or e-mail askepri@epri.com.

Acknowledgments

The following organization, under contract to the Electric Power Research Institute (EPRI), prepared this report:

MPR Associates, Inc.
320 King Street
Alexandria, VA 22314

Principal Investigators
V. Angelici Avincola
C. Dame
S. Kauffman

EPRI acknowledges the efforts and support of the following organizations and individuals that assisted and participated in the PIRT process. The participation of individuals in the PIRT process was funded by these organizations, permitting the concerted effort of a team of professionals supporting this work.

Organizations:

United States Department of Energy
U.S. Nuclear Regulatory Commission
Argonne National Laboratory
Oak Ridge National Laboratory
Pacific Northwest National Laboratory
Tennessee Valley Authority
Holtec International
NAC International
Orano TN

This publication is a corporate document that should be cited in the literature in the following manner:

Phenomena Identification and Ranking Table (PIRT) Exercise for Used Fuel Cladding Performance
EPRI, Palo Alto, CA: 2020.
3002018439.

PIRT Panel Members:

Michael Billone	Argonne National Laboratory
Brady Hanson	Pacific Northwest National Laboratory
Albert Machiels	John Kessler and Associates
Rose Montgomery	Oak Ridge National Laboratory
Joseph Rashid	Structural Integrity Associates
Ricardo Torres	U.S. Nuclear Regulatory Commission

Additional individuals who shared their key experience in the dry storage industry:

George Carver	NAC International
Raheel Haroon	Orano TN
Zita Martin	Tennessee Valley Authority
Debu Majumdar	Holtec International

This report describes research sponsored by EPRI.

Abstract

A panel of experts developed a Phenomenon Identification and Ranking Table (PIRT) to assess the significance and state of knowledge of parameters that may affect the integrity of used nuclear fuel during dry storage and transport. Regulatory guidance is intended to provide reasonable assurance that the configuration of the fuel is maintained. A key acceptance criterion is avoiding "gross rupture" that could allow fuel material to relocate from fuel rods. Criteria given in the guidance stem from fuel and cladding knowledge when it was formulated in the 2000 to 2003 timeframe. Since then, research and experience have improved the understanding of the topic, but assessment is complex because phenomena are inter-related and depend on both prior and current conditions. In general, thermal analysis is performed to conservatively calculate fuel temperatures. Many temperature-dependent physical phenomena may occur that could affect the likelihood of gross rupture, and accurate estimates of fuel temperatures are necessary to assess the likelihood of their occurrence. Significance of the phenomena on both an individual and synergistic basis must be understood. The PIRT process was applied to determine the conditions that could lead to gross rupture of fuel in dry storage and transportation casks during five scenarios: short-term loading activities, dry storage, long-term dry storage, off-normal transients, and normal conditions of transport. As a result of this process, the panel concluded that hydride reorientation and mechanical overload are potentially significant and some panelists identified fuel oxidation as potentially significant. The panel did provide recommendations for further work.

No regulatory positions are taken in this document by the PIRT panel members.

Keywords
Cladding integrity
Dry storage
Fuel oxidation
Gross rupture
Hydride reorientation
PIRT
Spent fuel

ELECTRIC POWER RESEARCH INSTITUTE

Deliverable Number: 3002018439

Product Type: Technical Report

Product Title: Phenomena Identification and Ranking Table (PIRT) Exercise for Used Fuel Cladding Performance

---

PRIMARY AUDIENCE: Regulators and dry storage system designers

SECONDARY AUDIENCE: Dry storage system users

KEY RESEARCH QUESTION

Regulatory guidance for used fuel during dry storage and transport is intended to provide reasonable assurance that the configuration of the fuel is maintained. A key acceptance criterion is avoiding "gross rupture" that could allow fuel material to relocate from fuel rods. Criteria given in the current guidance stem from fuel and cladding knowledge when it was formulated in the 2000 to 2003 timeframe. Considering the current state of knowledge, what phenomena are the most significant to the occurrence of gross rupture of spent fuel cladding during dry storage and transportation?

RESEARCH OVERVIEW

A panel of recognized fuel and cladding material experts was assembled to evaluate the current state of knowledge and to consider changes that may enable establishing improved criteria for preventing gross rupture in spent fuel storage and transportation. This report documents the deliberations by the expert panel using the phenomena identification and ranking (PIRT) process to determine the conditions that could lead to gross rupture of fuel in dry storage casks. Each of the phenomena was evaluated for its significance in five scenarios: short-term operations (e.g., cask loading and drying), storage up to 60 years, long-term storage (60 to 100 years), off-normal events involving short-duration temperature excursions, and normal conditions of transport.

KEY FINDINGS

- The onset of fuel failure is not abrupt, but rather a continuum – no cliff-edge effect is associated with the current NRC recommended 400 °C limit (ISG-11 Rev. 3). Therefore, exceeding 400 °C does not mean that multiple fuel rods will fail simultaneously.
- The evaluation performed by the panel did not quantify the extent of ruptures. Although probability of gross rupture of several rods in a single cask is lower than that of a single rod, the probabilities are not independent, as rods in a single fuel assembly have been exposed to similar conditions.
- Because experiments have focused on the current 400 °C limit, there is a gap in knowledge to go beyond the limit, especially about the kinetics of annealing and cold-work recovery.
- At temperatures beyond 400 °C, there is interaction among phenomena (synergistic, competitive, and aggravating effects) that should be further explored.
- The extent or consequences of rupture may be minor, even where a phenomenon may have a significant effect on possibility of gross rupture.

- The state of knowledge of most parameters is good, although applicability at temperatures higher than 400 °C has lower confidence due to limited testing above 400 °C.
- The panel identified two phenomena that have medium significance (hydride reorientation cladding embrittlement and mechanical overload). One phenomenon was identified by some panelists as having high significance (UO_2 oxidation in fuel with cladding breaches), yet other panelists considered UO_2 oxidation as low significance.

WHY THIS MATTERS

Identifying which phenomena are significant, and those that are not, and the state of knowledge of each assists analysts and regulators in focusing on the conditions that are most important to maintaining the integrity of fuel in dry casks during storage and transportation.

HOW TO APPLY RESULTS

The panel identified several opportunities for proposed changes to current guidance.

- Re-assess the limitations on thermal cycling during short-term loading operations (65 °C, 10 cycles).
- Investigate implementation of a temperature criterion (e.g., average temperature for a percentage of the cladding) as an alternative to the peak cladding temperature limit (400 °C) currently in the regulatory acceptance criteria.
- Extend use of hoop stress analysis from low burnup to high burnup fuel as a secondary justification when approaching the 400 °C limit.
- Implement a graded approach to define gross rupture. A possible approach would be defining different limits for different cladding materials based on the variation of effects each phenomenon has for different materials.

LEARNING AND ENGAGEMENT OPPORTUNITIES

- Discuss the definition and significance of gross rupture. This should be done during a meeting with subject matter experts in criticality, fuels/cladding, and shielding technical disciplines.
- Hold a synthesis PIRT of the three recently conducted PIRTs on Used Fuel Cladding Performance (this one), Thermal Modeling, and Decay Heat to discuss the cumulative impact of overlapping margins/uncertainties. Assess the effect of overlapping conservatisms, margins, and uncertainties from the Thermal Modeling and Decay Heat PIRTs on the Used Fuel Cladding Performance PIRT.

EPRI CONTACT: Keith Waldrop, Principal Technical Leader, kwaldrop@epri.com

PROGRAMS: Nuclear Power, P41 and Used Fuel and High-Level Waste, P41.03.01

IMPLEMENTATION CATEGORY: Reference: Technical Basis

Electric Power Research Institute
3420 Hillview Avenue, Palo Alto, California 94304-1338 • PO Box 10412, Palo Alto, California 94303-0813 USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com

Acronyms

ANL	Argonne National Laboratory
ASTM	American Society for Testing and Materials
CANDU	Canada Deuterium Uranium
CFR	Code of Federal Regulations
CIRFT	Cyclic integrated reversible-bending fatigue tester
CRIEPI	Central Research Institute of Electric Power Industry
DCCG	Diffusion-controlled cavity growth
DHC	Delayed hydride cracking
DtB	Ductile to brittle
EDF	Électricité de France
EOL	End of Life
EPRI	Electric Power Research Institute
HAC	Hypothetical accident conditions
HBU	High Burnup
HRO	Hydride reorientation
IAEA	International Atomic Energy Agency
ID	Inner diameter
ISG	Interim Staff Guidance
LT	Long Term (storage)
LWR	Light Water Reactor
MOX	Mixed oxide fuel
NCT	Normal conditions of transport
NRC	U.S. Nuclear Regulatory Commission
OD	Outer diameter
ON	Off-normal
ORNL	Oak Ridge National Laboratory

PCI	Pellet-cladding interaction
PCT	Peak cladding temperature
PIRT	Phenomena Identification and Ranking Table
PNNL	Pacific Northwest National Laboratory
PWR	Pressurized Water Reactor
RH	Relative humidity
SCC	Stress corrosion cracking
SME	Subject matter experts
SNF	Spent nuclear fuel
SNL	Sandia National Laboratories
ST	Short-term operation
Zr-2	Zircaloy 2
Zr-4	Zircaloy 4

Table of Contents

List of Figures

List of Tables

Section 1: Introduction

This report documents the results of a Phenomena Identification and Ranking Table (PIRT) exercise for cladding performance during dry storage and under normal conditions of transport (NCT) of spent fuel. The exercise was conducted at the Electric Power Research Institute (EPRI) office in Washington, D.C. (USA) in October 2019, facilitated by EPRI and MPR Associates, and with follow-up teleconferences. The objective of the exercise was to review the current state of knowledge regarding spent fuel cladding integrity and recommend possible revisions to the current regulatory guidance for preserving spent nuclear fuel integrity during dry storage and transportation operations. In parallel, EPRI also conducted a PIRT process for spent nuclear fuel decay heat level [31] and for thermal analyses of dry storage and transportation systems [30].

A PIRT exercise is a structured and facilitated expert elicitation process. Its objectives are to:

- Identify phenomena associated with the intended application,
- Rank the significance and current state of knowledge, and
- Characterize the potential effects of each phenomenon.

The PIRT discussed in this document focused on phenomena that could occur to used light water reactor oxide fuel (UO_2) and cladding (zirconium alloys) during cask/canister loading, transfer, dry storage, and transportation operations. The PIRT considered the fuel rod (fuel and cladding) and not the fuel assembly structure. The expert panel was comprised of six internationally recognized nuclear fuel experts from the following organizations: Oak Ridge National Laboratory (ORNL), Argonne National Laboratory (ANL), Pacific Northwest National Laboratory (PNNL), Structural Integrity Associates (SIA), and the U.S. Nuclear Regulatory Commission (NRC). An independent consultant, formerly EPRI employed, was also included in the panel.

1.1 Background

Regulators have established requirements pertaining to used nuclear fuel integrity in dry storage and transportation of used nuclear fuel. At the highest level are applicable rules and regulations, which are sometimes supplemented by guidance for ways to meet the regulations. Regulatory requirements and guidance vary somewhat from country to country. For purposes of this PIRT, NRC regulatory requirements were used as a basis. In the U.S., applicable NRC regulations are contained in Title 10 of the Code of Federal Regulations (10 CFR) Parts 71 and 72 for transportation and dry storage, respectively. Additionally, NRC Interim Staff Guidance (e.g., ISG-11 Rev. 3 [79]) identifies criteria to provide reasonable assurance for the integrity of fuel cladding. More detail on regulations and guidance is provided in the next chapter.

Vendors and licensees conduct thermal analyses of dry storage systems to ensure that the limits provided by the regulatory guidance are not exceeded. Currently, licensing basis design-specific thermal models typically use conservative assumptions and substantial engineering margins to offset uncertainties in the data and analysis methodology. The combined application of conservative assumptions and large uncertainty margins may lead to over prediction of storage system and fuel system temperatures. Since most fuel degradation mechanisms are related to temperature, over prediction of the temperature may lead to longer than necessary fuel cooling times in spent fuel pools, reduced capacity of dry storage systems to store high-burnup fuel, and operational activities that limit ability to reduce radiological dose to workers. Moreover, bounding or over conservative approaches could lead models to over predict canister external wall temperatures, resulting in potentially non-conservative estimates for chloride-induced stress corrosion cracking susceptibility of the canister shell.

The criteria issued in NRC guidance stem from fuel and cladding knowledge at the time the guidance was formulated in 2003. Since then, research and experience have improved the state of knowledge on the topic. Identifying, quantifying, and ranking the safety significance of phenomena that occur and affect the fuel cladding system, including the latest data might provide insights that could improve thermal models. Considering updated data in the regulatory guidance could increase operational flexibility, facilitate earlier transfer from wet to dry storage, reduce the number of dry storage systems needing transportation, and further contribute to risk-informed dry storage and aging management activities.

1.2 Regulations and Guidance

1.2.1 Regulations

NRC regulatory requirements for dry storage of spent nuclear fuel, including packaging and transportation of radioactive materials are listed below.

10 CFR Part 71 "Packaging and Transportation of Radioactive Material" includes fuel-specific and package-specific transportation system requirements for the following conditions:

- Normal conditions of transport (NCT): the regulations provide ranges of conditions that should be accounted for, such as temperature, external pressure, water spray, free drop, corner drop, compression, vibration, and tests specified in 10 CFR Part 71.71.
- Hypothetical accident conditions (HAC): the regulations provide test conditions and limits for the hypothetical accident conditions, but are outside the scope of this PIRT evaluation.

NRC regulations found in 10 CFR 71 are largely compatible with the International Atomic Energy Agency (IAEA) "Regulations for the Safe Transport of Radioactive Material" [41] [3].

10 CFR Part 72 "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-level Radioactive Waste, and Reactor-Related Greater Than Class C Waste" provides requirements for independent storage of spent fuel. The regulation includes controls for fuel loading, storage, and unloading, in order to provide reasonable assurance that cooling and subcriticality are maintained. The regulation introduces the concept of gross rupture of spent fuel cladding, including that:

> *"…The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. This may be accomplished by canning of consolidated fuel rods or unconsolidated assemblies or other means as appropriate."*[1]

Additionally, 10 CFR Part 72.122(l) requires that SNF be designed to allow ready retrieval of spent fuel without risk to the public for further processing or disposal.

1.2.2 Guidance

ISG-1, Revision 2 "Classifying the Condition of Spent Nuclear Fuel for Interim Storage and Transportation Based on Function" [81] provides guidance for categorizing the spent nuclear fuel (SNF) (both rods and assembly) as either damaged or not, based upon their ability to perform their designated functions without requiring the fuel to be handled in a non-standard manner. This includes considering whether the material properties, and possibly the configuration, of the SNF assemblies can be altered during transportation or dry storage. Fuel integrity categories established by the NRC are summarized in Table 1-1.

It should be noted that, depending on design-specific limits, a limited amount of damaged fuel can be loaded into dry casks, stored, and transported by enclosing it in a damaged fuel can that limits redistribution of fissile material within the cask.

[1] 10CFR72.122(h)(1)

Table 1-1
Fuel Integrity Categories [81]

Name	Definition
Intact SNF	Any fuel that can fulfill all fuel-specific and system-related functions, and that is not breached. All intact SNF is undamaged.
Undamaged SNF	SNF that can meet all fuel-specific and system-related functions. Undamaged fuel may be breached (e.g., can have rod defects). Not all undamaged fuel is intact.
Damaged SNF	Any fuel rod or fuel assembly that cannot fulfill its fuel-specific or system-related functions
Breached spent fuel rod	Spent fuel rod with cladding defects that permit release of gas or fuel particulate. A breach may be a pinhole or hairline crack, or a gross breach.
Grossly breached spent fuel rod	Spent fuel rod with a breach larger than a pinhole leak or a hairline crack (a subset of breached rods).

ISG-2, Revision 2 "Fuel Retrievability in Spent Fuel Storage Applications" [84] provides guidance in meeting the retrievability requirement defined in 10 CFR 72.122(l), and defines the ability to safely remove spent fuel from storage for further processing or disposal. The licensee demonstrates ready retrieval when there is the ability to perform one or more of the following actions: 1) remove individual or canned spent fuel assemblies from wet or dry storage, 2) remove a canister loaded with spent fuel assemblies from a storage cask/overpack, and 3) remove a cask loaded with spent fuel assemblies from the storage location.

Per ISG-2, retrievability is applicable only during normal and off-normal conditions (i.e., it does not apply to accident conditions).

ISG-11, Revision 3 "Cladding Considerations for the Transportation and Storage of Spent Fuel" [79] introduces the criteria that the licensee should meet to reasonably assure integrity of the cladding material. The criteria are summarized below:

- For normal conditions of storage and short-term operations, peak cladding temperature (PCT) for low and high burnup should not exceed 400°C (752°F).
- For low burnup fuel, a higher short-term temperature limit may be used if the cladding hoop stress is equal to or less than 90 MPa (13,053 psi) for the temperature limit proposed.
- For loading operations, repeated thermal cycling should be limited to less than 10 cycles, with cladding temperature variations less than 65°C (117°F) in each cycle.
- For off-normal and accident conditions, the maximum cladding temperature should not exceed 570°C (1058°F).

ISG-22, "Potential Rod Splitting Due to Exposure to an Oxidizing Atmosphere during Short-Term Cask Loading Operations in LWR or Other Uranium Oxide Based Fuel," [80] provides guidance to prevent uranium dioxide-based fuel oxidation that could lead to gross cladding breaches. The options discussed in the guidance are the following:

- Use inert atmospheres,
- Assure the absence of cladding breaches, and
- Determine the time-at-temperature profile to calculate the expected oxidation rate.

1.3 PIRT Process

The PIRT process has been utilized multiple times to support decision-making, research, and/or methodology development. A PIRT process can focus on a specific issue by identifying and ranking all plausible phenomena affecting the safety margin to acceptance criteria for applicable scenarios. For example, the PIRT process was previously used to assess the burnup credit appropriate for spent fuel criticality analysis [77]. PIRT results can provide guidance for initial development of a methodology or technology where regulatory guidance does not currently exist. Most commonly, the process is used to determine where additional data are necessary to assure appropriate understanding of available margins.

In the case discussed in this document, the PIRT process was used to determine the level of knowledge and, therefore, confidence in predicting fuel and cladding performance of spent fuel being loaded, transferred, stored, and transported. If the outcome indicates sufficient confidence in prediction of fuel cladding degradation, it could be used to develop better informed regulatory guidance and utility operations. Vendors or other stakeholders could also use the results to develop a topical report for NRC review to provide an alternative methodology that ensures that fuel performance is adequate during short and long-term storage operations.

According to the NRC, the nine steps of the PIRT process are [82]:

1. Define the issue that is driving the need for a PIRT.
2. Define the specific objectives for the PIRT.
3. Define the hardware, equipment, and scenarios that the PIRT is expected to assess.
4. Define the evaluation criteria, which are the key figures of merit used by the subject-matter experts (SMEs) to judge the relative importance for each phenomenon. All PIRT SMEs must have a clear understanding of the evaluation criteria and how they should be used to rank phenomena.
5. Identify, compile, and review applicable research that captures the experimental and analytical knowledge relative to the issues driving the PIRT.

6. Identify all plausible phenomena.
7. Develop the importance ranking and rationale for each phenomenon. Importance is ranked relative to the evaluation criteria.
8. Assess the level of knowledge and uncertainty in understanding and ability to model each phenomenon.
9. Document the PIRT results.

The results to be documented in the PIRT are:

- The identified phenomena and their associated definitions,
- The ranking of each phenomenon and associated rationale for that ranking, and
- The level of knowledge or associated uncertainty for each phenomenon.

The rationale for selection and ranking phenomena should be given.

1.4 Objective and Report Organization

The objective of this PIRT is to identify, evaluate, and rank the various physical phenomena that affect nuclear fuel cladding performance in a dry storage and transportation environment. Understanding the technical bases of the current fuel performance limits for storage and transportation could support regulatory guidance modifications according to the current state of knowledge on the topic. Development of the PIRT was accomplished via an expert panel elicitation: the experts identified, evaluated, and ranked the most influential phenomena occurring during spent fuel storage and transportation. Additionally, the panel identified 1) scenarios (i.e., external conditions) to be considered for the phenomena evaluation, 2) the parameters that affect the phenomena, 3) the level of understanding in terms of available data, and 4) the quality of the data available, according to the experts' perception.

Membership of the PIRT was established through initial collaboration activities and discussions with the Steering Committee members from several key stakeholder organizations. The list of experts involved is provided in Table 1-2 (additionally, Antoine Ambard of Électricité de France provided input to the initial team meeting). Resumes for the experts and facilitators are provided in Appendix A. Cask vendors and utilities also participated as observers and provided value by answering questions from the expert panel members.

Table 1-2
Fuel Cladding Performance Expert Panel

Name	Affiliation
Ricardo Torres	U.S. Nuclear Regulatory Commission
Brady Hanson	Pacific Northwest National Laboratory
Michael Billone	Argonne National Laboratory
Rose Montgomery	Oak Ridge National Laboratory
Albert Machiels	John Kessler and Associates (JKA)
Joseph Rashid	Structural Integrity Associates

This report is organized following the NRC PIRT process description discussed in Section 1.3 above.

Section 2 discusses the PIRT bases including scenarios, evaluation criteria, and phenomena considered for this exercise (Steps 3 to 5 of the PIRT process).

Section 3 presents the phenomena discussed during the PIRT and the parameters that influence the phenomena. Data available and ranking provided by the PIRT panel are also provided in this section (Steps 4 to 8 of the PIRT process).

Section 4 discusses the effects of simultaneous occurrence of phenomena presented in Section 3 (synergistic, competitive, and aggravating effects).

Section 5 presents the results of the process, summarizes the current state of knowledge, and presents future opportunities for changes in guidance.

Section 2: PIRT Bases

This chapter discusses Steps 3 and 4 of the NRC PIRT process described in Section 1.3: the definition of hardware, equipment, and scenarios that the PIRT is expected to assess; and the evaluation criteria. The review of the applicable research and results relative to the issues driving the PIRT is performed in this chapter and in Section 3.

2.1 Definition of Evaluated Hardware

This exercise focused on nuclear fuel comprised of low enriched ^{235}U uranium dioxide pellets (mixed oxide, advanced fuels and enrichments higher than 5 weight percent were not considered) and cladding of zirconium-based alloys (stainless steel or accident tolerant fuel claddings were not considered) up to the current NRC approved burnup limit of 62 GWd/MTU (rod average). The materials considered are consistent with the vast majority of the commercial fuel designs used in the operating fleet.

A variety of zirconium alloy types are produced by fuel vendors with varying elemental compositions and manufacturing techniques (e.g., ZIRLO®, M5®, Zircaloy 4 (Zr-4), Zircaloy 2 (Zr-2), Optimized ZIRLO™). The minor differences in compositions and manufacturing can affect the behavior of the cladding during reactor operations, such as the susceptibility to certain phenomena and their effects on the end-of-life cladding condition, which is the starting consideration for the spent fuel (wet and dry) storage process. The evaluation conducted in the PIRT process aims at capturing the overall performance of the fuel and cladding system in dry storage conditions. Therefore, the influence of cladding composition and manufacturing techniques is addressed only for those phenomena where considered significant.

2.2 Definition of Scenarios

Scenarios are used to evaluate the variation of the fuel-cladding system response to the phenomena occurring for expected and regulatory-required events.

During this PIRT exercise, the panel agreed upon the evaluation of four scenarios for dry storage conditions and one scenario for transportation, as described below:

- Short-term (ST) operation in dry storage: includes loading, drying, and transfer, without a clearly defined time limit, but which should be expected in the day/week timeframe, and including the possibility of several heating and cooling cycles.
- Storage: normal conditions of dry storage envisions a storage time up to 60 years in one of the currently approved dry storage systems.
- Long-term storage (LT Storage): considers normal conditions of dry storage for a storage time between 60 and 100 years in one of the currently approved dry storage systems.
- Off-normal (ON) conditions: these events occur infrequently during dry storage, although a specific limit of number of occurrences is not identified. For consistency with ISG-11 Revision 3 and with present inspection frequencies, it was assumed that temperatures would remain below 570 °C. Off-normal conditions were considered to generally last less than 48 hours (based on historical operating experience). ISG-11 Revision 3 specifically refers to thermal transients as examples of such off-normal conditions.
- Normal Conditions of Transport (NCT): considers the conditions defined in 10 CFR Part 71.71.

Postulated accidents and hypothetical accident conditions were not considered in this analysis. The panel noted that these are rare events that would require a case-by-case evaluation to demonstrate continued safety or determine recovery actions for the specific conditions. Hence including them would unnecessarily complicate the PIRT evaluation.

Seismic events were not included. The primary reason for exclusion was that seismic events are rare conditions for which a spent fuel cask and its contents would be evaluated on a case-by-case basis. Additionally, the fuel bending loads during a seismic event are expected to be relatively small, except for possible cask tip over under severe earthquakes, in which case the event becomes subject to special-case evaluation [87]. The safety analyses of a dry storage system also generally evaluate the impacts of a cask tip over accident, even when the accident is considered non-mechanistic (i.e., unlikely to occur).

Spent fuel storage in a geologic repository was not considered because of the uncertainties related to the repository conditions and unknown acceptance criteria.

2.3 Evaluation Criteria

The evaluation conducted for this PIRT focuses on the spent fuel rod, namely the cladding and fuel pellets, including their interactions. The structure of the assemblies is not considered.

The evaluation criteria and figures of merit selected by the panel were the following:

- Storage: prevent gross ruptures (10 CFR 72.122(h)(1)) resulting in fuel material dispersion that would adversely affect retrieval of spent fuel for further processing or disposal (10 CFR 72.122(l))
- Transportation: maintain fuel geometry configuration

During the PIRT process, gross rupture was interpreted according to the current definition provided in ISG-1 Revision 2 (i.e., a cladding breach 1-mm wide or larger). Avoiding gross rupture is a requirement in 10 CFR 72.122(h)(1) (or the fuel must be otherwise confined to avoid operational safety problems during removal from storage). Also, 10 CFR 71.55(d)(2) stipulates that the geometric form of the package contents must not be substantially altered. ISG-1 Revision 2 notes that not meeting 10 CFR 72.122(h)(1) would also mean that 10 CFR 71.55(d)(2) might not be met. The panel adopted the gross rupture definition from ISG-1 Revision 2 during the PIRT process, although it was considered non-actionable in practice (i.e., the sizing of breaches in the cladding is not practical). Reevaluation of the definition of gross rupture was recommended for follow up work.

The rod condition after in-reactor operation was considered the starting point for the panel discussion and was described using one of the categories, according to ISG-1 Revision 2 [81], as summarized in Table 1-1.

During the evaluation, the panel considered intact and undamaged fuel (pinhole leak and hairline crack) but did not consider damaged fuel because gross rupture would have already occurred. Additionally, the panel considered specific rod conditions beyond its condition category. For example, the typical order of magnitude of in-reactor cladding hydrogen pickup and hydrogen concentration at the beginning of storage is known for some cladding alloys and was used to draw conclusions about cladding susceptibility to certain phenomena (e.g., hydride reorientation).

2.3.1 Ranking Rules

The panel utilized the following ranking scale to determine the importance of each phenomenon and the ability to satisfy the evaluation criteria:

- **Operability:** Does the phenomenon exist with effects that are detectable for the scenarios that are considered? The answer is yes or no (Y/N). The knowledge and confidence criteria are still important, as they provide the foundation for concluding that a phenomenon is not operable. However, significance will always be low for a non-operable phenomenon.
- **Knowledge**: Are data on the phenomenon available, and are they relevant? The ranking is low, medium, or high (L, M, H), with low assigned when minimum data are available, and high assigned when a satisfactory amount of data are available.

- **Confidence**: What is the quality of the existing data and models (e.g., are they consistent, can the phenomenon be modeled, verified, and replicated)? The ranking is low, medium, or high (L, M, H), with low assigned when there is low confidence in the available data, and high assigned when there is high confidence in the available data.
- **Significance**: To what extent does the phenomenon contribute to gross cladding rupture? The ranking is low, medium, or high (L, M, H), with low assigned when the phenomenon has low significance and likely does not lead to gross rupture, and high when the event has high significance and is likely to cause gross rupture.

Using these ranking rules, the lowest ranking phenomena are rated No for Operability, high for Knowledge, high for Confidence, and low for Significance; the highest-ranking phenomena are rated "Yes" for Operability and high or medium for Significance.

2.4 Phenomena Considered during the PIRT

As mentioned in Step 6 of the PIRT process, the panel used previously established phenomena based on their experience and a number of references [23] [26] [87] to identify those that could affect the integrity of the spent fuel rods: thermal creep, low-temperature (athermal) creep, diffusion-controlled cavity growth, hydrogen induced embrittlement, delayed hydride cracking, hydride reorientation, thermal fatigue, mechanical fatigue, annealing, mechanical overload, fuel oxidation and cladding oxidation.

The interplay between some of these phenomena was also discussed, as presented in Section 4 of this report. Among the phenomena analyzed in Section 3, low-temperature creep, diffusion-controlled cavity growth, and fatigue, were judged as substantially independent of other phenomena and not included in the synergistic effect's discussion.

2.5 Phenomena Discussed but not Considered Further

During the meeting and subsequent interactions by conference call and e-mail, experts presented the latest results and their vision of the phenomena that play a role in the performance of cladding and fuel during transport and dry storage. Some of the phenomena that were originally identified as affecting the integrity of fuel cladding were categorized as inoperable, as described below.

2.5.1 Stress Corrosion Cracking

Stress corrosion cracking (SCC) has occured in zirconium cladding when the following three conditions are present:

- Sufficient tensile stress level
- Material susceptibility

- Aggressive environment (such as the presence of specific fission products such as gaseous forms of iodine compounds)

During dry storage, the environment inside the fuel rods is not expected to be aggressive. When one of the three conditions listed above is not met, the SCC phenomenon is not operative. Therefore, the phenomenon was discussed by the panel, but not ranked.

2.5.2 Radiation Embrittlement during Storage

The exposure of the cladding to radiation from fast neutrons affects the mechanical properties of the cladding. Fast neutrons cause atomic displacements that lead to point defects and dislocations that change the mechanical properties of the cladding (e.g., increased strength and reduced ductility). The phenomenon occurs during in-reactor operation and tends to saturate when the cumulative fast neutron fluence (E>1 MeV) reaches 10^{21} n/cm^2 [38] [36]. In-reactor radiation embrittlement is accounted for in the mechanical properties assumed in the design basis analyses of both transportation packages and dry storage systems.

Embrittlement does not occur outside the reactor because the neutron fluence in dry storage is about five orders of magnitude lower than in-reactor service [87]. Therefore, additional embrittlement during dry storage does not affect the cladding mechanical properties, and the phenomenon was not considered for any of the scenarios. However, annealing of radiation damage cannot be ruled out when storage temperatures are high enough, and the resulting lowering in yield stress and increase in ductility need to be properly taken into account.

Section 3: Discussion and Results

In this section, the phenomena selected for examination that could affect nuclear fuel rod performance in a dry storage and transportation environment are evaluated. Each phenomenon is described, as well as the parameters that affect it and availability and quality of supporting data. Finally, the consensus ranking proposed by the panel for each phenomenon is reported.

3.1 Thermal Creep

3.1.1 Phenomenon Description

Thermal creep can be defined as the tendency of a solid material to deform plastically in a time-dependent manner under the influence of temperature and stress. This phenomenon results from long-term exposure to levels of stress that are below the material yield strength. As shown in Figure 3-1 [49], creep starts at a rapid rate (primary creep) and reaches a steady-state (secondary creep) where the creep rate is essentially constant. The last regime is called tertiary creep and is characterized by an accelerated creep rate, which terminates with rupture.

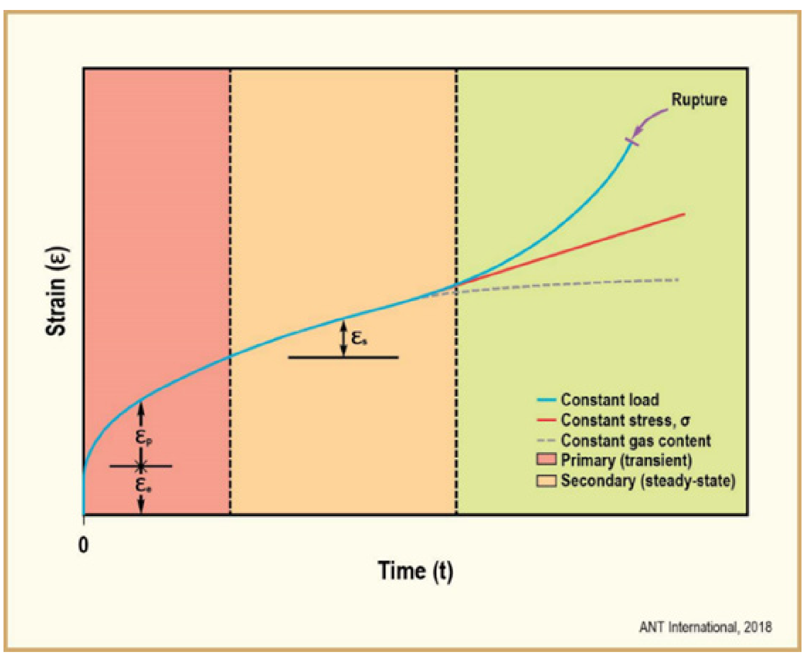

Figure 3-1
Three Regimes of Creep [49]

Thermal creep of spent fuel rods in dry storage is often described as a self-limiting phenomenon that does not follow the above classical definition of creep in which the forcing functions, namely stress and temperature, are held constant with time. Assessment of fuel rod creep assumes that 1) fission gas in the void volume is free to move along the pellet stack and that the amount of fission gas in the void volume of the fuel rod does not change over time, 2) decay heat of the fuel rods decreases over time, and 3) temperature of the fission and fill gas decreases. These factors lead to an overall reduction in the fuel rod cladding stress [60] and cladding creep with time (Figure 3-2). Creep of the cladding would be expected to manifest itself as an increase in rod diameter without a corresponding reduction in rod length, yielding a net increase in rod internal volume, which in itself produces a lower rod internal pressure.

This triad interaction during storage between temperature, pressure and void volume, which is governed by the ideal gas law, is responsible for the gradual decrease in cladding strain rate over time, eventually approaching zero as indicated by the dashed line in Figure 3-1. It should be noted that the large majority of the fission gases produced during reactor operation are confined within fuel pellets, and the trapped gas represents a large potential pressure source if it were released into the free volume of the fuel rod; however, mobility of the gas requires sustained high temperature or a substantially reduced rod internal pressure, which is not usually encountered in dry storage. As the rod internal pressure in the void volume is reduced and the pressure inside the pellets is increased by production of helium (i.e., alpha decay), some additional fission gas that was previously trapped in porosity or in grain boundaries is also released until equilibrium is achieved. The additional gas inventory is expected to be small over the timespan considered (less than 100 years) and it cannot produce a large increase in average rod internal pressure, but it should be noted that helium is a highly mobile gas that is likely to diffuse when fuel temperature exceeds 526 °C [88] [40].

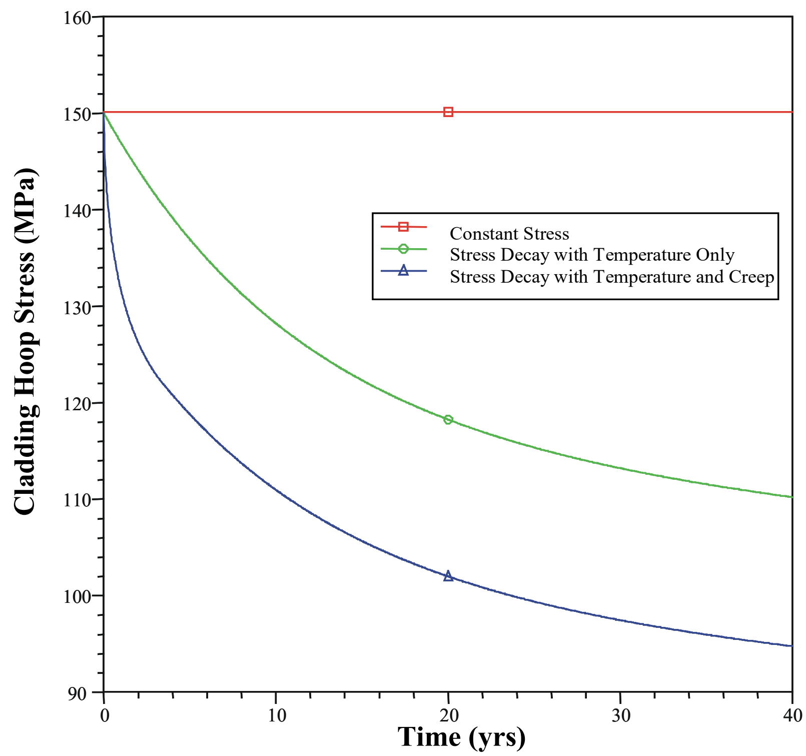

Figure 3-2
Stress History During Dry Storage, Initial Conditions: T = 400 °C, 150 MPa [60]
Copyright 2007 by the American Nuclear Society, LaGrange Park, Illinois.

Initial NRC review guidance relative to spent fuel storage included thermal creep as the mechanism that required limiting peak cladding temperature[2]. A creep limit was present in the first revision of ISG-11 (1% creep strain limit of the cladding in Revision 1) [76], but it was removed in the Revision 2 of ISG-11 [78] that concluded that creep under normal conditions of storage will not cause gross rupture of the cladding and that the geometric configuration of the spent fuel will be preserved, provided that the maximum cladding temperature does not exceed 400 °C.[3]

3.1.2 Available Data and Quality

There are many factors that influence cladding creep rate. For a given temperature and stress, the parameters that affect thermal creep include:

- Alloy composition, such as the presence of tin, niobium, oxygen, sulfur
- Alloy microstructure, such as the degree of recrystallization, grain size and orientation

[2] The German regulations include a limit for the diametral strain ≤ 1% [21], as did the NRC review guidance prior to ISG-11, Rev. 2 [78].

[3] Based on experimental creep data on high-burnup fuel cladding, which show creep strains above 5% with no failure, it was decided that a 1% creep strain limit was not a suitable limit [59 [19] [48].

- Irradiation damage accumulated during reactor operation and any annealing of radiation damage post-irradiation
- Hydrogen content

If excessive creep occurs, it could lead to cladding thinning and rupture. References cited in [87] provide experimental evidence that cladding failures (e.g., gross rupture) are not expected for creep strain below 2% [61] [16]. Earlier work by Woodford supports the conclusions of Raynaud and Einziger, which suggests that strain ruptures would be expected to require strains greater than 10% for the conditions tested [89].

Data on thermal creep are available mainly for temperatures up to 400˚C, with limited datasets up to 570 °C. Above 400˚C, the data on thermal creep is less comprehensive and complicated by effects of annealing and recrystallization. It is known that at temperatures approaching 400˚C, the alloy microstructure changes (i.e., radiation damage annealing, cold work annealing and recrystallization might occur over time), affecting the creep behavior.

References on thermal creep include:

- Machiels, A.J., Waldrop, K., Rudling P. 2019. *Perspective on Thermal Creep and Hydride Reorientation for Dry Storage and Transportation Applications,* International Conference on the Management of Spent Fuel from Nuclear Power Reactors: Learning from the Past, Enabling the Future, IAEA, June 2019, Vienna, Austria.
- Chin, B.A., M.A. Khan, J.C.L. Tarn, and E.L. Gilbert. 1986. *Deformation and Fracture Map Methodology for Predicting Cladding Behavior during Dry Storage*, PNNL-5998, Auburn University, September 1986.
- Seibold, A. and F. Garzarolli. 2002. *Influence of composition and condition on behavior of Zr-Sn-FeCrV alloys*, in Zirconium in the Nuclear Industry: Thirteenth International Symposium, ed. G. Moan and P. Rudling (West Conshohocken, PA: ASTM International, 2002), 743-757.
- Electric Power Research Institute (EPRI). 2001. *Creep Modeling and Analysis Methodology for Spent Fuel in Dry Storage*, EPRI, Palo Alto, CA: 2001. 1003135
- Rashid, Y.R., R. Montgomery, W. Lyon, and R. Yang. 2000. *A Cladding Failure Model for Fuel Rods Subjected to Operational and Accident Transients*, IAEA-TECDOC-1233, IAEA, Vienna, 2000.
- Electric Power Research Institute (EPRI). 2000. *Creep as the Limiting Mechanism for Spent Fuel Dry Storage*, EPRI, Palo Alto, CA: 2000. 1001207.
- Rashid, Y.R. and A. Machiels. 2007. *Threat of Hydride Re-orientation to Spent Fuel Integrity During Transportation Accidents: Myth or Reality?,* Proceedings of the 2007 International LWR Fuel Performance Meeting, Paper ID-1039, San Francisco, September 30-October 3, 2007.

- Einziger R.E., S. D. Atkin, D.E. Stellrecht and Pasupathi, V. 1982. *High Temperature Post-Irradiation Materials Performance of Spent Pressurized Water Reactor Fuel Rods under Dry Storage Conditions*, Nuclear Technology, v. 57, p. 65, 1982.
- Einziger, R.E., H. Tsai, M.C. Billone, B.A. Hilton. 2003. *Examination of Spent PWR Fuel Rods after 15 Years in Dry Storage*, NUREG/CR-6831, Argonne National Laboratory, September 2003.
- Electric Power Research Institute (EPRI). 2016. *Creep and Creep-to-Rupture Testing of Optimized ZIRLO (TM)*, EPRI, Palo Alto, CA: 2016. 3002008040.
- Electric Power Research Institute (EPRI). 2011. *Structural Integrity of Advanced Claddings during Spent Nuclear Fuel Transportation and Storage – Vol. 2: Creep Testing of LK3 Zircaloy-2 Cladding Tubes*, EPRI, Palo Alto, CA: 2011. 1022918.
- Electric Power Research Institute (EPRI). 2011. *Structural Integrity of Advanced Claddings during Spent Nuclear Fuel Transportation and Storage -* Volume 1: *Creep Testing of ZIRLO™ Cladding* Tubes, EPRI, Palo Alto, CA: 2011. 1022919.
- Cappelare, C., R. Limon, C. Duguay, G. Pinte, M. Le Breton, and P. Bouffioux. 2012. *Thermal Creep Model for CWSR Zircaloy-4 Cladding Taking into Account the Annealing of the Irradiation Hardening*, Nuclear Technology, 177:2, 257-272, 2012.
- Goll, W. H. Spilker, and E.H. Toscano. 2001. *Short time Creep and Rupture Tests on High Burnup Fuel Rod Cladding*, Journal of Nuclear Materials, Volume 289, Issue 3, March 2001.

3.1.3 Ranking

The panel agreed upon the following:

- Thermal creep, as an inherent regime for spent fuel in storage, is operable for all scenarios considered in this study. However, it is shown as not operable for LT Storage because at 60 years and beyond, the creep mechanism is designated as low temperature creep (next section), not thermal creep.
- Knowledge of thermal creep is rated high for ST and NCT scenarios. More limited knowledge is attributed to the behavior of fuel when in storage for much longer times (storage and LT storage have medium ranking) and when repetitive off-normal conditions are experienced (ON has low ranking) [15]. Confidence is rated high for all scenarios considered.
- Overall, thermal creep is considered of low significance. This phenomenon is a self-limiting mechanism in closed systems such as fuel rods due to the temperature decrease over time and creep-induced volume expansion, which increases void volume assuming the fission gas is free to travel along the pellet stack. For short-term scenarios (NCT, ON, and ST) creep is not

considered a mechanism of concern with regard to gross rupture[4] due to the limited time frames. For long-term scenarios, the temperature is expected to decrease. Therefore, systematic fuel failure during dry storage by thermal creep is highly unlikely.

The panel identified the limited knowledge concerning creep strain at temperatures higher than 470 °C, and it was recognized that creep strain could become significant in this temperature range and above. Currently, cladding temperatures above 400°C will occur only for ON conditions and be of limited duration. Therefore, even with greater uncertainty on the rate of thermal creep at temperatures up to 570°C, the phenomenon is still considered of low significance, especially due to the self-limiting characteristic of thermal creep deformation in closed systems, as demonstrated in the experiments reported in [15].

Additionally, despite high confidence in data for Zr-4 and Zr-2, the panel recognized that there are limited publicly available data for the newer zirconium alloy claddings (e.g., M5®, Optimized ZIRLO™). The availability of data that correlate the behavior of zirconium alloy to the amount of alloying elements (e.g., Sn, Nb, O_2) [69] helps to predict the behavior despite the absence of experimental data for specific materials.

A summary of the panel evaluation is presented in Table 3-1.

Table 3-1
Ranking Table for Thermal Creep

Phenomena/ Scenarios	Operable	Knowledge	Confidence	Significance
ST	Y	H	H	L
Storage	Y	M	H	L
LT Storage	N	M	H	L
ON	Y	L	H	L
NCT	Y	H	H	L

3.2 Low-Temperature Creep

3.2.1 Phenomenon Description

Low-temperature creep (i.e., athermal creep) refers to the plastic deformation that occurs at temperatures close to room temperature. Key parameters for low temperature creep are stress (which for spent fuel is mainly due to rod internal

[4] For ON conditions at 570°C occurring for 48 hours, thermal creep may occur. Gas-induced hoop stresses could be >130 MPa for some fuel rods and annealing of all irradiation hardening and some cold work are likely to occur. For example, during fabrication all cold-work is annealed and the material becomes recrystallized-annealed for temperatures in the range of 580-620°C for 1-2 hours. [68] [46]

pressure) and time. Other parameters related to the condition of the fuel at the beginning of the storage period also play a role, such as fuel swelling and level of corrosion.

3.2.2 Available Data and Quality

The behavior during long-term storage is uncertain due to the scarcity of data on long time exposure to creep, especially at temperatures close to room temperatures. Very little data is available at long-term dry storage temperatures, and the available data are extrapolated from short-time tests. However, even if data extrapolation has much uncertainty, measurable deformation of the cladding tube by thermal creep would require very high stresses that are not predicted for dry storage.

References include:

- Chin, B.A., M.A. Khan, J.C.L. Tarn, and E.L. Gilbert. 1986. Deformation and Fracture Map Methodology for Predicting Cladding Behavior during Dry Storage, PNNL-5998, Auburn University, September 1986.
- Ankem, R and T. Wilt. 2006. A Literature Review of Low Temperature ($<0.25T_{mp}$) Creep Behaviour of α, α-ß, and ß Titanium Alloys, Center for Nuclear Waste Regulatory Analyses, San Antonio, Texas, September 2006.
- Matsuo. 1987. Thermal Creep of Zircaloy-4 Cladding under Internal Pressure, *Journal of Nuclear Science and Technology*, 24:2. pp. 111-119, February 1987.

3.2.3 Ranking

The panel agreed upon the following:

- Low-temperature creep is not considered operable for the period of time evaluated here due to the very long times required to have a significant strain of the material (approximately 1% in > 30,000 years for unirradiated Zircaloy-4 [11]).
- Knowledge of low-temperature creep and confidence in the data are high for all scenarios but LT storage. Knowledge of LT storage is low and confidence medium because data are limited for performance predictions over periods of 60 years and beyond.
- Low-temperature creep is considered to be of low significance for all scenarios considered in this analysis due to the long times required to have significant strain in the material.

A summary of the panel evaluation is presented in Table 3-2.

Table 3-2
Ranking for Low-Temperature Creep

Phenomena/ Scenarios	Operable	Knowledge	Confidence	Significance
ST	N	H	H	L
Storage	N	H	H	L
LT Storage	N	L	M	L
ON	N	H	H	L
NCT	N	H	H	L

3.3 Diffusion-Controlled Cavity Growth (DCCG)

3.3.1 Phenomenon Description

DCCG postulates that some metallic materials subjected to high temperatures and stress might develop micro-cavities on grain boundaries, leading to decohesion of the metal grains. This phenomenon was considered by the panel because it was the limiting mechanism discussed in the pre-2000 NRC guidance [75]. At that time, diffusion-controlled cavity growth was considered a limiting degradation phenomenon because it was theorized that cavities were present in the material, which could migrate to the grain boundaries, causing decohesion, which would be classified as a gross rupture mechanism. However, these cavities have not been observed in Zr-based claddings.

3.3.2 Available Data and Quality

Cavities have never been observed in fresh or irradiated zirconium alloy. Therefore, it was concluded that diffusion-controlled cavity growth is not an active mechanism for zirconium alloys [21].

Available references include:

- Chin, B.A., M.A. Khan, J.C.L. Tarn, and E.L. Gilbert. 1986. Deformation and Fracture Map Methodology for Predicting Cladding Behavior During Dry Storage, PNNL-5998, Auburn University, September 1986.
- Electric Power Research Institute (EPRI). 2002. *Technical Bases for Extended Dry Storage of Spent Nuclear Fuel,* EPRI, Palo Alto, CA: 2002. 1003416.
- Electric Power Research Institute (EPRI). 2000. *Creep as the Limiting Mechanism for Spent Fuel Dry Storage*, EPRI, Palo Alto, CA: 2000. 1001207.

3.3.3 Ranking

The panel agreed upon the following:

- Diffusion-controlled cavity growth is non-operable for the scenarios considered in this study.
- Knowledge of diffusion-controlled cavity growth and confidence in the data are high for all scenarios except LT Storage, due to possible uncertainties related to long-term exposure to uncertain conditions.
- Diffusion-controlled cavity growth is overall considered a low significance phenomenon.

A summary of the panel evaluation is presented in Table 3-3.

Table 3-3
Ranking Table for Diffusion-Controlled Cavity Growth

Phenomena/ Scenarios	Operable	Knowledge	Confidence	Significance
ST	N	H	H	L
Storage	N	H	H	L
LT Storage	N	M	M	L
ON	N	H	H	L
NCT	N	H	H	L

3.4 Hydrogen-Induced Embrittlement Resulting from Hydrogen Migration

Three of the phenomena considered by the panel are related to the effect of hydrogen on the performance of fuel cladding. Hydrogen is picked up by the cladding during reactor operation due to oxidation of zirconium by the coolant (water). Depending on cladding temperature, a certain amount of hydrogen is soluble in zirconium, according to its solubility curve [44]. When the dissolved hydrogen exceeds the solubility limit of the cladding, zirconium hydrides precipitate in the cladding. Macroscopic hydrides are composed of stacks of smaller, microscopic hydride platelets having planes close to the basal plane of the parent α–Zr grains [55] [58].

The presence of a weakly alloyed zirconium liner in the BWR cladding system acts as a hydrogen sink, absorbing hydrogen and decreasing the hydrogen available for migration and precipitation.

The presence of hydrogen causes different phenomena in the cladding such as hydrogen-induced embrittlement resulting from hydrogen migration, delayed hydride cracking, and hydride reorientation. These phenomena are all related to hydrogen behavior, but are considered in isolation in this report and discussed and ranked in the following sections.

3.4.1 Phenomenon Description

In regard to hydrogen-induced embrittlement resulting from hydrogen migration, hydrogen in the cladding can redistribute itself as a consequence of temperature gradients during operation, cask loading, and long-term storage. When subjected to temperature gradients, hydrogen migrates to the colder regions. In a fuel rod in dry storage, hydrogen mainly moves axially, because of the existing temperature gradients along the length of the fuel rod. Hydrogen can, therefore, migrate towards the end-cap welds, which experience the lowest temperatures, resulting in hydride precipitation and cladding embrittlement. If severe enough, this can lead to ruptures especially under accident conditions. In modern fuel design, chamfers on pellets usually act as low temperature traps that should limit hydrogen mobility along the pellet stack.

3.4.2 Available Data and Quality

Hydrogen migration has been studied by the Central Research Institute of Electric Power Industry (CRIEPI) [66] in both irradiated and unirradiated Zircaloy-4 cladding specimens. Time, temperature (controlling the hydrogen diffusion coefficient), and temperature gradients (including heat of transport) are the main parameters that influence hydrogen migration. The stress distribution within the cladding, which is usually unknown, might play a role in the migration of hydrogen, as is the case in delayed hydride cracking (next section). Migration of hydrogen depends on the diffusion coefficient of hydrogen in solid solution in the alloy matrix. Calculations of diffusion length as a function of temperature indicate that hydrogen-induced embrittlement resulting from hydrogen migration should be considered operable in scenarios involving large axial temperature gradients (requiring high peak cladding temperatures and low temperatures at top and bottom fuel rod ends) for sufficiently long times.

Available references include:

- Kammenzind, B. D. Franklin, H. Peters, and W. Duffin. 1996. "Hydrogen Pickup and Redistribution in Alpha-Annealed Zircaloy-4," Zirconium in the Nuclear Industry: Eleventh International Symposium, E. Bradley and G. Sabol (West Conshohocken, PA: ASTM International, 1996), 338-370.
- Perovic V., Weatherly G.C. and Simpson C.J. 1983. "Hydride Precipitation in α/β Zirconium Alloys," Acta Metall. 31, p. 1381-1391, 1983.
- Puls, M. P. 2012. *The Effect of Hydrogen and Hydrides on the Integrity of Zirconium Alloy Components*, ISSN 1612-1370, Springer-Verlag London 2012.
- Sasahara, A. 2006. *Experiment and Calculation of Axial Hydrogen Migration in Spent Fuel Cladding*, IAEA-SPAR-II, Central Research Institute of Electric Power Industry, November 6-10, 2006.

3.4.3 Ranking

The panel agreed upon the following:

- Hydrogen-induced embrittlement resulting from hydrogen migration is operable for all scenarios considered in this study.
- Knowledge of hydrogen-induced embrittlement resulting from hydrogen migration and confidence in the data is medium for scenarios considered in this study. Though the phenomenon is considered well-understood, a medium ranking was chosen due to the few data points available in the literature.
- Hydrogen-induced embrittlement resulting from axial hydrogen migration is overall considered a low significance phenomenon, because the limited experimental data and assessments show that the amount of hydrogen migrating to the rod ends are too low to cause severe embrittlement.

A summary of the panel evaluation is presented in Table 3-4.

Table 3-4
Ranking Table for Hydrogen Induced Embrittlement Resulting from Hydrogen Migration

Phenomena/ Scenarios	Operable	Knowledge	Confidence	Significance
ST	Y	M	M	L
Storage	Y	M	M	L
LT Storage	Y	M	M	L
ON	Y	M	M	L
NCT	Y	M	M	L

3.5 Delayed Hydride Cracking

3.5.1 Phenomenon Description

Delayed hydride cracking (DHC) refers to the migration of hydrogen to a crack tip (incipient defect) favored by local tensile stress gradients where embrittlement of the near-tip region due to hydride precipitation and fracturing of the precipitated hydrides results in the propagation of the crack. When cracking occurs, the DHC process repeats itself over and over again, until the crack eventually propagates through the entire cladding thickness.

Stress intensity factors ($K_I = f\sigma\sqrt{\pi\, a}$, where "f" is a crack shape factor, "σ" is the stress, and "a" is the flaw size) in SNF cladding are typically well below the critical stress intensity factors ($K_{IH} = f\sigma\sqrt{\pi a_c}$, where "$a_c$" is the critical flaw size) required for sustaining crack growth. Knowledge of K_{IH} is important because it can be used to evaluate the structural integrity of components and their resistance to failure by estimating the size of the flaw that can be tolerated under postulated or actual conditions.

3.5.2 Available Data and Quality

DHC has been extensively studied, although most of the studies have been performed on CANDU pressure tubes (2.5% Nb) and Zircaloy-2 cladding [87]. The parameters that influence DHC are the following:

- Presence and amount of hydrogen
- Existing cladding cracks
- Stress intensity (hoop stress)
- Temperature history (e.g., heating versus cooling, cooling rate)

The NRC addresses the technical basis (and supporting references) on the credibility of DHC to impact fuel performance during dry storage in NUREG-2214 [87]. NUREG-2214 references sources of K_{IH} values [10] [45] which range from 2.5 to 14 MPa$\sqrt{m}$, with most of the data around 5-6 MPa$\sqrt{m}$ at temperatures of interest to dry storage and transport. Although new data on stress intensity factors continue to be generated by the international community, the current data available on DHC are mainly from tests performed on either Zr-2 or Zr/2.5 Nb, with limited information generated for Zr-4 and no information for ZIRLO® and M5®. Additional gaps concerning DHC knowledge include the following:

- The effect of various degrees of hydride reorientation on K_{IH}.
- The effect of the hydride phase transformation from γ to δ phase at temperatures below 180°C.

The main references available are the following:

- Hanson, B. Alsaed, C. Stockman, D. Enos, R. Meyer, and K. Sorenson. 2012. *Gap Analysis to Support Extended Storage of Used Nuclear Fuel*, PNNL-20509, Revision 0, January 2012.
- Cox, B. 1997. *Hydrogen Trapping by Oxygen and Dislocations in Zirconium Alloys*, Journal of Alloys and Compounds, Volume 256, Issues 1-2, July 1997.
- Kubo, T. Y. Kobayashi, and H. Uchikoshi. 2012. *Measurements of Delayed Hydride Cracking Propagation Rate in the Radial Direction of Zircaloy-2 Cladding Tubes*, Journal of Nuclear Materials, Volume 427, Issue 1-3, August 2012.
- Sasahara, A., and T. Matsumura. 2008. *Post-Irradiation Examinations Focused on Fuel Integrity of Spent MWR-MOX and PWR-UO_2 Fuels Stored for 20 Years*, Nuclear Engineering and Design, Volume 238, pp. 1250-1259, 2008.
- Electric Power Research Institute (EPRI). 2002. *Technical Bases for Extended Dry Storage of Spent Nuclear Fuel*, EPRI, Palo Alto, CA: 2002. 1003416.

- Chan, K.S. 2013. *An Assessment of Delayed Hydride Cracking in Zirconium Alloy Cladding Tubes Under Stress Transients*, International Materials Reviews, Volume 58, Number 6, pp. 349-373, 2013.
- Electric Power Research Institute (EPRI). 2000. *Creep as the Limiting Mechanism for Spent Fuel Dry Storage*, EPRI, Palo Alto, CA: 2000. 1001207.
- Electric Power Research Institute (EPRI). 2001. *Fracture Toughness Data for Zirconium Alloys: Application to Spent Fuel Cladding in Dry Storage*, EPRI, Palo Alto, CA: 2001. 1001281.
- Electric Power Research Institute (EPRI). 2011. *Delayed Hydride Cracking Considerations Relevant to Spent Nuclear Fuel Storage*, EPRI, Palo Alto, CA: 2011. 1022921.
- Patterson, C. F. Garzarolli, R. Adamson, and K. Coleman. 2015. *Dry Storage Handbook: Fuel Performance in Dry Storage*, Advanced Nuclear Technology International, Mölnlycke, Sweden, March 2015.
- Rodgers, D., Griffiths, M., Bickel, G., Buyers, A., Coleman, C., Nordin, H., and St. Lawrence, S. 2016. *Performance of Pressure Tubes in CANDU Reactors,* CNL Nuclear Review, Vol. 5, pp. 1-15, 2016.
- Dutton, R. K. Nuttall, M.P. Puls and L.A. Simpson. 1977. *Mechanism of Hydrogen-Induced Delayed Hydride Cracking in Hydride Forming Materials*, Metall. Trans. A, 8A 1553–1562.
- Shimada, S. E. Etoh, H. Hayashi, and Y. Tukuta. 2004. *A metallographic and fractographic study of outside-in cracking caused by power ramp tests*, J. Nucl. Materials. 327. 97–113.
- Grigoriev, V., Jakobsson, R. 2005. *Delayed hydrogen cracking velocity and J-integral measurements on irradiated BWR cladding*, in *Zirconium in the Nuclear Industry: Fourteenth International Symposium*, ed. P. Rudling and B. Kammenzind (West Conshohocken, PA: ASTM International, 2005), 711-728.

Based on the available data, the NRC considered a reference stress intensity factor (K_{IH}) value of 5.0 MPa$\sqrt{m}$ [87]. For illustration, minimum (critical) crack sizes to sustain DHC in PWR fuel rod 17 x 17 design, using the definition of the stress intensity factor shown above, are tabulated in Table 3-5 when K_{IH} = 5.0 MPa$\sqrt{m}$.

Table 3-5
Minimum (Critical) Crack Sizes for K_{IH} = 5.0 MPa$\sqrt{m}$ for 17 x17 PWR Fuel Rods

Stress [MPa]	Minimum (critical) Crack Size [μm]		Relative Size of Crack vs. Cladding Wall Thickness [=570 μm]
	Elliptical flaw	Sharp flaw	
60	2763	1806	>100%
80	1554	1036	>100%
100	995	663	>100%
120	691	461	~80% (sharp flaw)

As can be seen by inspection of Table 3-5, critical flaw sizes are typically greater than the cladding wall thickness for K_{IH} = 5.0 MPa$\sqrt{m}$. For a flaw equal to 25% through the cladding wall (~140 μm for 17 x 17 PWR fuel design) to propagate, K_{IH} would have to be very low (~1.3 for a semi-elliptical flaw or 1.6 MPa$\sqrt{m}$ for a sharp flaw). There is no evidence for K_{IH} values this low [22].

3.5.3 Ranking

The panel agreed upon the following:

- DHC is operable for storage and LT storage. DHC is not considered operable for ST, ON, and NCT due to the short duration of these scenarios.
- Knowledge of DHC and confidence in the data are high for ST, ON, and NCT. For the other scenarios, the knowledge was ranked medium and the confidence low due to:
  - Lack of data (data are available mainly for CANDU zirconium alloy and Zr 2).
  - Discrepancies between different proposed theories [22].
  - Unknown effects of irradiation [63].
- DHC is considered low significance for all scenarios because cracks have to be unrealistically large to sustain DHC. In addition, DHC cracks are not expected to lead to gross rupture of the rods (leak-before-break principle).

A summary of the panel evaluation is presented in Table 3-6.

Table 3-6
Ranking Table for Delayed Hydride Cracking

Phenomena/ Scenarios	Operable	Knowledge	Confidence	Significance
ST	N	H	H	L
Storage	Y	M	L	L
LT Storage	Y	M	L	L
ON	N	H	H	L
NCT	N	H	H	L

3.6 Hydride Reorientation Cladding Embrittlement

3.6.1 Phenomenon Description

Hydride reorientation (HRO) requires a temperature excursion whereby hydrides in fuel rod cladding that are oriented in the circumferential-axial direction are dissolved during the heating phase of the temperature excursion and then reprecipitate in a radial-axial direction during the cooling phase. Reorientation of hydrides from a circumferential to a radial direction requires that the applied hoop stress exceed a threshold, or critical, stress. The threshold stress is the minimum value of the cladding hoop stress which is required before reorientation is first reliably detected in cladding specimens subjected to thermal cycling. As the applied hoop stress increases beyond the threshold stress value, the fraction of reoriented hydrides increases.

The cladding tube manufacturing process generates a texture in the cladding that favor hydride precipitation in the form of platelets in a predominantly tangential (or circumferential) orientation. This hydride morphology evolves very slowly during reactor operation as a consequence of hydrogen uptake during the waterside corrosion process, causing significant, but operationally tolerable, changes in cladding mechanical properties.

The thermal cycle inherent to dry storage (possibly multiple thermal cycles in rare cases) causes some or all hydrides to dissolve upon heating and re-precipitate upon cooling. Re-precipitation can result in a mixed, circumferential, and radial hydride structure, depending on temperature and cladding hoop stress histories. A radially oriented hydride structure can have a significant effect on cladding mechanical properties and failure limits, potentially causing embrittlement in the lower range of temperatures (<200°C).

3.6.2 Available Data and Quality

The reorientation threshold stress (or critical stress) for hydride reorientation depends on several factors:

- The quantity of hydrogen present in the cladding after operation.
- Partitioning of the cladding hydrogen content between hydrogen in solid solution and hydrogen in the form of hydrides; such partitioning is determined by cladding hydrogen content and temperature.
- Temperature. Higher cladding temperature results in lower threshold stresses.
- Cladding microstructure: texture, amount of residual cold work and recrystallization (residual stress, grain shape and size).
- Neutron fluence.

HRO has not been extensively investigated for irradiated cladding subjected to ON temperatures above 400 ˚C. The amount of hydrogen in solution at these temperatures will vary between 200 wppm (at 400˚C) and >700 wppm (at 570˚C) if that much hydrogen is present in the cladding. Cladding hoop stresses at 570˚C could be high enough to precipitate radial hydrides during cooling. The radial hydride stacking lengths increase as the cooling rate decreases.

The impact of hydride reorientation on mechanical properties is negligible for fuel rod designs that incorporate an inner layer of weakly alloyed zirconium, such as BWR barrier cladding, or an outer layer of highly corrosion resistant alloy, such as Duplex cladding.

Minimization of HRO is considered in ISG-11, Revision 3, by limiting the peak cladding temperatures to a maximum of 400 ˚C and limiting the temperature change during drying-transfer operation to less than 65 ˚C [86]. Data [2] [4] [5] [6] have shown that hydride reorientation can occur at temperatures below 400 ˚C when cladding hoop stresses are higher than the applicable reorientation threshold stress. At temperatures between 300 to 400 ˚C, hoop stress values typically range from 60 to 90 MPa for the current inventory of spent fuel in storage. The limit on the number of thermal cycles was included in ISG-11 Revision 3 based on earlier testing at high cladding hoop stresses (>140 MPa) and fast cooling rates (>25˚C/h) that found multiple temperature cycles could enhance the lengths of radial hydrides. However, more recent results at relevant hoop stress limits and slower cooling rates (5 ˚C/h) found no effect from multiple drying cycles on radial hydride lengths and cladding ductility [6].

Available references are the following:

- Aomi, M., B. Toshikazu, M. Toshiyasu, K. Katsuichiro, Y. Takayoshi, S. Yasunari and T. Toru. 2008. “Evaluation of Hydride Reorientation Behaviour and Mechanical Properties for High-Burnup Fuel-Cladding Tubes in Interim Dry Storage,” *Journal of ASTM International*, Volume 5, Number 9, 2008.

- Billone, M.C., T.A. Butseva, Z. Han, and Y.Y. Liu. 2013. *Embrittlement and DBTT of High Burnup PWR Fuel Cladding Alloys*, ANL-13/16, Argonne National Laboratory, September 2013.
- Billone, M.C., T.A. Butseva, Z. Han, and Y.Y. Liu. 2014. *Effects of Multiple Drying Cycles on High-Burnup PWR Cladding Alloys*, ANL-14/11, Argonne National Laboratory, September 2014.
- Billone, M.C., T.A. Butseva, M.A. Martin-Rengel. 2015. "Effects of Lower Drying-Storage Temperatures on the DBTT of High-Burnup PWR Cladding," ANL-15/21, Argonne National Laboratory, August 2015.
- Motta, A., L. Capolungo, L-Q Chen, M. Cinbiz, M. Daymond, D. Koss, E. Lacroix, G. Pastore, P-C Simon, M. Tonks, B. Worth, and M. Zikry. 2019. "Hydrogen in Zr alloys: A Review," *Journal of Nuclear Materials*, Vol. 518, May 2019, pp. 440-460.
- Puls M.P. 2012. Effect of Hydrogen and Hydrides on the Integrity of Zirconium Alloy Components – Delayed Hydride Cracking, ISSN 1612-1370, Springer-Verlag, London, UK, 2012.
- Perovic V., Weatherly G.C. and Simpson C.J. 1983. Hydride Precipitation in α/β Zirconium Alloys, Acta Metall. 31, p. 1381-1391, 1983.
- Pettersson, K. 2007. The Ductility of Radial Hydrides, ASTM Workshop on Hydride Reorientation in Zirconium Alloys, January 29-February 1, 2007, Anaheim, California.
- U.S. Nuclear Regulatory Commission. 2017. *Mechanical Fatigue Testing of High-Burnup Fuel for Transportation Applications*, NUREG/CR-7198, Revision 1, ORNL/TM-2016/689, Oak Ridge National Laboratory, October 2017.
- U.S. Nuclear Regulatory Commission. 2018. *Dry Storage Transportation of High Burnup Spent Nuclear Fuel,* NUREG-2224, July 2018.

3.6.3 Ranking

The panel agreed upon the following:

- Cladding embrittlement caused by hydride reorientation is operable for storage and LT storage. There are different opinions from the experts as to whether HRO is operable or not for NCT. NCT would typically occur over days or weeks which could be too short for a large enough decrease in temperature to reprecipitate and reorient hydrides. Yet HRO could be operable during NCT at temperatures in the range of about 200°C to 335°C if cladding hoop stress was high enough and sufficient cooling occurred during the transportation evolution. To activate the phenomenon, the cladding would need to have: a) appreciable dissolution of hydrides caused by a temperature increase during cask loading and drying combined with b) sufficient hoop stress (i.e., greater than the threshold stress) during reprecipitation of the hydrides as the cladding temperature decreases. Brittle fracture may occur when the cladding is subjected to pinch loads, its temperature decreases, and the lower temperature results in the fracture strength of cladding becoming larger than the fracture strength of the hydrides.

- Knowledge of hydride reorientation and confidence in the data are high for ST, and are medium for the other scenarios.
- Hydride reorientation is of low significance for ST and ON because embrittlement temperatures tend to be below the operating temperature. The significance is medium for scenarios that occur for longer times where the temperature is lower, such as storage and LT storage. Should hydride reorientation be suspected to occur at a significant level, these conditions would be more problematic due to the temperature decrease with time and the resulting loss in ductility. Hydride reorientation is of low significance during NCT because, while some reorientation during NCT cannot be ruled out, it would not be significant, and the loads during NCT are low compared to the load capacity of the cladding-fuel system.

The panel agreed upon the need for an external load, in addition to HRO, to challenge the structure, which could result in fracture due to loss of cladding ductility. This external load could be present, or postulated, during hypothetical transportation and handling accidents. A summary of the panel evaluation is presented in Table 3-7.

Table 3-7
Ranking Table for Hydride Reorientation

Phenomena/ Scenarios	Operable	Knowledge	Confidence	Significance
ST	N	H	H	L
Storage	Y	M	M	M
LT Storage	Y	M	M	M
ON	N	M	M	L
NCT	Y or N	H	H	L

3.7 Thermal Fatigue

3.7.1 Phenomenon Description

Thermal fatigue occurs when the cladding is exposed to cyclic stresses due to repetitive fluctuations of temperature (e.g., temperature cycling between day and night and seasonal cycling). The magnitude and frequency of temperature cycling influence the occurrence of damage (cracks) of the material.

3.7.2 Available Data and Quality

A conservative maximum cladding temperature daily change due to a seasonal or diurnal cycle of 25°C is expected to cause 10-30 MPa fluctuations in cladding hoop stress. Considering one steady-state temperature cycle every day, 21,900 thermal cycles are expected for 60 years. According to data in the literature, Zr-4 is expected to be capable of sustaining up to 10^7 cycles of 260 MPa stress amplitude without rupture [74].

Available references include:

- U.S. Nuclear Regulatory Commission (NRC). 2019. Managing Aging Processes in Storage (MAPS) Report, NUREG-2214, July 2019.
- Devoe, R. and K.R. Robb. 2015. "COBRA-SFS Dry Cask Modeling Sensitivities in High-Capacity Canisters." Proceedings of the International High-Level Radioactive Waste Management Conference, April 12–16, 2015. Paper No. 12701. Charleston, South Carolina. 2015.
- Lin, X. and G. Haicheng. 1998. "High Cycle Fatigue Properties and Microstructure of Zirconium and Zircaloy-4 Under Reversal Bending." Materials Science and Engineering A. Vol. 252. pp. 166–173. 1998.
- Raynaud, P. and R.E. Einziger. 2015. "Cladding Stress During Extended Storage of High Burnup Spent Nuclear Fuel." Journal of Nuclear Materials. Vol. 464. pp. 304–312. September 2015.

3.7.3 Ranking

The panel agreed upon the following:

- Thermal fatigue is operable for all scenarios considered in this study.
- Knowledge of thermal fatigue and confidence in the data are high for all scenarios considered in this study.
- Thermal fatigue is overall considered of low significance.

Over the course of storage (Storage and LT Storage), the cladding typically sees one cycle with a significant temperature change (200-300°C). During short term operation (ST) such as loading and vacuum drying, additional thermal cycles may be experienced with smaller temperature changes (per ISG-11 Revision 3, the number of cycles for vacuum drying is limited to ten cycles with differences in temperature limited to 65 °C). Although the temperature change can be large, the cycles are very few. Therefore, the panel attributed a low significance to thermal fatigue during ST.

During storage, thermal cycling and related fatigue are caused by diurnal and seasonal temperature changes. The available data show that thermal fatigue is not significant due to the high thermal inertia of the canisters currently used and the low-temperature differential. Note that a much smaller fuel cask system might possibly turn thermal fatigue into a more significant phenomenon due to the lower thermal inertia. After these considerations, the panel agreed on the results in Table 3-8.

Table 3-8
Ranking Table of Thermal Fatigue

Phenomena/ Scenarios	Operable	Knowledge	Confidence	Significance
ST	Y	H	H	L
Storage	Y	H	H	L
LT Storage	Y	H	H	L
ON	Y	H	H	L
NCT	Y	H	H	L

3.8 Mechanical Bending Fatigue

3.8.1 Phenomenon Description

Mechanical fatigue, also called cyclic loading, occurs when the material is exposed to repeated and fluctuating stresses, with a maximum value less than the material tensile strength. The exposure to cyclic loading can cause microstructural damage that can lead to macroscopic cracks and failure at loads less than the material yield stress.

3.8.2 Available Data and Quality

To evaluate fatigue rupture in rod bending due to cyclic stresses (such as those associated with transportation). ORNL developed a cyclic integrated reversible-bending fatigue tester (CIRFT) under an NRC research program to test high-burnup fuel rods under cyclic fatigue conditions [85]. The results of the fatigue tests can be compared with data produced, collected and analyzed by Sandia National Laboratories (SNL) and PNNL during a multi-modal transportation test that included heavy haul truck, ship, and rail tests. The rail test data shows that the total accumulated cladding damage during transportation is below 1E-10 [37]. Comparing the ORNL fatigue dataset shows that a large margin exists relative to the PNNL measured and modeled transportation loads, including the ORNL fatigue data for rods that were heat-treated to simulate a vacuum drying thermal cycle. According to these calculations, 10 billion cross-country trips (2,000 miles distance per trip) are required to challenge the fatigue strength of irradiated fuel cladding [37].

Note that this PIRT does not encompass earthquake events, but a seismic event is too short to lead to fuel rod fatigue failure.

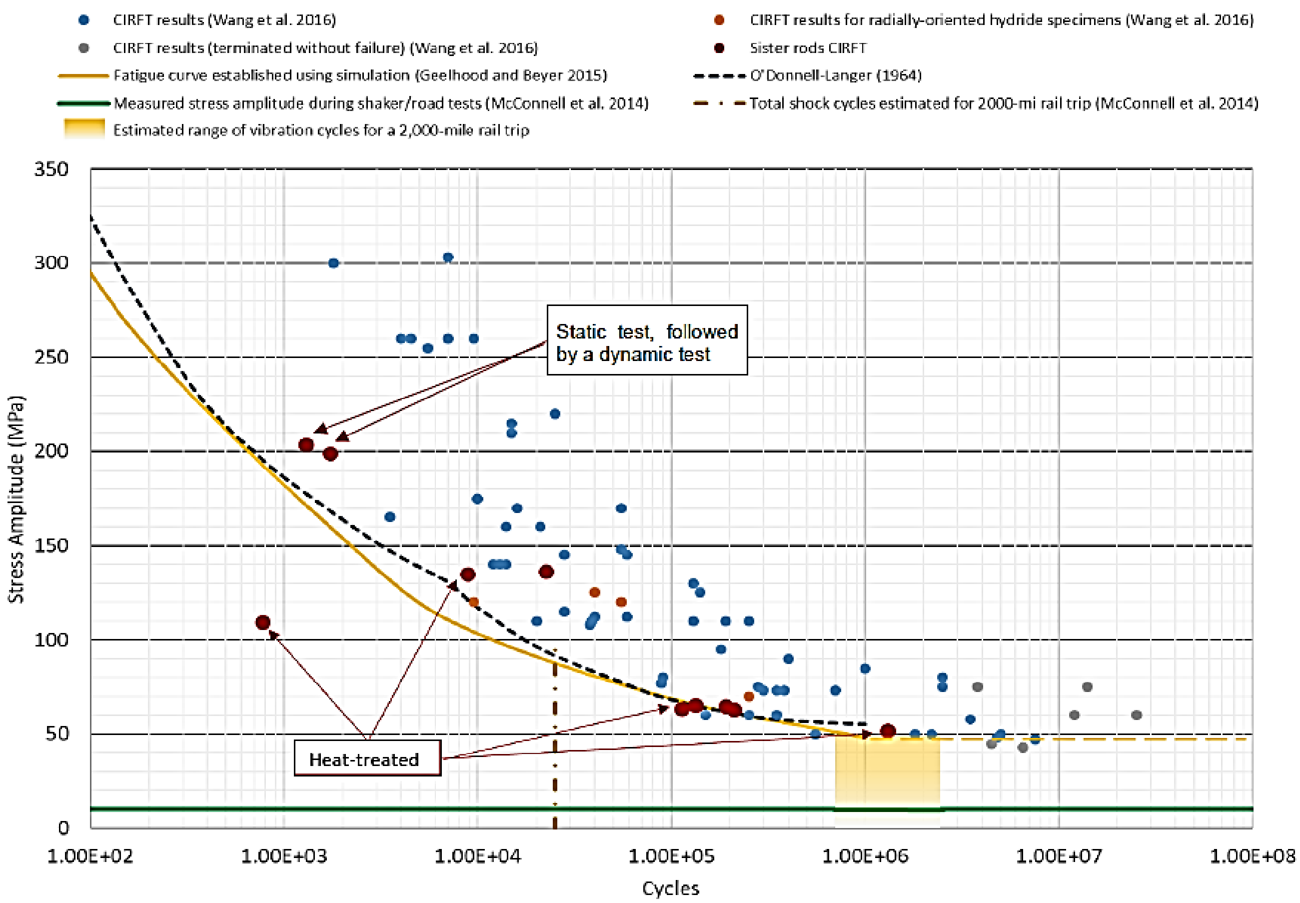

Figure 3-3
Stress Amplitude as Function of Cycles to Failure for Sister Rods [51]

Available references include:

- Hanson, B. for E. Kalinina. 2019. International Multi-Modal Surrogate Spent Nuclear Fuel Transportation Test: The SNF Transportation Test Triathlon, PNNL-147626, IAEA Technical Meeting Technical and Operational Issues Related to the Transportation of High-Burnup and Irradiated Mixed Oxide Fuels and the Transportability of Long-Term Stored Spent Fuel, Vienna, September 2019.
- U.S. Nuclear Regulatory Commission. 2017. Mechanical Fatigue Testing of High-Burnup Fuel for Transportation Applications, NUREG/CR-7198, Revision 1, ORNL/TM-2016/689, Oak Ridge National Laboratory, October 2017.

3.8.3 Ranking

The panel agreed upon the following:

- Mechanical fatigue is not operable during ST and ON scenarios due to the low-stress amplitude and short duration. Purely mechanical cyclic loading is not expected during dry storage operations; therefore, mechanical fatigue is not considered operable during storage and LT storage. The phenomenon is considered operable only during NCT.
- Knowledge of mechanical fatigue and confidence in the data are high for all scenarios considered in this study.
- Mechanical fatigue is overall considered as low significance due to the large margin shown by the available data for the distances of travel in the U.S. and packaging/transport systems used for SNF.

A summary of the panel evaluation is presented in Table 3-9.

Table 3-9
Ranking Table for Mechanical Fatigue

Phenomena/ Scenarios	Operable	Knowledge	Confidence	Significance
ST	N	H	H	L
Storage	N	H	H	L
LT Storage	N	H	H	L
ON	N	H	H	L
NCT	Y	H	H	L

3.9 Annealing of Irradiation Defects in the Rod Cladding

3.9.1 Phenomenon Description

Irradiation defect annealing occurs when the material is exposed to elevated temperatures for a long enough period of time. This phenomenon results in recovery of in-reactor irradiation-induced damage and possibly recovery of defects intentionally introduced by cold work during cladding manufacture for sustained and elevated temperatures.

3.9.2 Available Data and Quality

Key parameters for the annealing phenomenon are the temperature history, time at the elevated temperature, alloy composition, microstructure, and hydrogen content. Annealing is not considered a degradation mechanism per se, but it induces changes in the microstructure that can potentially affect other phenomena and the safety margins for drop accident scenarios.

Available references include:

- Bouffioux P., A. Ambard, A. Miquet, C. Cappelaere, Q. Auzoux, M. Bono, O. Rabouille, S. Allegre, V. Chabretou, and C.P. Scott. 2013. "Hydride Reorientation in M5® Cladding and its Impact on Mechanical Properties." Top Fuel 2013, Paper 8557, pp. 879-886.
- Bouffioux P., and L. Legras. 2000. "Effect of Hydriding on the Residual Cold Work Recovery of Zircaloy-4 Cladding Tubes. Consequences on the Creep Behavior of the Cladding Tubes Under Dry Storage Conditions." ANS International Topical Meeting on LWR Fuel Performance. Park City, Utah, 2000.
- Legras L. 1998. Influence of Transport and Dry Storage on the Metallurgical Structure of the Cladding Material. HT-41/98/008/A, Document EDF R&D, 1998.

3.9.3 Ranking

Temperatures during dry storage and transportation conditions are lower than the typical temperatures required for annealing, but the fuel is exposed to moderate temperatures for a long period of time and some partial annealing of the irradiation-induced effects can occur. The panel agreed upon the following:

- Annealing is operable for all scenarios considered except for LT Storage when the temperature drops below the temperature at which annealing of the radiation damage occurs.
- Knowledge and confidence are high for ST, ON, and NCT and decrease to medium for storage conditions and low for LT storage due to uncertainties in the effect of low temperature over the long-time scale.
- Annealing is presently considered of low significance because of the limited amount of annealing due to lower-than-postulated temperatures in existing storage systems and the increase in ductility is more beneficial than the decrease in yield strength.

A summary of the panel evaluation is presented in Table 3-10.

Table 3-10
Ranking Table of Annealing of Irradiation Defects in Cladding

Phenomena/ Scenarios	Operable	Knowledge	Confidence	Significance
ST	Y	H	H	L
Storage	Y	M	M	L
LT Storage	N	L	L	L
ON	Y	H	H	L
NCT	Y	H	H	L

3.10 Mechanical Overload

3.10.1 Phenomenon Description

Mechanical overload is defined as cladding stress due to fuel pellet swelling resulting from alpha decay damage accumulation, retained helium in the fuel matrix, or pellet-cladding interactions (i.e., phenomena internal to the rod).

Alpha decay of transuranic nuclides in spent fuel results in the production of helium. Fuel pellet swelling due to the precipitation of He bubbles in the UO_2 fuel matrix, or possibly increased rod internal pressure from He released into the rod's free volume could affect cladding performance during storage. After 100 years of storage, the production of helium in spent UO_2 fuel with assembly-average burnup between 50 and 60 GWd/MTU is slightly over 100 cm^3 at room temperature or ~5 x 10^{-3} mol [64]. Assuming total release of helium and an end of life (EOL) void volume in a PWR rod of ~13 cm^3 yields a helium contribution to the rod internal pressure of ~0.8 MPa. Given that EOL rod internal pressures for PWR fuel rods (including those with integral fuel burnable absorber) are of the order of 4 to 6 MPa at room temperature [62], this He contribution would represent a pressure increase of 13 to 20% after 100 years. Assuming a rod average temperature of greater than ~175 °C (or ~450K) at the beginning of storage and ~100 °C (or ~373K) after 100 years, results in a decrease of temperature (and pressure) ≥ 17 % over 100 years of storage, which would compensate for the added He partial pressure.

Thermal desorption studies and defect annealing studies showed that, at the temperatures expected for dry storage and transport, diffusion of helium is not likely [88]. Studies on He release from spent uranium dioxide fuel indicates that not only is He retained in the UO_2 material during the first 100 years of storage, but that temperatures greater than ~526 °C (or ~800 K) are needed to release the He from the UO_2 fuel [88] [40]. Therefore, it can be reasonably assumed that He will remain trapped in the fuel matrix under conditions typical of storage and transportation.

Fuel swelling (potential for mechanical overload) may result from (1) alpha decay damage accumulation and (2) retained He in the fuel matrix (since it can be assumed that it will not be released), and 3) pellet-cladding interactions.

- Alpha decay damage: Investigations of damage by XRD, TEM and Vickers Hardness have shown that the fluorite crystalline structure of UO_2 is preserved but with an increase in lattice parameters (i.e., swelling). The latter is estimated to be 0.2 – 0.3% up to 100 years of storage for UO_2 fuel (maximum swelling is 0.4 – 0.5% for > 100 years) [64].
- Helium accumulation: Proposed He solubility limits [65] indicate that as He is formed, it should precipitate in the form of bubbles in UO_x fuel a century after removal from the reactor. However, the presence of irradiation defects – particularly vacancies—in the spent fuel should increase the He solubility. For example, the quantity of He that infuses under irradiation (in-reactor)

appears to be considerably greater than the published helium solubility limits in UO_2 [32]. According to Ab-initio calculations [56], He is soluble in stoichiometric UO_2 containing uranium vacancies without any increase in the crystal lattice parameter (i.e., no swelling). Given that the spent fuel already contains pores and bubbles in parts of its structure, in which He can be partially accommodated [88], swelling due to He accumulation does not appear to be an issue, at least up to 100 years of storage for UOx fuel, and swelling from alpha decay damage may be all that needs to be taken into account.

- Pellet-cladding interaction (PCI): As the fuel is operated longer in the reactor, the cladding and pellet begin to interact in two ways: 1) developing a strong mechanical contact that results in the pellet adhering to the cladding, with the pellet loaded in compression and the cladding in tension and 2) developing a chemical pellet–cladding diffusion bond condition at the interface at high burnup [70]. In addition, early in operation, the fuel pellet develops cracks which can extend to the edge of the pellet at the fuel-cladding interface. While alpha decay damage and helium accumulation are the primary factors impacting the potential for mechanical overload, pellet cracks at the cladding interface and pellet-cladding bonding introduce local stress risers that may have to be considered for mechanical overload conditions [29].

3.10.2 Available Data and Quality

The parameters that influence mechanical overload are:

- Fuel-cladding interactions including mechanical interference and mechanical bonding
- Fuel pellet microstructure and its ability to retain helium
- Burnup
- Helium mobility (diffusion)

References available include:

- Ferry, C. C. Poinssot, V. Broudic, C. Cappelaere, L. Desgranges, P. Garcia, C. Jegou, P. Lovera, P. Marimbeau, J-P Piron, A. Poulesquen, D. Roudil, J-M Gras and P. Bouffioux. 2005. Synthesis on the Spent Fuel Long Term Evolution, CEA report CEA_R-6084, 2005.
- Petit T., P. Martin, M. Ripert, N. Olivier, A. Chevarier, and N. Moncoffre. 2001. "Localization of helium and volatile fission products in uranium dioxide: ab-initio and x-ray absorption spectroscopy studies." ICEM 2001, Session 12, Bruges, Belgium
- Rondinella, V.V., Wiss, T., Maugeri, E., Colle, J-Y. 2011. "Effects of helium build-up on nuclear fuel evolution during storage," Proc. IHLRWMC 2011, Albuquerque, New Mexico, 2011, ISBN 9781617828508, Curran Associates, NY. 2011, 230–233.

- Rufeh F., Olander D.R., Pigford T.H. 1965. "The solubility of helium in uranium dioxide," *Nuclear Science and Engineering*, 23, pp. 335-338, 1965.
- Sercombe, J. Aubrun, and C. Nonon. 2012 "Power ramped cladding stresses and strains in 3D simulations with burnup-dependent pellet–clad friction," *Nuclear Engineering and Design*, Volume 242, pp. 164-181, 2012.
- Wiss, T. J-P Hiernaut, D. Roudil, J-V Colle, E. Maugeri, Z. Talip, A. Janssen, V. Rondinella, R. Konings, H-J Matzke, and W. Weber. 2014. "Evolution of Spent Nuclear Fuel in Dry Storage Conditions for Millennia and Beyond," *Journal of Nuclear Materials*, Volume 451, pp. 198-206, April 2014.

3.10.3 Ranking

The panel agreed upon the following:

- Mechanical overload is a potential generic issue for either wet or dry storage, thus it was considered operable for storage, LT storage, and NCT.
- The knowledge of mechanical overload was ranked low because there are uncertainties on whether helium is trapped in fuel vacancies (without causing swelling) or if over long time periods some helium migrates outside the fuel, increasing the stress on the cladding.
- Mechanical overload significance is low for short term scenarios such as ST and ON. For storage and LT storage the significance increases to medium because of the amount of helium generated over time. A medium significance ranking was attributed to NCT due to the possibility of additional stresses occurring during transport.

Mechanical overload is a much greater issue for storage of MOX fuel and possibly for disposal scenarios.

A summary of the panel evaluation is presented in Table 3-11.

Table 3-11
Ranking Table for Mechanical Overload

Phenomena/ Scenarios	Operable	Knowledge	Confidence	Significance
ST	N	L	L	L
Storage	Y	L	L	M
LT Storage	Y	L	L	M
ON	N	L	L	L
NCT	Y	L	L	M

3.11 Fuel Pellet Oxidation

3.11.1 Phenomenon Description

The exposure of UO_2 to oxidizing species can cause the formation of U_3O_8, according to Eq. 3-1.

$$UO_2 \rightarrow U_4O_9 \rightarrow U_3O_8 \qquad \text{Eq. 3-1}$$

U_3O_8 has a lattice volume 36% larger than UO_2, which results in fuel expansion [39]. If the increase in fuel volume is not accommodated by taking up gap space or compressing open porosity in the fuel, pellet cladding contact causes stress in the cladding, resulting in cladding strain. If sufficient strain occurs, the cladding splits at the defect locations, exposing more fuel to an oxidizing atmosphere, and possibly allowing fuel pellet particles (fines) to be released into the container.

The following concurrent conditions are required for fuel oxidation to occur:

- Availability of oxidants (e.g., O_2, H_2O_2)
- Presence of cladding defects that expose the fuel inside the rods to air
- High enough fuel temperature (>250 °C)

Oxygen can become available inside the cask under two scenarios: intrusion of air through the confinement boundary; and radiolysis of residual water that remains in the cask. Intrusion of air through the confinement boundary could potentially occur for a welded canister due to chloride induced stress corrosion cracking that progresses thru-wall. This would take decades to occur and the temperatures inside the canister would have decreased below the temperature needed for significant conversion to U_3O_8. Air could potentially enter the cask through a seal failure in a bolted cask; however, both the inner and outer seals would have to fail, which is extremely unlikely. Licensees implement maintenance activities and aging management programs to ensure cask seals and welded canisters continue to maintain confinement per the approved design basis of the system. The generation of oxygen from radiolysis of residual water is slow. Pre-existing cladding failure, such as a through-wall pinhole leak or hairline crack, can allow some level of air, water vapor, or water inside the rod.

While oxidation of UO_2 cannot be ruled out for the dry storage and transportation scenarios considered, it should be understood and recognized that any breach in the cladding would likely have occurred in the reactor creating a path between the interior of the affected rod and the primary coolant system while under reactor operating conditions. The fuel at the breach, having been at high temperature and exposed to water, would have already undergone some oxidation and be at some higher oxidation state, including possibly U_3O_8 and hydrated forms of various uranium oxides (such as schoepite, $[(UO_2)_4O(OH)_6]$ $6(H_2O)$).

Fuel oxidation is controlled by kinetics and availability of oxidants. When enough oxidants are available, fuel oxidation kinetics (which depend on temperature) influence the extent of the oxidation reaction. If the kinetics are fast, the reaction is controlled by the amount of oxidants present inside the cask. When temperatures are much higher than 250 °C, oxidation of the cladding will consume most of the oxidants, thereby mitigating any reactions between fuel and oxygen. For significant oxidation of fuel to U_3O_8, the fuel temperature would need to be high enough to form U_3O_8, but not so high that cladding oxidation consumes most of the oxidants.

3.11.2 Available Data and Quality

The following parameters influence fuel oxidation:

- Presence of fuel with breaches (e.g., pinhole leaks, hairline cracks)
- Availability of oxidants or precursors (e.g., oxygen, relative humidity, residual water)
- Temperature (kinetics is faster for higher temperatures but cladding consumes most of the oxidants if hot enough)
- Time for decomposition of water, containment failure of seals or welds to allow oxygen ingress, reactions to go to completion, and temperatures to decrease
- Burnup (with increasing burnup, the amount of porosity and grain boundary opening increases, while oxidation kinetics decrease)

Modern fuel has about 0.01% breached rods upon removal from a reactor, while early fuel was about 0.1% [17]. This yields one to nine rods with breaches per cask on average, assuming inspection does not detect them prior to loading. Therefore, as the presence of breached rods cannot be precluded, the possibility of fuel oxidation must be addressed.

Through-wall chloride-induced stress corrosion cracking (CISCC) or bolted closure seal leakage are unlikely to introduce enough air and humidity to cause much oxidation because an out-to-in differential pressure is negligible. Even if the cask was filled with air, the amount of oxygen could only oxidize one or two rods [17]. Radiolysis is neutron flux dependent – fuel with more decay heat has higher flux. Residual water amounts of 5.5 to 10 moles (100 to 180 ml) are reasonable, based on experience. Time constants for 99.99% of the water to be decomposed range from 4.8 to 72.2 years [73]. Therefore, if the source of oxidants is radiolysis, fuel oxidation would be constrained by the rate of dissociation of water.

Data in the literature indicate that exposure to oxidants at temperatures higher than 250°C is required to allow formation of measurable U_3O_8 for low burnup fuel [39]. Temperatures higher than 250°C occur early in storage life, when significant decay heat is available. However, as noted above, when temperatures are much higher than 250 °C, oxidation of the cladding will consume most of the oxidants, thereby mitigating any reactions between fuel and oxygen. For an air atmosphere case, fuel oxidation is maximized if peak cladding temperature is about 250 to 275 °C [17].

Fuel in a cask has a spatial distribution of temperatures that decline over time, as decay heat wanes. A uniformly loaded cask (i.e., all fuel assemblies have about the same decay heat) will have highest temperatures at its center, whereas a selectively loaded cask may have a more even temperature distribution. For a uniformly loaded cask, using 302 °C for "nominal" average temperature (which is likely an overestimate), 80% of the fuel is below the 250 °C threshold for oxidation to be a concern at the start of dry storage [73]. Temperatures are expected to be below 175 °C after 20 years [21]. For fuel in extended wet storage before loading, it is likely few rods ever exceed 250 °C (except for a short period of vacuum drying when oxidants are quickly removed).

The fuel oxidation reaction rate increases with temperature and quickly consumes free oxygen. Once oxygen is available, the fuel oxidation reaction is fast, 0.41 years at 250 °C and 9.5 hours at 335 °C [73]. Similar, though somewhat slower, oxidation times of about 50 hours at 350 °C have also been documented [39].

Moisture in the rods influences UO_2 oxidation products only if the relative humidity (RH) is above 40% [73]. For RH below this value, UO_2 oxidation was comparable to oxidation occurring in dry air conditions.

In addition to temperature, the reaction is strongly dependent on burnup, resisting formation of U_3O_8 at high burnup, in part because presence of chemical species formed during irradiation stabilize the fluorite crystalline lattice [35]. Evaluation of water driven oxidation for a variety of fuel types and burnups concluded that higher burnup fuel (i.e., 62 GWd/MTU) did not undergo fuel oxidation for the nominal temperature case and that over 35% of the cases considered had no oxidation [73].

Thus, fuel oxidation is only significant for breached rods, failure of the cask containment boundary (not expected), or the presence of residual water (some amount expected); while fuel temperature is within a fairly narrow temperature range of about 250°C to 275 °C. As the fuel temperatures are highest right after vacuum drying, any oxidants from residual water would most likely react with the cladding as soon as produced and no longer be available to oxidize the fuel.

References available include:

- Shukla, P., R. Sindelar, P-S. Lam. 2019. *Consequence Analysis of Residual Water in a Storage Canister- Preliminary Report*, SRNL-STI-2019-00495, Savannah River National Laboratory, September 2019.
- Electric Power Research Institute (EPRI). 1998. *Data Needs for Long-Term Dry Storage of LWR Fuel*, TR-108757, April 1998.
- International Atomic Energy Agency (IAEA). 2012. *Spent Fuel Performance Assessment and Research: Final Report of a Coordinated Research Project* (SPAR-II), IAEA-TECDOC-1680, Vienna, 2012.

- Hanson, B. D. R.C. Daniel, A.M. Casella, R.S. Wittman, W. Wu, P.J. MacFarlan, and R.W. Shimskey. 2008. Fuel-In-Air FY07 Summary, U.S. Department of Energy, PNNL-17275, Rev. 1, 2008.
- U.S. Nuclear Regulatory Commission (NRC). 2006. *Potential Rod Splitting Due to Exposure to an Oxidizing Atmosphere During Short-term Cask Loading Operations in LWR or Other Uranium Oxide Based Fuel,* ISG-22, May 8, 2006.

3.11.3 Ranking

The panel concluded the following:

- Fuel pellet oxidation is operable in all scenarios, provided that a cladding defect exists. If the fuel is unbreached, fuel oxidation is not considered operable.
- The knowledge of fuel oxidation is high, with medium confidence due to the uncertainties on the atmosphere and amount of oxidants in the cask.
- There were differing opinions on the significance of fuel oxidation. One opinion is that the significance is low for all scenarios, because any failed fuel would likely have failed in the reactor and would already be at some higher oxidation state. Any additional alterations of the fuel exposed to water after drying are negligible compared to the fuel rod alterations that already occurred as a result of in-reactor operations with a failure. Additionally, if the temperature is high (>275°C) available oxidants would favor oxidation of the cladding, and if temperatures are lower (<250 °C) it would be below the threshold for fuel oxidation. Further, the amount of oxidants available to react with the fuel is extremely limited. On the other hand, some experts felt that fuel oxidation significance is high for storage, ON and NCT because the temperature could be high enough and there is enough time for oxidants to develop (although the duration is relatively short for ON and NCT limiting the extent of oxidation), significance is medium for ST because the temperatures are higher but oxidants should be minimal during the draining and drying process, and significance is low for LT storage because the temperatures would be below the threshold for the oxidation reaction to U_3O_8.

The unzipping is expected to be limited to a single rod or a few individual rods, and it is not expected to lead to widespread cladding failures. A summary of the panel evaluation is presented in Table 3-12.

Table 3-12
Ranking Table for Fuel Oxidation

Phenomena/ Scenarios	Operable	Knowledge	Confidence	Significance
ST	Y	H	M	L or M
Storage	Y	H	M	L or H
LT Storage	Y	H	M	L
ON	Y	H	M	L or H
NCT	Y	H	M	L or H

3.12 Cladding Oxidation

3.12.1 Phenomenon Description

Zirconium, which is the main component of the cladding considered in this PIRT, reacts with oxygen or water forming zirconium oxide, according to Eqs. 3-2 and 3-3.

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2 \qquad \text{Eq. 3-2}$$

$$Zr + O_2 \rightarrow ZrO_2 \qquad \text{Eq. 3-3}$$

Cladding oxidation in dry storage can occur when a suitable corrosive environment is present in the cask (which is not a normal condition). However, it is known that some residual water remains in the dry storage canisters after drying. When fuel rod temperatures are high enough, water will oxidize the Zr-based cladding as well as other metals such as Al components and Al-B_4C cermets located inside the storage cask/canister; oxidation reactions tie the oxygen from the water to the Zr and Al when forming ZrO_2 or Al_2O_3. If fuel rod temperatures are too low for any significant reaction of the water with the metals, typically < 260 °C, radiolytic decomposition of the water dominates over time; radiolysis typically occurs over long periods of time (typically years or decades). Oxygen produced by radiolysis of the water can possibly slowly oxidize the cladding.

3.12.2 Available Data and Quality

The limiting parameters for cladding oxidation are:

- Presence of oxidizing species
- Temperature
- Time (e.g., exposure time to oxidative species, also related to time in reactor)

In Chapter 4 of [42], it is concluded that “Data from the integration model … indicate that additional cladding oxidation is insignificant compared to the original cladding oxide layer thickness generated in the reactor. Cladding oxidation caused by radiolysis-generated oxygen is not expected to be a concern because … the loss of cladding material from oxidation in the canister environment is expected to be less than 1 percent. Therefore, the formation of the ZrO_2 layer resulting from cladding oxidation is not expected to affect the strength of the cladding.”

It should be noted that 5.6 moles were measured in the High Burnup Demonstration cask at North Anna [28], while lower amounts have generally been assumed [83].

References available include:

- Jung, H., P. Shukla, T. Ahn, L. Tipton, K. Das, X. He, and D. Basu. 2013. *Extended Storage and Transportation: Evaluation of Drying Adequacy*, Center for Nuclear Waste Regulatory Analyses, San Antonio, Texas, June 2013.
- U.S. Nuclear Regulatory Commission (NRC). 2019. *Managing Aging Processes in Storage (MAPS) Report,* NUREG-2214, July 2019.
- Shukla, P., R. Sindelar, P-S. Lam. 2019. *Consequence Analysis of Residual Water in a Storage Canister- Preliminary Report*, SRNL-STI-2019-00495, Savannah River National Laboratory, September 2019.
- Electric Power Research Institute (EPRI). 1998. *Data Needs for Long-Term Dry Storage of LWR Fuel*, TR-108757, April 1998.

3.12.3 Ranking

The panel agreed upon the following:

- Cladding oxidation is operable in all conditions, except during long term storage, due to the low temperature in the cask.
- Knowledge of cladding oxidation and confidence in the data are high.
- Cladding oxidation significance is low.

Section 3.6.1.6 of NUREG-2214 [87] provides a technical basis (and supporting references) for the credibility of cladding oxidation to impact fuel performance during dry storage. The analysis assumed up to 1 liter of water in the confinement cavity and it concluded that the mechanism should not present problems if an adequate drying procedure is followed. The same conclusion is reasonable for transport. Therefore, cladding oxidation is not expected to compromise safety analyses, and a low significance was attributed by the panel. A summary of the panel evaluation is presented in Table 3-13.

Table 3-13
Ranking Table for Cladding Oxidation

Phenomena/ Scenarios	Operable	Knowledge	Confidence	Significance
ST	Y	H	H	L
Storage	Y	H	H	L
LT Storage	N	H	H	L
ON	Y	H	H	L
NCT	Y	H	H	L

Section 4: Synergistic Effects

This section summarizes the discussion on the interaction between the phenomena described in the previous sections. To capture the synergistic effects of the phenomena, a matrix was created to discuss the interaction of thermal creep, delayed hydride cracking, mechanical fatigue, hydride reorientation (HRO) and subsequent ductile to brittle transition (DtB), annealing, H_2 migration, mechanical overload, UO_2 oxidation, and cladding oxidation. Low-temperature creep, diffusion-controlled cavity growth and fatigue, were judged as substantially independent of other phenomena and not included in the synergistic effect's evaluation. Table 4-1 captures the effect of the phenomenon at the top of the column on the phenomenon in the row.

Three categories are provided to characterize the interaction:

- Positive: is used when the occurrence of A mitigates the effect of B towards cladding gross rupture
- Negative: is used when the occurrence of A accelerates or worsens the effect of B towards cladding gross rupture
- Neutral: is used when the occurrence of A does not influence the effect of B towards cladding gross rupture

Therefore, if the interaction is considered negative, the phenomenon on the row is more likely to damage the rod when the phenomenon at the top of the column is occurring at the same time. For example, mechanical overload is more likely to cause damage when paired with DHC or HRO/DtB transition, but less likely with annealing. As discussed in Section 2, the interactions assumed UO_2 fuel and 100 years storage.

The following sections explain the rationales used by the experts to justify the choices made to complete Table 4-1, which tallies the votes of the panel members (e.g., "Negative: 3" indicates three experts concluded the effect of the phenomenon shown in the column heading made the phenomenon shown in the row heading more severe). This section documents the panel's exercise to understand the interaction of various phenomena, but is not intended to indicate higher significance or the need to perform similar evaluations for specific analysis of used fuel.

Table 4-1
Matrix Tallying Expert Panelist Judgment of Synergistic Effects

Effect of → / On ↓	**T Creep**	**DHC**	**M Fatigue**	**HRO/DtB transition**	**Annealing**	**H_2 Migration**	**M Overload**	**Oxidation of UO_2**	**Cladding Ox – OD**
T creep		Positive: - Negative: - Neutral: 6	Positive: - Negative: - Neutral: 6	Positive: - Negative: - Neutral: 6	Positive: 2 Negative: 3 Neutral: 1	Positive: - Negative: 1 Neutral: 5	Positive: - Negative: - Neutral: 6	Positive: - Negative: 1 Neutral: 5	Positive: - Negative: 3 Neutral: 3
DHC	Positive: 4 Negative: - Neutral: 2		Positive: - Negative: 3 Neutral: 3	Positive: - Negative: 4 Neutral: 2	Positive: 4 Negative: - Neutral: 2	Positive: - Negative: 2 Neutral: 4	Positive: - Negative: 6 Neutral: -	Positive: - Negative: 2 Neutral: 4	Positive: - Negative: 4 Neutral: 2
M fatigue	Positive: - Negative: 3 Neutral: 3	Positive: - Negative: 6 Neutral: -		Positive: - Negative: 1 Neutral: 5	Positive: - Negative: - Neutral: 6	Positive: - Negative: - Neutral: 6	Positive: - Negative: 2 Neutral: 4	Positive: - Negative: 2 Neutral: 4	Positive: - Negative: 3 Neutral: 3
HRO/DtB transition	Positive: 5 Negative: - Neutral: 1	Positive: - Negative: 2 Neutral: 4	Positive: - Negative: - Neutral: 6		Positive: 4 Negative: - Neutral: 2	Positive: - Negative: 1 Neutral: 5	Positive: - Negative: - Neutral: 6	Positive: - Negative: 2 Neutral: 4	Positive: - Negative: 4 Neutral: 2
Annealing	Positive: - Negative: - Neutral: 6	Positive: - Negative: - Neutral: 6	Positive: - Negative: - Neutral: 6	Positive: - Negative: - Neutral: 6		Positive: - Negative: - Neutral: 6	Positive: - Negative: - Neutral: 6	Positive: - Negative: - Neutral: 6	Positive: - Negative: - Neutral: 6

Table 4-1 (continued)
Matrix Tallying Expert Panelist Judgment of Synergistic Effects

Effect of → / On ↓	T Creep	DHC	M Fatigue	HRO/DtB transition	Annealing	H_2 Migration	M Overload	Oxidation of UO_2	Cladding Ox – OD
H_2 migration	Positive: - Negative: - Neutral: 6	Positive: - Negative: - Neutral: 6	Positive: - Negative: - Neutral: 6	Positive: - Negative: - Neutral: 6	Positive: - Negative: 1 Neutral: 5		Positive: - Negative: 1 Neutral: 5	Positive: - Negative: - Neutral: 6	Positive: - Negative: 1 Neutral: 5
M Overload	Positive: 3 Negative: 2 Neutral: 1	Positive: - Negative: 6 Neutral: -	Positive: - Negative: 2 Neutral: 4	Positive: - Negative: 6 Neutral: -	Positive: 6 Negative: - Neutral: -	Positive: - Negative: - Neutral: 6		Positive: - Negative: 1 Neutral: 5	Positive: - Negative: 4 Neutral: 2
Oxidation of UO_2	Positive: - Negative: 1 Neutral: 5	Positive: - Negative: 2 Neutral: 4	Positive: - Negative: - Neutral: 6	Positive: - Negative: 1 Neutral: 5	Positive: 1 Negative: - Neutral: 5	Positive: - Negative: - Neutral: 6	Positive: - Negative: 2 Neutral: 4		Positive: - Negative: 1 Neutral: 5
Cladding Ox – OD	Positive: - Negative: 1 Neutral: 5	Positive: - Negative: - Neutral: 6	Positive: - Negative: - Neutral: 6	Positive: - Negative: - Neutral: 6	Positive: - Negative: - Neutral: 6	Positive: - Negative: - Neutral: 6	Positive: - Negative: - Neutral: 6	Positive: 1 Negative: - Neutral: 5	

4.1 Impact of Thermal Creep

This section describes the impact of thermal creep on the other phenomena considered (i.e., DHC, mechanical fatigue, HRO/DtB, annealing, hydrogen migration, mechanical overload, fuel oxidation, and cladding oxidation). When occurring, thermal creep results in an increase of the void volume inside a fuel rod, and in a corresponding decrease of the rod internal pressure and hoop stress. Given the excellent gas communication observed by ORNL and PNNL in the sibling pins [72] [51], this pressure decrease will apply to the whole rod. The increase of the void volume corresponds to a thinning of the cladding wall, although this last phenomenon usually has a lower impact on the overall behavior of the rod.

A summary of the synergistic effects analyzed in this chapter is summarized in the figure below.

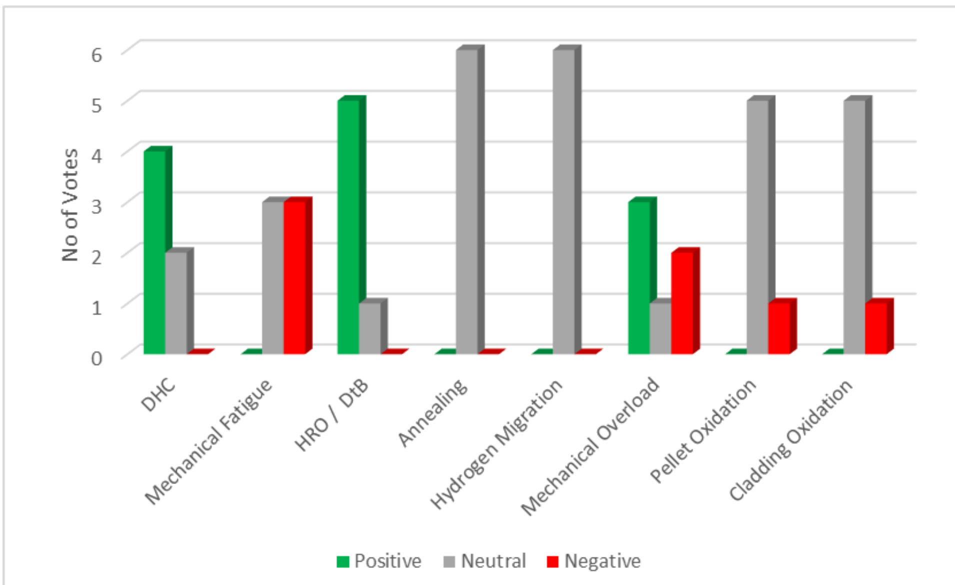

Figure 4-1
Impact of Thermal Creep on Other Phenomena

4.1.1 Impact of Thermal Creep on DHC

The presence of thermal creep has competing effects on DHC:

- Thermal creep occurring during the early period of dry storage when the temperature is high, is expected to result in internal volume expansion, which in turn would reduce the rod internal pressure and correspondingly reduce hoop stress. By the time the temperature drops to the DHC activation level (below 200°C), the stress intensity factor becomes too small to activate DHC, according to the K_{IH}. Hence, the effect might be positive because creep eliminates DHC threats to cladding failure.

- Thermal creep may also lead to cladding thinning, which would also lower the hoop stresses needed to initiate DHC (i.e., reduce the critical flaw size for DHC initiation relative to the cladding thickness). Vulnerability to DHC depends on the local stress field, with a higher vulnerability related to higher stress. Thus, combined DHC and thermal creep might have a higher associated risk of cladding failure. Because this effect is expected to be less significant than the volume expansion, it resulted in neutral votes.

4.1.2 Impact of Thermal Creep on Mechanical Fatigue

Potential cladding thermal creep at initial (higher) temperatures of dry storage may cause cladding thinning, which results in a lower fatigue lifetime given the same loads (e.g., vibrations during transport), reducing the fatigue life of the cladding. Additionally, thermal creep could cause breaking of the fuel-cladding bond or a weakened fuel region underneath the bond, which would result in less fuel support for the cladding (negative effect).

On the other side, reduction of the mechanical bonding effect reduces the flexural rigidity, which in turn increases the amplitude of any vibrations induced (given the same load input). Thus, the fatigue lifetime would be reduced by thermal creep. Results from the multi-modal transportation test [37] show that external loads (strains and accelerations) during handling and normal conditions of transport are very low given the low bending moments/strains typical of normal conditions of transport, the impact of thermal creep on fatigue is likely negligible (neutral effect).

4.1.3 Impact of Thermal Creep on HRO/DtB Transition

This phenomenon is not considered relevant without a temperature high enough to dissolve the hydrides. Potential cladding thermal creep at initial (higher) temperatures of dry storage is expected to result in internal volume expansion, which in turn would reduce the rod internal pressure and the resulting cladding hoop stress for HRO. The overall effect is likely positive.

4.1.4 Impact of Thermal Creep on Annealing

Thermal annealing is a function of temperature and the annealing rate depends on time at temperature, fabricated microstructure, and hydrogen content. These parameters are not expected to be affected by creep out of the cladding, because the temperatures are not high enough. Therefore, the overall effect of creep on annealing is considered neutral.

4.1.5 Impact of Thermal Creep on Hydrogen Migration

Axial hydrogen migration is dependent on the diffusion coefficient in the cladding. The coefficient is not expected to be dependent on creep out of the cladding. The overall effect is neutral.

4.1.6 Impact of Thermal Creep on Mechanical Overload

The effect of creep on mechanical overload implies more competing effects that do not allow a univocal judgment. On the one hand, diametral creep strains create larger fuel-cladding gaps and increase rod void volume that would be beneficial in accommodating fuel swelling resulting from alpha-decay damage (positive effect).

On the other hand, creep can cause a thinning of the cladding, which increases the cladding stress given the same applied load and decreases the strain capacity. A thinner wall would result in easier failure due to mechanical overload/pellet swelling (negative effect).

Thermal creep is a long-term mechanism that would be ameliorated by the diffusion and release of helium from the pellet to the free volume of the rod. Due to the long timing, laboratory observation is not practical. Therefore, a safety assessment of the implications of thermal creep on mechanical overload is based on a high degree of conservative assumption, which justifies the choice of neutral synergistic effects.

4.1.7 Impact of Thermal Creep on Oxidation of UO_2

Pellet oxidation is dependent on the amount of residual oxidants (at the time of SNF loading) and other oxidants generated by radiolysis of residual water. Since thermal creep does not affect these parameters, the overall effect is expected to be neutral.

However, for oxidation to occur, there should already be a through-wall failure to allow air or water to be in contact with fuel pellets. A thinned cladding due to the presence of creep will fail and split sooner, due to the lower amount of bulk oxidation required. Therefore, the overall effect might be negative.

4.1.8 Impact of Thermal Creep on Cladding Oxidation

Cladding oxidation is dependent on the amount of residual oxidants (at the time of SNF loading) and other oxidants generated by radiolysis of residual water. Therefore, thermal creep is not expected to affect these parameters.

However, thinner cladding is expected to oxidize through the cladding thickness faster due to less material, and therefore fail sooner, causing an overall negative effect.

4.2 Impact of DHC

This section describes the impact of DHC on the other phenomena considered (i.e., thermal creep, mechanical fatigue, HRO/DtB, annealing, hydrogen migration, mechanical overload, fuel oxidation, and cladding oxidation). As visible from the summary presented in the figure below, the synergistic effect is either neutral or negative.

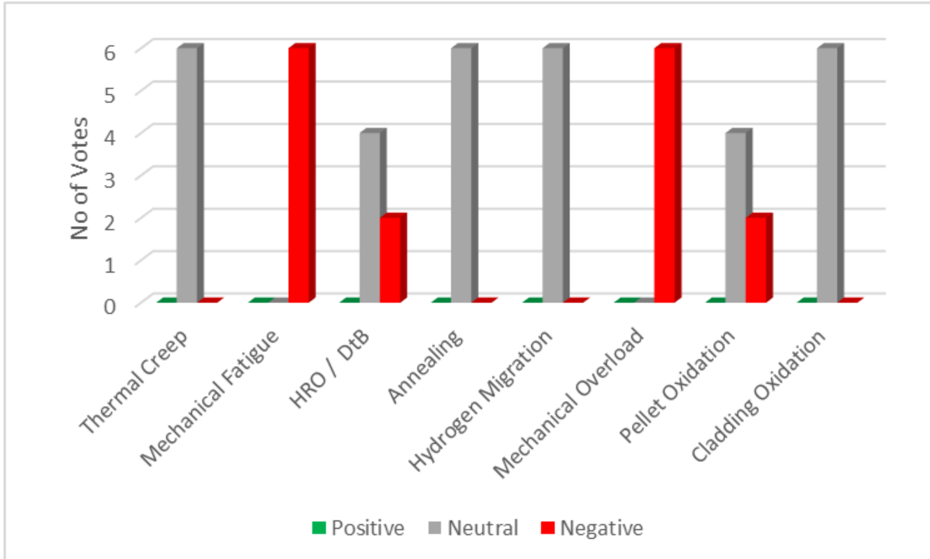

Figure 4-2
Impact of DHC on Other Phenomena

4.2.1 Impact of DHC on Thermal Creep

When the temperature is high enough to activate creep in dry storage, DHC, being a low-temperature phenomenon, would not be active. At lower temperatures where DHC would be likely to occur, the stress is too low to activate DHC; therefore, DHC is not expected to affect the creep strain capacity of the fuel cladding. The effect is neutral.

4.2.2 Impact of DHC on Mechanical Fatigue

Because dry storage thermal history can promote DHC, this is likely to reduce the bending and hoop strength of the cladding, which would in turn reduce the bending fatigue resistance to mechanical loads during transportation subsequent to dry storage. Therefore, the presence of DHC cracks can cause early mechanical fatigue related failures, with a negative synergistic effect.

4.2.3 Impact of DHC on HRO/DtB Transition

DHC is not expected to affect the susceptibility to hydride reorientation of the cladding, because by the time DHC occurs, the temperatures and hoop stresses are low and most of the hydrides will have reprecipitated and not subject to HRO. This consideration leads to the conclusion that the effect is neutral.

On the other hand, the presence of DHC cracks can cause hydrogen to flow towards the crack tip, which is the mechanism by which DHC evolves. This effect can make the effect of HRO worse because a hydride might favor the start of a crack.

4.2.4 Impact of DHC on Annealing

Annealing is no longer active for temperatures lower than 250 °C, when DHC is active. Additionally, thermal annealing is a function of temperature and it is not expected to be affected by DHC. Therefore, the phenomena are unrelated and the effect of DHC on annealing is evaluated neutral.

4.2.5 Impact of DHC on Hydrogen Migration

Axial hydrogen migration is dependent on the hydrogen diffusion coefficient in the cladding. The coefficient is not expected to be dependent on DHC. Additionally, by the time DHC can occur, the temperatures are below those for substantial H_2 migration. The overall effect is neutral.

4.2.6 Impact of DHC on Mechanical Overload

DHC would be expected to have negative consequences with regard to mechanical overload given the locally higher tensile stresses in the cladding due to DHC cracks: an incipient flaw or crack from DHC would be more susceptible to failure due to mechanical overload. Any DHC cracks are expected to reduce the long-term strain capacity available for mechanical overload. The effect of DHC on mechanical overload is evaluated negative.

4.2.7 Impact of DHC on Oxidation of UO_2

The two phenomena are unrelated unless DHC causes cladding rupture, which would allow moisture to enter fuel rod.

A negative effect could occur considering that due to an incipient through wall flaw or crack from DHC, the cladding will fail and split sooner (less bulk oxidation required) and could increase the probability of cladding unzipping.

4.2.8 Impact of DHC on Cladding Oxidation

Cladding oxidation is dependent on the amount of residual oxidants (at the time of SNF loading) and other oxidants generated by radiolysis of residual water. DHC is not expected to affect these parameters. Moreover, the additional hydrogen from cladding oxidation would be too small to affect DHC. Therefore, a neutral effect of DHC on cladding oxidation is expected.

4.3 Impact of Mechanical Fatigue

This section describes the impact of mechanical fatigue on the other phenomena considered (i.e., thermal creep, DHC, HRO/DtB, annealing, hydrogen migration, mechanical overload, fuel oxidation, and cladding oxidation). As visible from the summary presented in the figure below, the synergistic effect is neutral most of the time with some potential negative effects. The interaction between mechanical fatigue and the other phenomena is often considered neutral because mechanical fatigue occurs during post-storage transport while other phenomena occur during storage.

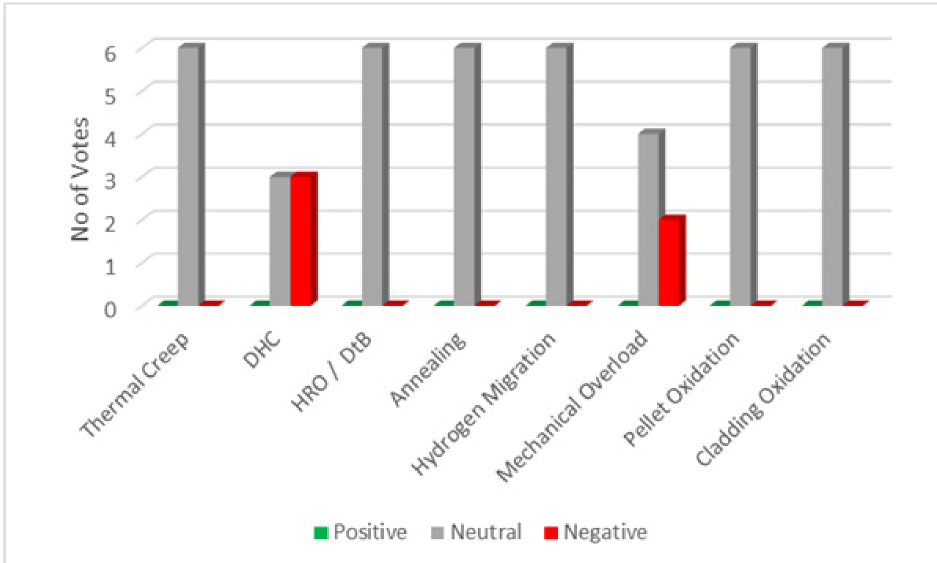

Figure 4-3
Impact of Mechanical Fatigue on Other Phenomena

4.3.1 Impact of Mechanical Fatigue on Thermal Creep

Thermal creep is a long-term phenomenon occurring during dry storage, but not during transportation due to limited timeframes. Mechanical fatigue is expected during transportation, but not expected during short-term operations, normal, and off-normal conditions of storage. Therefore, mechanical fatigue during normal conditions of transport is not expected to affect thermal creep, and a neutral effect of mechanical fatigue on thermal creep is expected.

4.3.2 Impact of Mechanical Fatigue on DHC

The interaction between mechanical fatigue and DHC might be negative because mechanical fatigue loads during transport can nucleate and grow partial or through-wall cracks. If the fuel is stored after transport, the crack may be a large enough incipient crack to propagate due to DHC, and mechanical fatigue can supply additional stress risers to facilitate DHC.

On the other hand, DHC is a long-term phenomenon only applicable during dry storage, as timeframes for transportation are limited. Mechanical fatigue is not expected during short-term operations, normal and off-normal conditions of storage. Therefore, mechanical fatigue during normal conditions of transport might not impact DHC.

4.3.3 Impact of Mechanical Fatigue on HRO/DtB Transition

As HRO requires a temperature excursion, an inherently short-term transient, but mechanical fatigue accumulates over a long period of time, mechanical fatigue and HRO are not expected to occur at the same time. In addition, while mechanical fatigue can supply additional local stress risers, the stress is very short term and would not impact HRO. Therefore, a neutral interaction is expected.

4.3.4 Impact of Mechanical Fatigue on Annealing

Thermal annealing is a function of temperature and it is not expected to be affected by mechanical fatigue.

4.3.5 Impact of Mechanical Fatigue on Hydrogen Migration

Axial hydrogen migration is dependent on the diffusion coefficient in the cladding. The coefficient is not expected to be dependent on mechanical fatigue associated with normal conditions of transport. Therefore, the effect of mechanical fatigue on hydrogen migration was evaluated neutral.

4.3.6 Impact of Mechanical Fatigue on Mechanical Overload

Mechanical overload, due to the generation and retention of decay gases causing pellet expansion, is not expected to be affected by mechanical fatigue.

However, the additional local stress risers caused by mechanical fatigue might reduce the strain capacity and facilitate failure from mechanical overload, causing a negative effect.

4.3.7 Impact of Mechanical Fatigue on Oxidation of UO_2

Pellet oxidation is dependent on the amount of residual oxidants (at the time of SNF loading) and other oxidants generated by radiolysis of residual water. Mechanical fatigue is not expected to affect these parameters. Therefore, the effect of mechanical fatigue on pellet oxidation was evaluated neutral.

4.3.8 Impact of Mechanical Fatigue on Cladding Oxidation

Cladding oxidation is dependent on the amount of residual oxidants (at the time of SNF loading) and other oxidants generated by radiolysis of residual water. Mechanical fatigue is not expected to affect these parameters. Therefore, the effect of mechanical fatigue on cladding oxidation was evaluated neutral.

4.4 Impact of HRO/DtB transition

This section describes the impact of hydride reorientation and ductile to brittle transition (HRO/DtB) on the other phenomena considered (i.e., thermal creep, DHC, mechanical fatigue, annealing, hydrogen migration, mechanical overload, fuel oxidation, and cladding oxidation). HRO/DtB occurs if initial temperatures and pressure/hoop stress are high enough, whereas hydride reprecipitation occurs at lower temperatures. If reprecipitation occurs with long enough radial hydrides, the weaker hydrides may increase the likelihood of a crack, and the cladding becomes less ductile and more susceptible to failure under pinch loads (as long as the fracture toughness of the hydride is lower than the fracture toughness of the cladding). As visible from the summary presented in the figure below, the synergistic effect is either neutral (because phenomena are unrelated or because HRO/DtB does not embrittle cladding until temperatures get low enough late in storage) or negative.

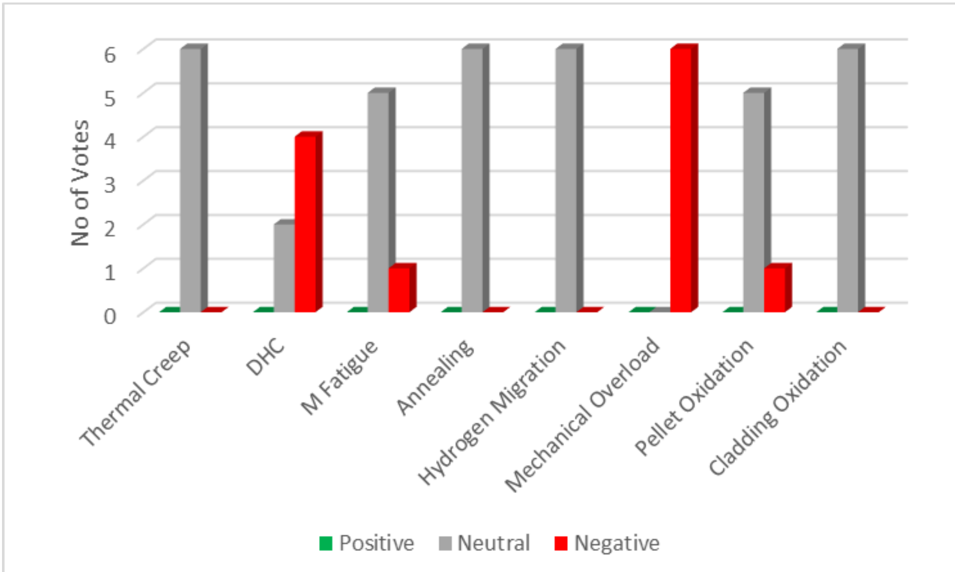

Figure 4-4
Impact of HRO/DtB on Other Phenomena

4.4.1 Impact of HRO/DtB on Thermal Creep

Hydrogen content and hydrides generally affect thermal creep rates. However, creep rates are not sensitive to the orientation of the hydrides, based on creep experiments performed after applying a reorientation heat treatment to the cladding segments that were subsequently creep tested [39].

Hydride reorientation is not expected to impact the creep strain capacity of the cladding. NRC-sponsored testing [86] on Zircaloy-4-clad HBU SNF rod segments suggests that the cladding mechanical properties remain unaffected by hydride reorientation (i.e., axial bending at high strain rates). However, confirmatory data on hoop strain capacity is needed. At this point, it is reasonable to conclude that the impact of HRO on thermal creep is neutral, subject to confirmation.

4.4.2 Impact of HRO/DtB on DHC

HRO and DHC might be unrelated or there could be a negative synergistic effect. A negative effect might be due to the presence of radial hydrides that can increase the possibility of starting cracks and facilitate DHC at the tip (though in all likelihood the K_{IH} will be too small to facilitate DHC). This would increase DHC's negative impact on cladding integrity and decrease its fracture resistance.

4.4.3 Impact of HRO/DtB on Mechanical Fatigue

NRC-sponsored testing [86] on Zircaloy-4-clad HBU SNF rod segments suggests that the cladding mechanical properties remain unaffected by hydride reorientation (i.e., axial bending at high strain rates and fatigue endurance). A confirmation is needed that the same performance is observed in the other zirconium alloy cladding.

A negative effect can be due to radial hydrides increasing the propensity for starter cracks for mechanical fatigue, accelerating fatigue failure.

4.4.4 Impact of HRO/DtB on Annealing

The presence of hydride precipitates does not affect the annealing temperature or rate. Therefore, annealing is not expected to be affected by hydride reorientation, and the impact of HRO on annealing was evaluated neutral.

4.4.5 Impact of HRO/DtB on Hydrogen Migration

Hydrogen migration refers to the diffusion of the hydrogen that is in solid solution and is dependent on the diffusion coefficient in the cladding. The coefficient is not expected to be dependent on hydride reorientation, which would be limited at the end of the rods due to lower hydrogen contents and the effect of HRO/DtB transition on hydrogen migration was evaluated neutral.

4.4.6 Impact of HRO/DtB on Mechanical Overload

Mechanical overload would be the phenomenon most negatively impacted by HRO, especially late in storage when the temperatures are well below DtB transition temperature. Hydrides that were reoriented and are located at the pellet ends may act as a weak point and produce local cladding vulnerability, facilitating failure when subjected to mechanical overload. Therefore, the effect of HRO/DtB transition on mechanical was evaluated negative.

4.4.7 Impact of HRO/DtB on Oxidation of UO_2

Pellet oxidation is dependent on the amount of residual oxidants (at the time of SNF loading) and other oxidants generated by radiolysis of residual water. Hydride reorientation is not expected to affect these parameters. Therefore, the effect of HRO/DtB transition on fuel oxidation was evaluated likely neutral.

However, because of the higher cladding hoop stress caused by pellet-cladding contact, the combined effect of radial hydrides and the contact force induced by the oxidation of UO_2 might have a negative effect on cladding integrity.

4.4.8 Impact of HRO/DtB on Cladding Oxidation

Cladding oxidation is dependent on the amount of residual oxidants (at the time of SNF loading) and other oxidants generated by radiolysis of residual water. Hydride reorientation is not expected to affect these parameters. Therefore, the effect of HRO/DtB transition on cladding oxidation was evaluated neutral.

4.5 Impact of Annealing

This section describes the impact of annealing on the other phenomena considered (i.e., thermal creep, DHC, mechanical fatigue, HRO/DtB, hydrogen migration, mechanical overload, fuel oxidation, and cladding oxidation). Thermal annealing is expected to increase ductility and reduce strength, restoring the values of these properties close to unirradiated. Annealing of radiation defects that contain hydrogen may release that hydrogen into the cladding.

A summary of the synergistic effects analyzed in this chapter is summarized in the figure below.

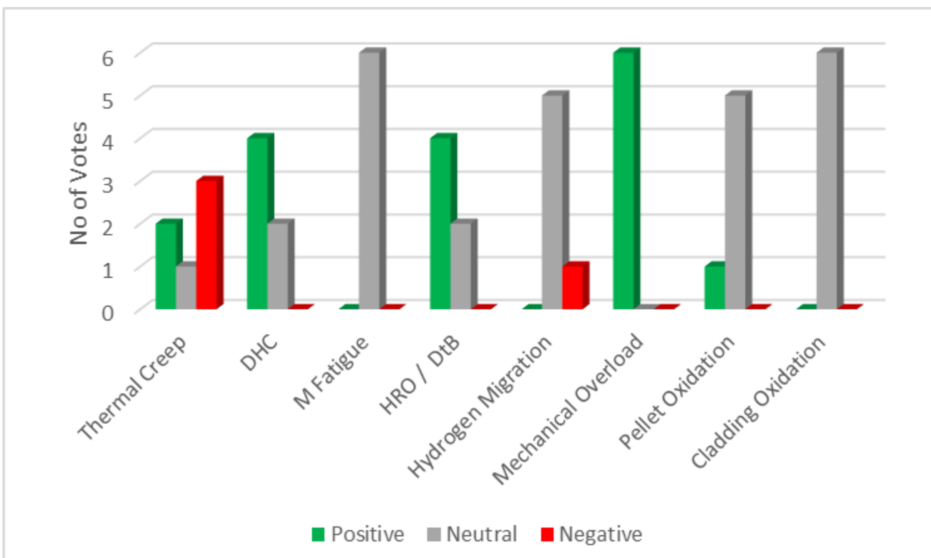

Figure 4-5
Impact of Annealing on Other Phenomena

4.5.1 Impact of Annealing on Thermal Creep

The effect of annealing on thermal creep is characterized by more competing effects that do not allow a complete judgment. The two phenomena both occur at high temperature, and therefore occur simultaneously. At the typical dry storage temperature immediately following vacuum drying, annealing increases the creep rate and accelerates stress relaxation caused by cladding expansion and reduction of internal rod pressure, which are positive effects. On the other hand, annealing would increase ductility which could facilitate additional thermal creep, reducing creep resistance.

Additionally, the effect of annealing on thermal creep depends on the acceptance criterion: assuming regulatory acceptance criterion based on a figure-of-merit either for maximum diametral strain or minimum yield stress, the effect would be negative. Assuming an acceptance criterion based on a figure-of-merit for creep rupture, the effect would likely be positive.

4.5.2 Impact of Annealing on DHC

Annealing is inactive during the low temperature regime of DHC, and would have already occurred when the temperature reaches the DHC activation level (~200°C).

According to studies performed on unirradiated and irradiated materials, the stress intensity factor required for DHC (K_{IH}) is generally higher for unirradiated material. Therefore, annealing is expected to be beneficial in minimizing the potential for DHC, although this still requires confirmation. Based on engineering judgment, the increase in ductility due to annealing might outweigh the decrease in strength caused by DHC and the additional dissolved hydrogen available for precipitation. The effect of annealing on DHC is, therefore, evaluated positive or neutral.

4.5.3 Impact of Annealing on Mechanical Fatigue

The change in strength and ductility due to annealing can have competing effects, but with the realistic loads expected during transportation, annealing should have no effect on mechanical fatigue (neutral effect). However, a confirmation could be needed on whether thermal annealing could improve the bending fatigue resistance of the cladding.

4.5.4 Impact of Annealing on HRO/DtB Transition

Annealing causes softening of the cladding alloy matrix (decrease in strength, increase in ductility), and increases the amount of hydrogen available for precipitation. Although hydride reorientation would occur to a greater extent due to hydride dissolution and lowering of reorientation threshold stresses, the softer alloy matrix would largely mitigate the effects of radial hydrides.

Based on engineering judgment, the increase in ductility might outweigh the decrease in strength and the additional dissolved hydrogen available for precipitation. However, it should be confirmed that the mechanical performance of the cladding remains unchanged due to radial hydrides under hoop-loads at low strain rates. The overall impact of annealing on HRO is considered positive or neutral.

4.5.5 Impact of Annealing on Hydrogen Migration

Axial hydrogen migration is dependent on the diffusion coefficient in the cladding, and it is not affected by irradiation. Therefore, it is expected that the impact of annealing of irradiation damage is neutral on hydrogen migration.

The effect of annealing on hydrogen migration might be negative considering that annealing would result in more hydrogen in solution available for migration.

4.5.6 Impact of Annealing on Mechanical Overload

Thermal annealing is expected to increase cladding ductility. In the low temperature range, when swelling due to alpha radiation damage and, possibly, helium accumulation are considered, the increase in ductility due to annealing would increase the cladding strain capacity and mitigate the effects of mechanical overload.

4.5.7 Impact of Annealing on Oxidation of UO_2

Pellet oxidation is dependent on the amount of residual oxidants (at the time of SNF loading) and other oxidants generated by radiolysis of residual water. Thermal annealing is not expected to affect these parameters. Therefore, the interaction between annealing and fuel oxidation is likely neutral because the phenomena are unrelated.

On the other hand, the greater ductility in annealed cladding could allow for greater oxidation of UO_2 before splitting occurs. Therefore, annealing might also have a positive impact on fuel pellet oxidation.

4.5.8 Impact of Annealing on Cladding Oxidation

Cladding oxidation is dependent on the amount of residual oxidants (at the time of SNF loading) and other oxidants generated by radiolysis of residual water. Thermal annealing is not expected to affect these parameters. Therefore, the interaction between annealing and cladding oxidation is likely neutral because phenomena are unrelated.

4.6 Impact of Hydrogen Migration

This section describes the impact of hydrogen migration on the other phenomena considered (i.e., thermal creep, DHC, mechanical fatigue, HRO/DtB, annealing, mechanical overload, fuel oxidation, and cladding oxidation).

As shown in the figure below, the effect of hydrogen migration on the other phenomena is mainly neutral. This can be explained considering that hydrogen migration is driven by temperature gradient (from hot to cold) and tensile stress gradient (low to high), which are negligible in the scenarios considered. The temperature distribution in the cladding is nearly uniform radially and the gradient is small axially. Also, the hoop stress radial gradient and magnitude are both negligible. On the negative side, cladding affected by hydrogen migration may have decreased ductility and be embrittled, which could have a negative impact on some of the phenomena considered. More testing of these cladding regions is needed to generate data that could be used to estimate the synergistic effects.

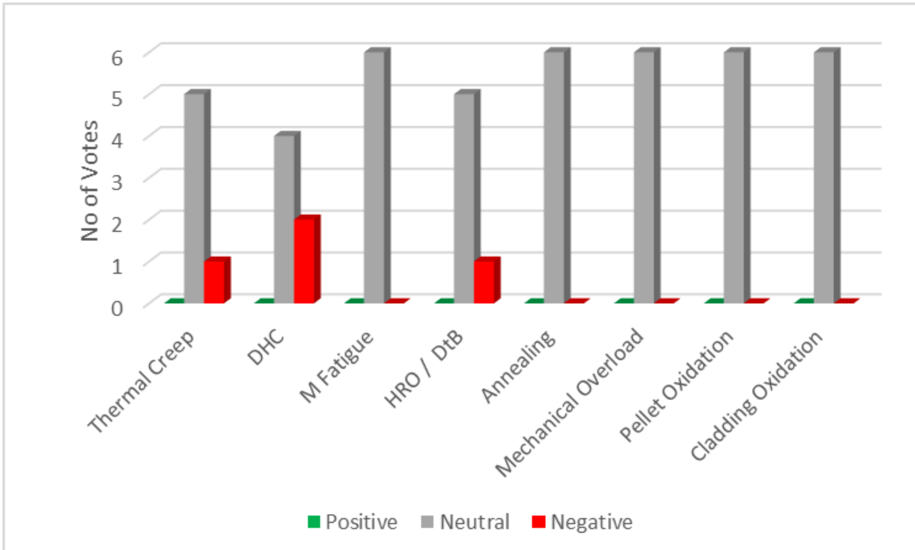

Figure 4-6
Impact of Hydrogen Migration on Other Phenomena

4.6.1 Impact of Hydrogen Migration on Thermal Creep

Hydrogen migration at axial-end locations of the rod is not expected to impact the creep strain capacity of the cladding, therefore the impact of hydrogen migration on thermal creep is likely neutral. However, migration of hydrogen from the hot areas (where creep is more likely) to the cooler areas could increase thermal creep, which would result in a negative effect.

4.6.2 Impact of Hydrogen Migration on DHC

The impact of hydrogen migration on DHC is likely neutral, due to the small gradients in the areas considered. However, hydrogen migration to axial-end locations of the rod (cooler parts of the cladding) may increase the flaw size and lower the stress intensity factor for DHC, causing a negative impact of hydrogen migration on DHC.

4.6.3 Impact of Hydrogen Migration on Mechanical Fatigue

The total amount of hydrogen generally affects the strength and ductility of the cladding. Tests have shown little variability over a wide range of total hydrogen and the external loads demonstrated in the transportation tests are too low to be of concern [37]. Therefore, hydrogen migration is not expected to meaningfully impact the bending fatigue resistance.

4.6.4 Impact of Hydrogen Migration on HRO/DtB Transition

Hydride reorientation would be limited at the end of the rods due to lower hydrogen contents. Therefore, hydrogen migration is not expected to meaningfully increase the extent of hydride reorientation.

However, significant migration to cooler areas of the cladding (such as at pellet-pellet interfaces) could result in more HRO, which could allow a greater radial hydride continuity factor. This would have a negative effect on the HRO phenomenon.

4.6.5 Impact of Hydrogen Migration on Annealing

Thermal annealing is a function of temperature and not expected to be affected by hydrogen migration. Therefore, the effect of hydrogen migration on annealing was rated neutral.

4.6.6 Impact of Hydrogen Migration on Mechanical Overload

Mechanical overload due to the generation and retention of decay gases, which could cause pellet expansion, is not expected to be affected by hydrogen migration at the axial ends of the rods. Therefore, the effect of hydrogen migration on mechanical overload was rated neutral.

4.6.7 Impact of Hydrogen Migration on Oxidation of UO_2

Pellet oxidation is dependent on the amount of residual oxidants (at the time of SNF loading) and other oxidants generated by radiolysis of residual water. Significant hydrogen migration might occur from the hot areas, where UO_2 oxidation is faster, resulting in potential larger clad opening when unzipping occurs. However, the amount of hydrogen migration to cause this effect is not expected and has not been observed. Therefore, it is not expected that hydrogen migration affects fuel oxidation, and the effect of hydrogen migration on fuel oxidation was rated neutral.

4.6.8 Impact of Hydrogen Migration on Cladding Oxidation

Cladding oxidation is dependent on the amount of residual oxidants at the time of SNF loading, and other oxidants generated by radiolysis of residual water. Hydrogen migration is not expected to affect these parameters. Therefore, the effect of hydrogen migration on cladding oxidation was rated neutral.

4.7 Impact of Mechanical Overload

This section describes the impact of mechanical overload on the other phenomena considered (i.e., thermal creep, DHC, mechanical fatigue, HRO/DtB, annealing, hydrogen migration, fuel oxidation, and cladding oxidation).

Mechanical overload discussed in this section is related to swelling of the pellets due to the helium released from alpha decay (see Section 3.10.1) and pellet material composition changes. It is expected that helium is retained by the pellet, but some may be released to the rod void volume. The fuel swelling applies a load to the cladding, but it is expected to occur very late in storage life.

As seen in the figure below, mechanical overload has negative or no impact on other phenomena analyzed because helium release from alpha decay is small enough to be easily accommodated by the void volume expansion caused by thermal creep. More details are provided below.

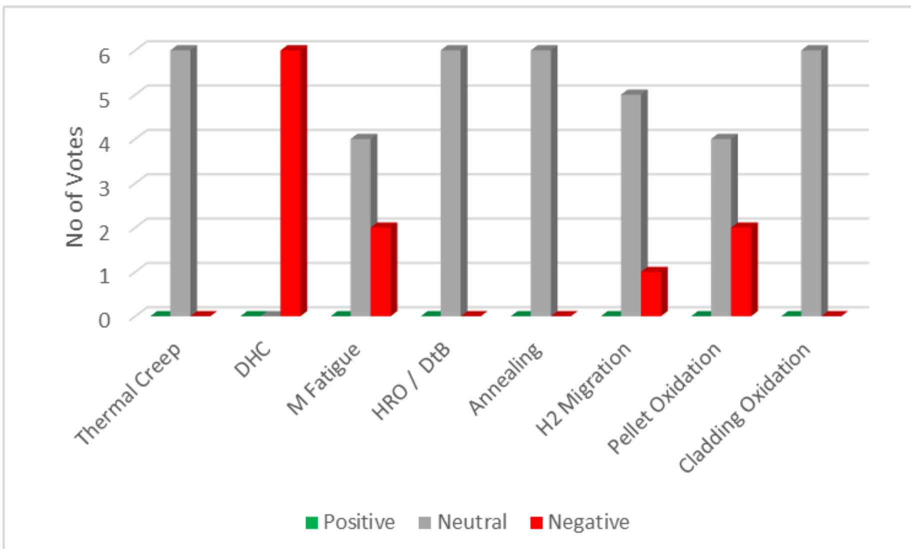

Figure 4-7
Impact of Mechanical Overload on Other Phenomena

4.7.1 Impact of Mechanical Overload on Thermal Creep

Mechanical overload is expected to occur late in storage life, when the temperatures are too low for thermal creep, and when thermal creep has already occurred. For this reason, the effect of mechanical overload on thermal creep is considered neutral.

4.7.2 Impact of Mechanical Overload on DHC

Mechanical overload due to the generation and retention of decay gases, which could cause pellet expansion, is likely to result in higher hoop stresses. These stresses could potentially exceed the critical stress intensity and facilitate DHC. Although a more complete understanding of the synergism is still required, the overall effect of mechanical overload on DHC is expected to be negative.

4.7.3 Impact of Mechanical Overload on Mechanical Fatigue

The effect of mechanical overload on mechanical fatigue depends on the magnitude of mechanical overload. A small mechanical overload (e.g., with cladding deformation within the elastic range) would improve fuel-cladding mechanical contact and increase bending cycles to failure, whereas a large mechanical overload would induce plastic strains in the cladding and decrease bending cycles to failure.

The effect of mechanical overload on mechanical fatigue might be neutral if the loads involved are low enough not to be a concern. On the other hand, the pellet swelling at the pellet-pellet interface increases the local hoop stress where rods are known to fracture in bending fatigue tests, having a negative effect on mechanical fatigue. A better understanding of the impact of higher hoop stresses due to mechanical overload on the bending fatigue resistance of the cladding is needed.

4.7.4 Impact of Mechanical Overload on HRO /DtB Transition

Mechanical overload would occur potentially late in storage, when the temperature of the fuel is likely below 200°C and the solubility of hydrogen is negligible (hydrides will already have re-precipitated and not subjected to reorientation). Therefore, reorientation of the hydrides caused by higher hoop stresses due to mechanical overload is unlikely and the interaction between the two phenomena is neutral.

4.7.5 Impact of Mechanical Overload on Annealing

Thermal annealing is a function of temperature and not expected to be affected by mechanical overload. Additionally, mechanical overload is expected to occur late in storage, when the temperature is too low for annealing to occur. Therefore, the interaction between the two phenomena is neutral.

4.7.6 Impact of Mechanical Overload on Hydrogen Migration

The effect of mechanical overload on hydrogen migration might be negative considering that hydrogen is known to migrate along stress gradients, which would be enhanced locally by mechanical overload, potentially causing a negative synergistic effect. On the other hand, the effect might be neutral because mechanical overload occurs late in storage, after hydrogen migration has already occurred.

4.7.7 Impact of Mechanical Overload on Oxidation of UO_2

The effect of mechanical overload on UO_2 oxidation is likely neutral because they occur at two different times during the storage (mechanical overload occurs late in storage after UO_2 oxidation has already occurred), and mechanical overload is not expected to affect the amount of residual oxidants and other oxidants generated by radiolysis of residual water that cause pellet oxidation. A negative effect could be due to the increased local cladding stress from mechanical overload, which would cause the cladding to fail sooner as UO_2 oxidizes.

4.7.8 Impact of Mechanical Overload on Cladding Oxidation

The effect of mechanical overload on cladding oxidation is expected to be neutral because the two phenomena occur at different times during storage (mechanical overload occurs late in storage after cladding oxidation has already occurred). Additionally, mechanical overload does not affect the amount of residual oxidants and other oxidants generated by radiolysis of residual water that cause cladding oxidation.

4.8 Impact of Oxidation of UO_2

This section describes the impact of UO_2 oxidation on the other phenomena considered (i.e., thermal creep, DHC, mechanical fatigue, HRO/DtB, annealing, hydrogen migration, mechanical overload, and cladding oxidation).

Oxidation of the fuel requires a through-wall failure to allow air or water to contact the fuel. Therefore, it would only occur in fuel rods with a breach in the cladding when loaded, or in fuel rods that fail during storage. Failed fuel that is loaded into a damaged fuel can is beyond the scope of this PIRT. For fuel loaded with a cladding breach (either considered intact or not detected), oxidation would have begun at the time of failure and would also contain water within the rod. For fuel that may fail during drying-transfer-storage, oxidation of the fuel would not occur until after failure.

As shown in the figure below, the effect of UO_2 oxidation on the other phenomena is considered largely neutral, mainly because UO_2 oxidation can be partially accommodated by the increase in the void volume induced by creep. However, pellet increase of volume due to oxidation can still exert significant forces on the cladding to cause negative effects on phenomena driven by stress (e.g., DHC, HRO). More details are provided below.

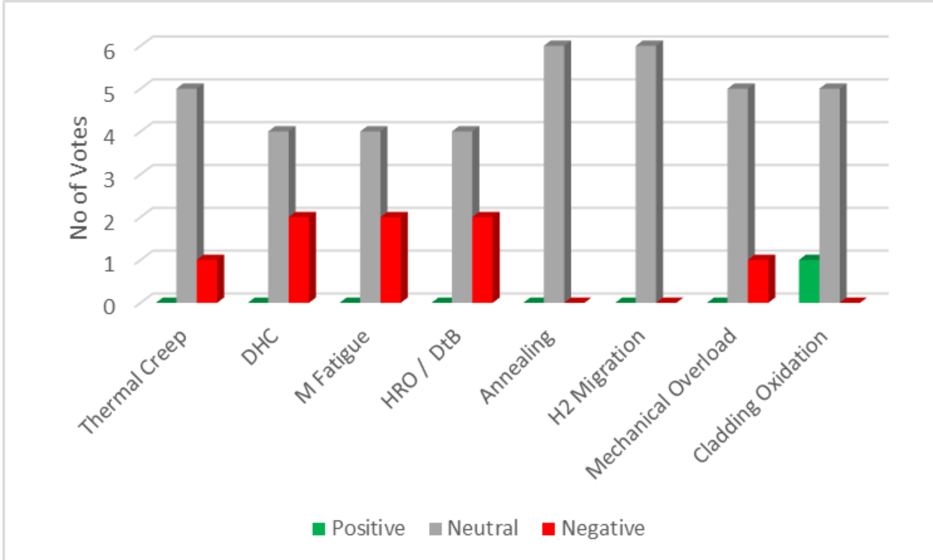

Figure 4-8
Impact of Oxidation of UO2 on the Other Phenomena

4.8.1 Impact of Oxidation of UO_2 on Thermal Creep

Fuel pellet oxidation has likely no effect on thermal creep. However, higher localized mechanical stresses due to UO_2 oxidation may reduce the localized creep strain capacity of the cladding, having a negative effect on thermal creep.

4.8.2 Impact of Oxidation of UO_2 on DHC

UO_2 oxidation can cause pellet cladding contact force that can cause localized mechanical stresses to exceed the critical stress intensity for DHC, with a negative synergistic effect.

However, the overall effect might also be neutral because U_3O_8 formation is much more detrimental than DHC: upon fuel oxidation, the fuel volume increases and the load on the cladding can produce a large cladding rupture. Therefore, variations related to DHC might be negligible in comparison to the effect of fuel oxidation on loading of the cladding.

4.8.3 Impact of Oxidation of UO_2 on Mechanical Fatigue

As already mentioned, UO_2 oxidation can cause higher localized mechanical stresses that may reduce the bending fatigue resistance of the cladding, causing a negative interplay between pellet oxidation and mechanical fatigue. However, the overall effect might also be neutral because variations related to mechanical fatigue are negligible in comparison to the stress on the cladding due to fuel oxidation.

4.8.4 Impact of Oxidation of UO_2 on HRO /DtB Transition

Localized pellet oxidation and subsequent stresses may increase the susceptibility to localized hydride reorientation and potentially decrease the cladding strength during hoop loads related to design-basis side drop accidents, causing a negative effect on HRO. However, the overall effect might also be neutral because variations related to HRO are negligible in comparison to the stresses on the cladding due to fuel oxidation, which are much more detrimental than HRO and can lead to gross rupture.

4.8.5 Impact of Oxidation of UO_2 on Annealing

Thermal annealing is a function of temperature and is not expected to be affected by UO_2 pellet oxidation. The overall effect of pellet oxidation on annealing is neutral.

4.8.6 Impact of Oxidation of UO_2 on Hydrogen Migration

Axial hydrogen migration is dependent on the hydrogen diffusion coefficient in the cladding. Although there is a need to confirm that the coefficient is not affected by higher localized hoop stresses due to UO_2 pellet oxidation, the effect of fuel pellet oxidation on hydrogen migration is likely neutral.

4.8.7 Impact of Oxidation of UO_2 on Mechanical Overload

The effect of fuel oxidation on mechanical overload is likely neutral, because it can be partially accommodated by the increase in the void volume induced by pellet oxidation. Additionally, the load applied on the cladding by fuel oxidation is large and variations related to mechanical overload are negligible in comparison.

A negative effect could be due to cladding thinning caused by strains associated with UO_2 pellet oxidation, which may decrease the strain capacity to accommodate mechanical overload.

4.8.8 Impact of Oxidation of UO_2 on Cladding Oxidation

The effect of fuel oxidation on cladding oxidation is likely neutral because the load applied by fuel oxidation on cladding outweigh variations related to cladding oxidation, which are negligible in comparison.

Fuel oxidation can have a positive effect on cladding oxidation because pellet oxidation depends on the amount of residual oxidants (at the time of SNF loading) and other oxidants generated by radiolysis of residual water. Consumption of oxidants by pellet oxidation decreases the extent of cladding oxidation, causing a lower cladding oxidation rate, which is a positive effect.

4.9 Impact of Cladding Oxidation

This section describes the impact of outer diameter cladding oxidation on the other phenomena considered (i.e., thermal creep, DHC, mechanical fatigue, HRO/DtB, annealing, hydrogen migration, mechanical overload, and pellet oxidation). Oxidation of the inner diameter (ID) is not considered in this section, because ID oxidation can only occur with an existing through-wall flaw.

Fuel-rod cladding discharged from the reactor has an outer-surface oxide layer in the range of 10–100 μm. Part of this oxide layer is effective in decreasing the oxidation rate during storage, which depends on the amount of oxidant (moisture) in the cask, the cladding surface temperature/area, and other metals within the cask that act as oxygen getters. The oxidation process in storage has slower kinetics compared to oxidation occurring in reactor due to the lower temperature and significantly reduced neutron flux.

As shown in the figure below, the effect of cladding oxidation on the other phenomena considered is either neutral or negative. More information is provided in the following sections.

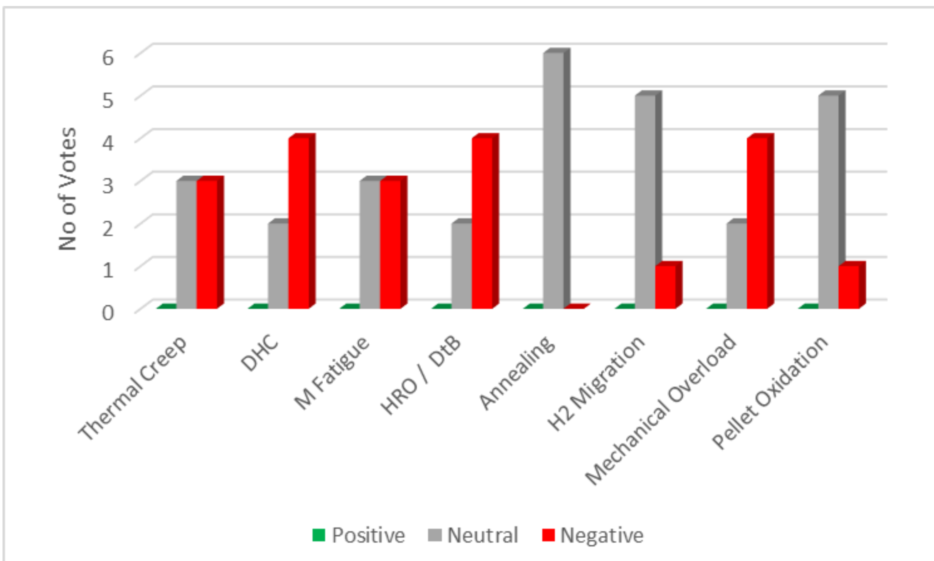

Figure 4-9
Impact of Cladding Oxidation on Other Phenomena

4.9.1 Impact of Cladding Oxidation on Thermal Creep

The effect of cladding oxidation on thermal creep might be neutral considering that the cladding is expected to already have an oxide layer on the outer diameter from in-reactor exposure, the additional oxide thickness might have a negligible effect on creep rate.

However, if the amount of cladding further oxidized is not negligible, cladding oxidation would reduce the zirconium-based metal thickness, which could reduce the strain capacity of the cladding. Therefore, the effect of cladding oxidation on thermal creep would be negative.

4.9.2 Impact of Cladding Oxidation on DHC

Cladding oxidation would reduce the zirconium-based metal thickness, which would reduce the critical flaw size (relative to the cladding thickness) needed for DHC initiation. Therefore, the effect of cladding oxidation on DHC would be negative. However, the corrosion rate might be very small, and the corresponding hydrogen uptake would also be small. In this case, the effect of cladding oxidation on DHC would likely be insignificant (neutral).

4.9.3 Impact of Cladding Oxidation on Mechanical Fatigue

Depending on the magnitude of the phenomenon, the effect of cladding oxidation on mechanical fatigue could be negative or neutral. Cladding oxidation would reduce the zirconium-based metal thickness in favor of an oxide layer, which would decrease the bending fatigue resistance of the cladding, and therefore increase the likelihood of failure from mechanical fatigue. On the other side, if the oxidation results in a minimal increase to oxide layer and the additional oxide layer formed is negligible compared to the existing oxide layer, the effect of cladding oxidation on mechanical fatigue results neutral.

4.9.4 Impact of Cladding Oxidation on HRO /DtB Transition

Cladding oxidation would reduce the zirconium-based metal thickness, which would reduce the cladding strain capacity during hoop loads related to design-basis side drop accidents, increasing the likelihood of failure from HRO. Therefore, the effect would be negative. Depending on the amount of oxidation, the effect would be negative, or neutral if the oxidation results in a minimal increase to the existing oxide layer.

4.9.5 Impact of Cladding Oxidation on Annealing

Thermal annealing is a function of temperature and not expected to be affected by cladding oxidation. Therefore, the effect of cladding oxidation on annealing is expected to be neutral.

4.9.6 Impact of Cladding Oxidation on Hydrogen Migration

Axial hydrogen migration is dependent on the diffusion coefficient in the cladding. The coefficient is not expected to be dependent on cladding oxidation. Therefore, it is likely that cladding oxidation has a neutral effect on hydrogen migration. However, cladding oxidation causes wall thinning and introduces more hydrogen having a negative effect.

4.9.7 Impact of Cladding Oxidation on Mechanical Overload

If the cladding oxidation rate is insignificant compared to in-reactor oxidation, the corresponding hydrogen uptake and wall thinning will also be small, having a neutral effect on mechanical overload.

On the other hand, if the oxidation rate is significant, the cladding oxidation would reduce the zirconium-based metal thickness, which would reduce the cladding strain capacity to account for mechanical overload. Therefore, the effect would be negative.

4.9.8 Impact of Cladding Oxidation on Oxidation of UO_2

Cladding oxidation would reduce the zirconium-based metal thickness, which would reduce the cladding strain capacity to account for UO_2 pellet oxidation. Therefore, the effect would be negative. However, the impact could also be neutral because the reduced thickness will be negligible compared to the stress of U_3O_8 formation.

Section 5: Conclusions

5.1 Summary

A panel of recognized fuel and cladding material experts was assembled to evaluate the current state of knowledge and to consider changes that may enable establishing new criteria for preventing gross rupture in spent fuel storage and transportation. Although the panel considered whether to propose changes to the regulations, it was agreed that the existing regulations are adequate. For this PIRT, the panel considered "regulations as written in pen and guidance as written in pencil." This led to some recommendations for changes in guidance discussed later in this section.

The key takeaways from the panel were:

- The onset of fuel failure is not abrupt, but rather a continuum – no cliff-edge effect is associated with the current NRC recommended 400°C limit [79]. Therefore, exceeding 400°C does not mean that multiple fuel rods will fail simultaneously.
- The evaluation performed by the panel did not quantify the extent of ruptures. Although probability of gross rupture of several rods in a single cask is lower than that of a single rod, the probabilities are not independent, as rods in a single fuel assembly have been exposed to similar conditions.
- Because experiments have focused on the current 400°C limit, there is a gap in knowledge to go beyond the limit, especially about the kinetics of annealing and cold-work recovery.
- At temperatures beyond 400°C, there is interaction among phenomena (synergistic, competitive, and aggravating effects) that should be further explored.
- The extent or consequences of rupture may be minor, even where a phenomenon may have a significant effect on possibility of gross rupture.
- The state of knowledge of most parameters is good, although applicability at temperatures higher than 400°C has lower confidence due to limited testing above 400°C.

The outcome of the PIRT exercise to determine the significance of various phenomena on propensity for gross rupture of fuel in dry storage and transport is summarized in Table 5-1. Nine of 12 phenomena were judged of low significance for all scenarios. The panel identified two phenomena that have medium significance (HRO and mechanical overload) in certain scenarios. One phenomenon was identified by some panelists as having high significance (UO_2 oxidation in fuel with cladding breaches), yet the other panelists considered UO_2 oxidation as low significance.

Table 5-1
Summary of Ranking of Phenomena Potentially Causing Gross Rupture

Scenarios / Phenomena	ST	Storage	LT Storage	ON	NCT
Thermal Creep	L	L	L	L	L
Low-Temperature Creep	L	L	L	L	L
Diffusion-Controlled Cavity Growth	L	L	L	L	L
Hydrogen Embrittlement	L	L	L	L	L
Delayed Hydride Cracking	L	L	L	L	L
Hydride Reorientation	L	M	M	L	L
Thermal Fatigue	L	L	L	L	L
Mechanical Fatigue	L	L	L	L	L
Annealing	L	L	L	L	L
Mechanical Overload	L	M	M	L	M
Fuel Oxidation*	L or M	L or H	L	L or H	L or H
Cladding Oxidation	L	L	L	L	L

Panel did not reach consensus on ranking (see Section 3.11), but fuel oxidation is potentially significant only for fuel rods with through wall cladding defects.

The discussion on the synergistic effects showed that it was difficult to reach unanimous opinions, suggesting that additional tests are required to define the interplay between the effects considered. According to the aggregated experts' opinion, most of the phenomena have a neutral effect on the other effects considered (73%) or a negative effect (20%). Only creep and annealing might mitigate the intensity of other phenomena. Improving understanding would require more evaluation and likely need experimental data aimed at isolating the interactions. Given that the majority of the synergistic effects were rated neutral and that the PIRT concluded that all but three phenomena were of low significance, better understanding would potentially only be beneficial for HRO and mechanical overload.

5.2 Opportunities Identified

The panel identified several opportunities for proposed changes to current guidance that could be explored based on the improved state of knowledge on cladding performance. These opportunities could increase operational flexibility, facilitate earlier transfer from wet to dry storage, reduce the number of dry storage systems needing transportation, and further contribute to risk-informed dry storage and aging management activities.

- Re-assess the limitations on thermal cycling (65°C, 10 cycles, as mentioned in ISG-11 Revision 3)
- Investigate implementation of a temperature criterion (e.g., average temperature for a percentage of the cladding) as an alternative to the peak cladding temperature limit (400°C) currently in the regulatory acceptance criteria.
- Extend the use of hoop stress analysis as a secondary justification from low burnup to high burnup fuel when approaching the 400°C limit (ISG-11 Revision 3).
- Implement a graded approach to protect against gross rupture. A possible approach could consist of defining different limits for different cladding materials based on the different effects each phenomenon has for different materials. The possible metrics worth exploring are:
  - Use of average temperature on a defined percentage of surface as an additional criterion in case the 400°C limit is exceeded.
  - Use of hoop stress analysis as a secondary justification when approaching the 400°C limit.

5.3 Recommendations

Finally, the panel understood that this PIRT was an initial step and recommended follow on work to get the most value from this exercise. In particular, the following actions are recommended:

- Discuss the definition of gross rupture. This should be done during a meeting with subject matter experts in criticality, fuels/cladding, and shielding technical disciplines.
- Hold a synthesis PIRT after completion of the 3 recently conducted PIRTs on Fuel Cladding Performance, Thermal Modeling, and Decay Heat to discuss the cumulative impact of overlapping margins/uncertainties. Feed the margins/uncertainties from the Thermal Modeling and Decay Heat PIRTs to the Fuel Cladding Performance PIRT to better understand and address overlapping conservatisms/uncertainties.

Section 6: References

1. Ankem, R and T. Wilt. 2006. *A Literature Review of Low Temperature (<0.25T_{mp}) Creep Behaviour of α, α-ß, and ß Titanium Alloys,* Center for Nuclear Waste Regulatory Analyses, San Antonio, Texas, September 2006.
2. Aomi, M., B. Toshikazu, M. Toshiyasu, K. Katsuichiro, Y. Takayoshi, S. Yasunari and T. Toru. 2008. *Evaluation of Hydride Reorientation Behaviour and Mechanical Properties for High-Burnup Fuel-Cladding Tubes in Interim Dry Storage*, Journal of ASTM International, Volume 5, Number 9, 2008.
3. Bajwa, Christopher S., Easton, Earl P., Osgood, Nancy L., and Pope, Ronald B. 2008. "A Comparison of 10 CFR 71 and the IAEA TS-R-1 Radioactive Material Transportation Regulations." *Proceedings of the ASME 2008 Pressure Vessels and Piping Conference. Volume 7: Operations, Applications and Components*. Chicago, Illinois, USA. July 27–31.
4. Billone, M.C., T.A. Butseva, Z. Han, and Y.Y. Liu. 2013. *Embrittlement and DBTT of High Burnup PWR Fuel Cladding Alloys*, ANL-13/16, Argonne National Laboratory, September 2013.
5. Billone, M.C., T.A. Butseva, Z. Han, and Y.Y. Liu. 2014. *Effects of Multiple Drying Cycles on High-Burnup PWR Cladding Alloys*, ANL-14/11, Argonne National Laboratory, September 2014.
6. Billone, M.C., T.A. Burtseva, and M.A. Martin-Rengel. 2015. *Effects of Lower Drying-Storage Temperatures on the DBTT of High-Burnup PWR Cladding*, ANL-15/21, Argonne National Laboratory, August 2015.
7. Bouffioux P., A. Ambard, A. Miquet, C. Cappelaere, Q. Auzoux, M. Bono, O. Rabouille, S. Allegre, V. Chabretou, and C.P. Scott. 2013. "Hydride Reorientation in M5® Cladding and its Impact on Mechanical Properties." Top Fuel 2013, Paper 8557, pp. 879-886.
8. Bouffioux P., and L. Legras. 2000. "Effect of Hydriding on the Residual Cold Work Recovery of Zircaloy-4 Cladding Tubes. Consequences on the Creep Behavior of the Cladding Tubes Under Dry Storage Conditions." ANS International Topical Meeting on LWR Fuel Performance. Park City, Utah, 2000.
9. Cappelare, C., R. Limon, C. Duguay, G. Pinte, M. Le Breton, and P. Bouffioux. 2012 *Thermal Creep Model for CWSR Zircaloy-4 Cladding Taking into Account the Annealing of the Irradiation Hardening,* Nuclear Technology, 177:2, 257-272, 2012.

10. Chan, K.S. 2013. *An Assessment of Delayed Hydride Cracking in Zirconium Alloy Cladding Tubes Under Stress Transients*, International Materials Reviews, Volume 58, Number 6, pp. 349-373, 2013.

11. Chin, B.A., M.A. Khan, J.C.L. Tarn, and E.L. Gilbert. 1986. *Deformation and Fracture Map Methodology for Predicting Cladding Behavior During Dry Storage*, PNL-5998, Auburn University, September 1986.

12. Cox, B. 1997. *Hydrogen Trapping by Oxygen and Dislocations in Zirconium Alloys*, Journal of Alloys and Compounds, Volume 256, Issues 1-2, July 1997.

13. Devoe, R. and K.R. Robb. 2015. "COBRA-SFS Dry Cask Modeling Sensitivities in High-Capacity Canisters." Proceedings of the International High-Level Radioactive Waste Management Conference, April 12–16, 2015. Paper No. 12701. Charleston, South Carolina. 2015.

14. Dutton, R., K. Nuttall, M.P. Puls and L.A. Simpson. 1977. *Mechanism of Hydrogen Induced Delayed Hydride Cracking in Hydride Forming Materials*, Metall. Trans. A, 8A, 1553–1562.

15. Einziger, R.E E., S. D. Atkin, D. E. Stellbrecht, and V. Pasupathi. 1982. *High Temperature Postirradiation Materials Performance of Spent Pressurized Water Reactor Fuel Rods Under Dry Storage Conditions*. Nuclear Technology, v. 57, p. 65, 1982.

16. Einziger, R.E., H. Tsai, M.C. Billone, B.A. Hilton. 2003. *Examination of Spent PWR Fuel Rods after 15 Years in Dry Storage*, NUREG/CR-6831, Argonne National Laboratory, September 2003.

17. Electric Power Research Institute (EPRI). 1998. *Data Needs for Long-Term Dry Storage of LWR Fuel*, TR-108757, April 1998.

18. Electric Power Research Institute (EPRI). 2000. *Creep as the Limiting Mechanism for Spent Fuel Dry Storage*, EPRI, Palo Alto, CA: 2000. 1001207.

19. Electric Power Research Institute (EPRI). 2001. *Creep Modeling and Analysis Methodology for Spent Fuel in Dry Storage*, EPRI, Palo Alto, CA: 2001. 1003135.

20. Electric Power Research Institute (EPRI). 2001. *Fracture Toughness Data for Zirconium Alloys: Application to Spent Fuel Cladding in Dry Storage*, EPRI, Palo Alto, CA: 2001. 1001281.

21. Electric Power Research Institute (EPRI). 2002. *Technical Bases for Extended Dry Storage of Spent Nuclear Fuel*, EPRI, Palo Alto, CA: 2002. 1003416.

22. Electric Power Research Institute (EPRI). 2011. *Delayed Hydride Cracking Considerations Relevant to Spent Nuclear Fuel Storage*, EPRI, Palo Alto, CA: 2011. 1022921.

23. Electric Power Research Institute (EPRI). 2011. *Extended Storage Collaboration Program (ESCP): Progress Report and Review of Gap Analyses*. EPRI, Palo Alto, CA: 2011. 1022914.

24. Electric Power Research Institute (EPRI). 2011. *Structural Integrity of Advanced Claddings During Spent Nuclear Fuel Transportation and Storage: Volume 2: Creep Testing of LK3 Zircaloy-2 Cladding Tubes*, EPRI, Palo Alto, CA: 2011. 1022918.

25. Electric Power Research Institute (EPRI). 2011. *Structural Integrity of Advanced Claddings During Spent Nuclear Fuel Transportation and Storage: Volume 1: Creep Testing of ZIRLO™ Cladding Tubes*, EPRI, Palo Alto, CA: 2011. 1022919.

26. Electric Power Research Institute (EPRI). 2012. *Extended Storage Collaboration Program International Subcommittee Report: International Perspectives on Technical Data Gaps Associated with Extended Storage and Transportation of Used Nuclear Fuel*. EPRI, Palo Alto, CA: 2012. 1026481.

27. Electric Power Research Institute (EPRI). 2016. *Creep and Creep-to-Rupture Testing of Optimized ZIRLO™,* EPRI, Palo Alto, CA: 2016. 3002008040.

28. Electric Power Research Institute (EPRI). 2019. *High Burnup Dry Storage Research Project Cask Loading and Initial Result*, EPRI, Palo Alto, CA: 2019. 3002015076.

29. Electric Power Research Institute (EPRI). 2019. *Impact of Fuel-Cladding Bonding on the Response of High-Burnup Spent PWR Fuel Subjected to Dry Storage and Transportation Accidents*, EPRI, Palo Alto, CA: 2019. 3002015075.

30. Electric Power Research Institute (EPRI). 2020. *Spent Fuel Dry Storage Thermal Modeling: Results from A Phenomena Identification and Ranking Table (PIRT) Process*, EPRI, Palo Alto, CA: 2020. 3002018441.

31. Electric Power Research Institute (EPRI). 2020. *Phenomena Identification and Ranking Table (PIRT) for Decay Heat*, EPRI, Palo Alto, CA: 2020. 3002018440.

32. Ferry, C., C. Poinssot, V. Broudic, C. Cappelaere, L. Desgranges, P. Garcia, C. Jegou, P. Lovera, P. Marimbeau, J-P Piron, A. Poulesquen, D. Roudil, J-M Gras and P. Bouffioux. 2005. Synthesis on the Spent Fuel Long Term Evolution, CEA report CEA_R-6084, 2005.

33. Goll, W., H. Spilker, and E.H. Toscano. 2001. *Short-time Creep and Rupture Tests on High Burnup Fuel Rod Cladding*, Journal of Nuclear Materials, Volume 289, Issue 3, March 2001.

34. Grigoriev, V., Jakobsson, R. 2005. *Delayed Hydrogen Cracking Velocity and J-Integral Measurements on Irradiated BWR Cladding*, in *Zirconium in the Nuclear Industry: Fourteenth International Symposium*, ed. P. Rudling and B. Kammenzind (West Conshohocken, PA: ASTM International, 2005), 711-728.

35. Hanson, B. D., R.C. Daniel, A.M. Casella, R.S. Wittman, W. Wu, P.J. MacFarlan, and R.W. Shimskey. 2008. Fuel-In-Air FY07 Summary, U.S. Department of Energy, PNNL-17275, Rev. 1, 2008.

36. Hanson, B., H. Alsaed, C. Stockman, D. Enos, R. Meyer, and K. Sorenson. 2012. *Gap Analysis to Support Extended Storage of Used Nuclear Fuel*, PNNL-20509, Revision 0, January 2012.

37. Hanson, B., for E. Kalinina. 2019. *International Multi-Modal Surrogate Spent Nuclear Fuel Transportation Test: The SNF Transportation Test Triathlon*, PNNL-SA-147626, IAEA Technical Meeting Technical and Operational Issues Related to the Transportation of High-Burnup and Irradiated Mixed Oxide Fuels and the Transportability of Long-Term Stored Spent Fuel, Vienna, September 2019.

38. Hermann, A., Martin, M., Poerschke, P., and Yagnik, S. 2001. *Ductility Degradation of Irradiated Fuel Cladding*, (INIS-CH--034). Gschwend, Beatrice (Ed.). Switzerland, March 2001.

39. International Atomic Energy Agency (IAEA). 2012. *Spent Fuel Performance Assessment and Research: Final Report of a Coordinated Research Project* (SPAR-II), IAEA-TECDOC-1680, Vienna, 2012.

40. International Atomic Energy Agency (IAEA). 2015. *Spent Fuel Performance Assessment and Research Final Report of a Coordinated Research Project on Spent Fuel Performance Assessment and Research* (SPAR-III) 2009–2014, IAEA-TECDOC-1771, Vienna, 2015.

41. International Atomic Energy Agency (IAEA). 2018. *Regulations for the Safe Transport of Radioactive Material*, SSR-6, Revision 1, 2018.

42. Jung, H., P. Shukla, T. Ahn, L. Tipton, K. Das, X. He, and D. Basu. 2013. *Extended Storage and Transportation: Evaluation of Drying Adequacy*, Center for Nuclear Waste Regulatory Analyses, San Antonio, Texas, June 2013.

43. Kamimura, IS., Y. Kobayashi, and T. Nomata. 2000. *Helium Generation and Release in MOX Fuels*, IAEA-SM-358/19, Nuclear Power Engineering Corporation, Tokyo, Japan, 2000.

44. Kammenzind, B., D. Franklin, H. Peters, and W. Duffin. 1996. “Hydrogen Pickup and Redistribution in Alpha-Annealed Zircaloy-4,” Zirconium in the Nuclear Industry: Eleventh International Symposium, E. Bradley and G. Sabol (West Conshohocken, PA: ASTM International, 1996), 338-370.

45. Kubo, T., Y. Kobayashi, and H. Uchikoshi. 2012. “Measurements of Delayed Hydride Cracking Propagation Rate in the Radial Direction of Zircaloy-2 Cladding Tubes,” *Journal of Nuclear Materials*, Volume 427, Issue 1-3, August 2012.

46. Legras, L. 1998. “Influence of Transport and Dry Storage on the Metallurgical Structure of the Cladding Material.” HT-41/98/008/A, Document EDF R&D, 1998.

47. Lin, X. and G. Haicheng. 1998. “High Cycle Fatigue Properties and Microstructure of Zirconium and Zircaloy-4 Under Reversal Bending.” Materials Science and Engineering A. Vol. 252. pp. 166–173. 1998.

48. Machiels, A. 2002. *Regulatory Applications Lessons Learned -- Industry Perspective*. NEI, 2002.

49. Machiels A.J., K. Waldrop, and P. Rudling. 2019. *Perspective on Thermal Creep and Hydride Reorientation for Dry Storage and Transportation Applications,* International Conference on the Management of Spent Fuel from Nuclear Power Reactors 2019: Learning from the Past, Enabling the Future, IAEA, June 2019, Vienna, Austria.

50. Matsuo. 1987. Thermal Creep of Zircaloy-4 Cladding under Internal Pressure, *Journal of Nuclear Science and Technology*, 24:2. pp. 111-119, February 1987.

51. Montgomery, R., R. Morris, R. Ilgner, B. Roach, J-A Wang, Z. Burns, J. Dixon, and S. Curlin. 2019. Sister Rod Destructive Examinations (FY19), ORNL/SPR-2019/1251 Rev. 1, Sept. 27, 2019

52. Motta, A., L. Capolungo, L-Q Chen, M. Cinbiz, M. Daymond, D. Koss, E. Lacroix, G. Pastore, P-C Simon, M. Tonks, B. Worth, and M. Zikry. 2019. *Hydrogen in Zr alloys: A Review*, Journal of Nuclear Materials, Volume 518, May 2019, pp. 440-460.

53. Patterson, C., F. Garzarolli, R. Adamson, and K. Coleman. 2015. *Dry Storage Handbook: Fuel Performance in Dry Storage*, Advanced Nuclear Technology International, Mölnlycke, Sweden, March 2015.

54. Perovic V., Weatherly G.C. and Simpson C.J. 1983. "Hydride Precipitation in α/β Zirconium Alloys," Acta Metall. 31, p. 1381-1391, 1983.

55. Perovic, V., Weatherly, G.C. 1984. *The nucleation of hydrides in a Zr-2.5 wt% Nb alloy,* Journal of Nuclear Materials, Volume 126, Issue 2, 1984.

56. Petit T., P. Martin, M. Ripert, N. Olivier, A. Chevarier, and N. Moncoffre. 2001. "Localization of Helium and Volatile Fission Products in Uranium Dioxide: Ab Initio and X-Ray Absorption Spectroscopy Studies." ICEM 2001, Session 12, Bruges, Belgium.

57. Pettersson, K. 2007. *The Ductility of Radial Hydrides,* ASTM Workshop on Hydride Reorientation in Zirconium Alloys, January 29-February 1, 2007, Anaheim, California.

58. Puls, M. P. 2012. The Effect of Hydrogen and Hydrides on the Integrity of Zirconium Alloy Components, ISSN 1612-1370, Springer-Verlag London 2012.

59. Rashid, Y.R., R. Montgomery, W. Lyon, and R. Yang. 2000. *A Cladding Failure Model for Fuel Rods Subjected to Operational and Accident Transients*, IAEA-TECDOC-1233. IAEA, Vienna, 2000.

60. Rashid, Y.R., and A. Machiels. 2007. "Threat of Hydride Re-orientation to Spent Fuel Integrity During Transportation Accidents: Myth or Reality?" Proceedings of the 2007 International LWR Fuel Performance Meeting, Paper ID-1039, San Francisco, CA, September 30 - October 3, 2007.

61. Raynaud, P. & Einziger, R.E. 2015. *Cladding Stress During Extended Storage of High Burnup Spent Nuclear Fuel,* Journal of Nuclear Materials, Volume 464, pp. 304-312, September 2015.

62. Richmond D.J., and KJ Geelhood. 2018. *FRAPCON Analysis of Cladding Performance during Dry Storage Operations*. PNNL-27418, April, 2018.

63. Rodgers, D., Griffiths, M., Bickel, G., Buyers, A., Coleman, C., Nordin, H., and St. Lawrence, S. 2016. *Performance of Pressure Tubes in CANDU Reactors*, CNL Nuclear Review, Vol. 5, pp. 1-15, 2016.

64. Rondinella, V.V., Wiss, T., Maugeri, E., Colle, J-Y. 2011. *Effects of helium build-up on nuclear fuel evolution during storage*, Proc. IHLRWMC 2011, Albuquerque, New Mexico, 2011, ISBN 9781617828508, Curran Associates, NY. 2011, 230–233.

65. Rufeh F., Olander D.R., Pigford T.H. 1965. "The Solubility of Helium in Uranium Dioxide," *Nuclear Science and Engineering*, 23, pp. 335-338, 1965.

66. Sasahara, A. 2006. *Experiment and Calculation of Axial Hydrogen Migration in Spent Fuel Cladding*, IAEA-SPAR-II, Central Research Institute of Electric Power Industry, November 6-10, 2006.

67. Sasahara, A. and T. Matsumura. 2008. *Post-Irradiation Examinations Focused on Fuel Integrity of Spent BWR-MOX and PWR-UO2 Fuels Stored for 20 Years*, Nuclear Engineering and Design, Volume 238, pp. 1250-1259, 2008.

68. Schemel J. 1989. *Zirconium Alloy Fuel Clad Tubing - Engineering Guide*, Sandvik Special Metals, pp. 70-71, December 1989.

69. Seibold, A. and F. Garzarolli. 2002. *Influence of Composition and Condition on Behavior of Zr-Sn-FeCrV Alloys*, in Zirconium in the Nuclear Industry: Thirteenth International Symposium, ed. G. Moan and P. Rudling (West Conshohocken, PA: ASTM International, 2002), 743-757.

70. Sercombe, J., I. Aubrun, and C. Nonon. 2012. "*Power ramped cladding stresses and strains in 3D simulations with burnup-dependent pellet–clad friction*, Nuclear Engineering and Design, Volume 242, pp. 164-181, 2012.

71. Shimada, S., E. Etoh, H. Hayashi, and Y. Tukuta. 2004. *A metallographic and fractographic study of outside-in cracking caused by power ramp tests*, J. Nucl. Materials. 327. 97–113.

72. Shimskey, RW, JR Allred, RC Daniel, MK Edwards, J Geeting, PJ MacFarlan, LI Richmond, TS Scott, and BD Hanson. 2019. PNNL Phase 1 Update on Sibling Pin Destructive Examination Results, PNNL-29179, September 27, 2019.

73. Shukla, P., R. Sindelar, P-S. Lam. 2019. *Consequence Analysis of Residual Water in a Storage Canister- Preliminary Report*, SRNL-STI-2019-00495, Savannah River National Laboratory, September 2019.

74. Torres, R. 2019. *Spent Fuel Performance in Dry Storage and Transportation*, U.S. Nuclear Regulatory Commission, EPRI Fuel/Cladding PIRT Meeting, October 2019.

75. U.S. Nuclear Regulatory Commission (NRC). 1999. *Storage of High Burnup Spent Fuel, ISG-11,* Revision 0, May 1999.

76. U.S. Nuclear Regulatory Commission (NRC). 2000. *Transportation and Storage of Spent Fuel Having Burnups in Excess of 45 GWD/MTU, ISG-11,* Revision 1, May 16, 2000.
77. U.S. Nuclear Regulatory Commission (NRC). 2002. *Burnup Credit PIRT Report,* NUREG/CR-6764, May 2002.
78. U.S. Nuclear Regulatory Commission (NRC). 2002. *Cladding Considerations for the Transportation and Storage of Spent Fuel, ISG-11,* Revision 2, July 30, 2002.
79. U.S. Nuclear Regulatory Commission (NRC). 2003. *Cladding Considerations for the Transportation and Storage of Spent Fuel, ISG-11,* Revision 3, November 17, 2003.
80. U.S. Nuclear Regulatory Commission (NRC). 2006. *Potential Rod Splitting Due to Exposure to an Oxidizing Atmosphere During Short-term Cask Loading Operations in LWR or Other Uranium Oxide Based Fuel*, ISG-22, May 8, 2006.
81. U.S. Nuclear Regulatory Commission (NRC). 2007. *Classifying the Condition of Spent Nuclear Fuel for Interim Storage and Transportation Based on Function,* ISG-1, Revision 2, May 11, 2007.
82. U.S. Nuclear Regulatory Commission (NRC). 2008. *Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs)*, NUREG/CR-6944, March 2008.
83. U.S. Nuclear Regulatory Commission (NRC). 2010. “Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility.” NUREG-1536 Rev 1. Washington, DC: U.S. Nuclear Regulatory Commission. July 2010.
84. U.S. Nuclear Regulatory Commission (NRC). 2016. *Fuel Retrievability in Spent Fuel Storage Applications,* ISG-2, Revision 2, April 26, 2016.
85. U.S. Nuclear Regulatory Commission (NRC). 2017. *Mechanical Fatigue Testing of High-Burnup Fuel for Transportation Applications*, NUREG/CR-7198, Revision 1, ORNL/TM-2016/689, Oak Ridge National Laboratory, October 2017.
86. U.S. Nuclear Regulatory Commission (NRC). 2018. *Dry Storage Transportation of High Burnup Spent Nuclear Fuel,* NUREG-2224, July 2018.
87. U.S. Nuclear Regulatory Commission (NRC). 2019. *Managing Aging Processes in Storage (MAPS) Report,* NUREG-2214, July 2019.
88. Wiss, T., J-P Hiernaut, D. Roudil, J-V Colle, E. Maugeri, Z. Talip, A. Janssen, V. Rondinella, R. Konings, H-J Matzke, and W. Weber. 2014. *Evolution of Spent Nuclear Fuel in Dry Storage Conditions for Millennia and Beyond*, Journal of Nuclear Materials, Volume 451, pp. 198-206, April 2014.
89. Woodford, D.A. 1969. *Strain Rate Sensitivity as a Measure of Ductility*, Transactions of the ASM, Vol. 62, pp. 291-293, 1969.

Appendix A: Panelist and Facilitator Resumes

Rosemary A. Montgomery, P.E.

EDUCATION/TRAINING

B.S., Mechanical Engineering, University of South Carolina, 1995
Licensed professional engineer, Mechanical, State of Tennessee (106908)

EXPERIENCE

Sr. Research Staff, Oak Ridge National Laboratory, Oak Ridge, TN, 2016 to present

Specification and implementation of nondestructive and destructive examinations of spent nuclear fuel in support of safe storage and transport of used commercial nuclear fuel.

Development of novel examination methods and applications.

Mechanical and thermal simulations supporting fuel performance predictions and evaluation and licensing of packaging for nuclear applications.

Compilation and synthesis of data and facilitation of collaboration in support of advancing the current understanding of nuclear fuel response to thermal and mechanical challenges.

Sr. Program Manager, Tennessee Valley Authority, Chattanooga, TN, 2009 to 2016

Responsible for TVA's contribution to DOE's Consortium for Advanced Simulation of LWRs (CASL).

Chair, Industry Advisory Committee to Idaho National Lab on advanced LWR fuel research topics.

Technical review of proposed mechanical fuel design changes and any related operational impacts, fabrication oversight of nuclear fuel used at TVA reactors, participation in industry teams and initiatives on fuel reliability (e.g., EPRI, INPO, Affinity); TVA training on fuel-related topics.

Supervisor, Fuel Mechanics/Dynamics, AREVA NP Inc., Lynchburg, VA, 2008 to 2009: Team responsibility for dynamic fuel response including seismic, LOCA, and FIV, including specification of assembly and component testing.

Principal Engineer, Fuel Mechanical & Structural Design, AREVA NP Inc., Lynchburg, VA, 2005 – 2008: Fuel rod analyses including stress, transient strain, creep collapse, fatigue, corrosion; fuel assembly structural analyses including normal hot operation, faulted conditions, shipping & handling, holddown; related component and assembly testing and evaluation; development and testing of new shipping containers for fresh fuel.

U.S. Post-irradiation Examination Coordinator, 2005 – 2009: Specification of post-irradiation examinations for customer plants, coordination of exam plans, data evaluation, and reporting. Recommendations for and development of new or improved exam techniques.

Consultant, Montgomery Engineering & Technical Services, Johnson City, TN, 1997 to 2005

Packaging design, evaluation, testing and SARP preparation for all types of radioactive materials including fresh fuel assemblies, MOX, UF6, spent fuel (on-site storage, transfer and transport), UO2 powder and pellets, liquid uranyl nitrate, low level and high level waste.

Project Engineer, Chem-Nuclear Systems, Columbia, SC, 1995 to 1997

Thermal design and evaluation of DOE's MPC (storage and transfer cask), radiation shielding analyses, structural and thermal evaluations for CNS packaging; supporting analyses for the waste disposal portion of the business.

Product Design Technician, Westinghouse Commercial Nuclear Fuels, Columbia, SC, 1989 to 1993

SELECTED PUBLICATIONS

1. Rose Montgomery, Robert N. Morris, Measurement and modeling of the gas permeability of high burnup pressurized water reactor fuel rods, Journal of Nuclear Materials, Volume 523, 2019, Pages 206-215, ISSN 0022-3115, https://doi.org/10.1016/j.jnucmat.2019.05.041.
2. Rose Montgomery, Robert N. Morris, Bruce Bevard & John Scaglione (2019) Key Results from Detailed Nondestructive Examinations of 25 Pressurized Water Reactor High Burnup Spent Nuclear Fuel Rods, Nuclear Science and Engineering, DOI: 10.1080/00295639.2019.1573602.
3. R.A. Montgomery. Non-Destructive Pressure Measurement Technique for Irradiated Nuclear Fuel Rods, TopFuel Water Reactor Fuel Conference, Prague Czech Republic, September 2018.
4. R.A. Montgomery et al. Sister Rod Nondestructive Examination Final Report, ORNL/SPR-2018/801.
5. R.A. Montgomery et al. Design and Fabrication of a Heat Treatment Oven for Full-Length Spent Nuclear Fuel Rods, ANS Summery Meeting and Technology Expo, June 2018.
6. R. Montgomery, R. Morris, B. Bevard and J. Scaglione. Gamma Scanning of 25 PWR Spent Fuel Rods in the High Burnup Spent Fuel Data Project, ANS Winter Meeting and Technology Expo, November 2017.
7. Rose Montgomery, J. Scaglione, B. Williamson, and B. Wakeman. Experience with Used Nuclear Fuel Reimmersion for Repackaging after Three Years in Dry Storage, ANS Winter Meeting and Technology Expo, November 2017.

8. Enhanced Accident Tolerant LWR Fuels: Metrics Development, Proceedings of the American Nuclear Society Top Fuel Conference, September 2013.
9. Industry-valued Design Objectives for Advanced LWR Fuels and Concept Screening Results, Proceedings of the American Nuclear Society Top Fuel Conference, September 2013.
10. M5® Cladding Behavior with Zinc Injection: Results Obtained at Sequoyah-2, International Conference On Water Chemistry Of Nuclear Reactor Systems, (September 17, 2008), Berlin, Germany.

Brady Hanson, Ph.D.

AFFILIATION: Pacific Northwest National Laboratory

FIELD OF EXPERTISE

Spent fuel performance; fuel oxidation and dissolution; cladding performance

EDUCATION/TRAINING

B.S. Chemical Engineering; M. Eng. Nuclear Engineering; Ph.D. Nuclear Engineering

EXPERIENCE

Dr. Hanson has performed testing of spent fuel (commercial and DOE) for over 25 years. He first supported the Yucca Mountain Project and more recently has served as the Lab Lead for Experiments in the DOE-NE Spent Fuel and Waste Science and Technology program. He served as the lead checker for the initial cladding models for Yucca Mountain. He led the team that developed the initial gap analysis to identify issues to be addressed for the extended storage and transportation of spent nuclear fuel. Dr. Hanson leads the team at PNNL performing destructive testing of the sibling pins from the DOE High Burnup Spent Fuel Data Project demonstration cask. He served as the Technical Program Co-Chair of the 16th International High-Level Radioactive Waste Management Conference, a chair for the 2015 IAEA International Conference on the Management of Spent Fuel from Nuclear Power Reactors, and the co-chair of the 2008 Effects of Radiation on Nuclear Materials and the Nuclear Fuel Cycle symposium. He has presented at numerous national and international meetings and conferences.

SELECTED PUBLICATIONS:

1. Shimskey et al., *PNNL Phase 1 Update on Sibling Pin Destructive Examination Results.* PNNL-29179, Sept. 2019.
2. Hanson, BD and HA Alsaed, *Gap Analysis to Support Extended Storage and Transportation of Spent Nuclear Fuel: Five-Year Delta.* PNNL-28711, May 2019.
3. Hanson, BD, H Alsaed, CT Stockman, DG Enos, R Meyer, and KB Sorenson. *Gap Analysis to Support Extended Storage of Used Nuclear Fuel*. FCRD-USED-2011-000136, PNNL-20509. Jan. 2012.
4. McConnell P, BD Hanson, M Lee, and KB Sorenson. 2011. "Extended Dry Storage of Used Nuclear Fuel: Technical Issues: A USA Perspective." *Nuclear Engineering and Technology* 43(5):405-412.
5. Hanson BD, RC Daniel, AM Casella, RS Wittman, W. Wu, PJ MacFarlan, RW Shimskey, *Fuel-In-Air FY07 Summary Report*, PNNL-17275, Rev. 1, Pacific Northwest National Laboratory, Richland, WA, September 2008.
6. Hanson BD, *The Burnup Dependence of Light Water Reactor Spent Fuel Oxidation*. PNNL-11929, Pacific Northwest National Laboratory, Richland, WA, July 1998.

Michael Billone, Ph.D.

SUMMARY

Dr. Billone is a Senior Mechanical Engineer with Argonne National Laboratory. He has 43 years of experience in modeling the behavior of reactor materials in fission and fusion environments and in pre-test planning and post-test analysis of single-effects and integral-performance experiments conducted to determine the behavior of metals and ceramics under normal and accident reactor conditions. He has coordinated international analytical and experimental research efforts dedicated to better understand the behavior of reactor materials, especially fission-reactor fuel-cladding and fusion-reaction tritium-breeding and structural materials. For the past 17 years, he has managed NRC-sponsored research programs in the areas of high-burnup fuel cladding behavior during and following loss-of-coolant accidents (LOCAs), during spent nuclear fuel (SNF) pre-storage operations and dry-cask storage, and during normal and hypothetical-accident SNF cask transport. The dry-storage work is currently funded by DOE-NE. He has participated in NRC-sponsored PIRTs for design-basis LOCAs, design-basis reactivity insertion accidents, anticipated transients without scram.

EDUCATION/TRAINING

Ph.D. Mechanical Engineering, Northwestern University, 1972
B.E. Engineering Sciences, Thayer School of Engineering, 1966
B.A. Engineering Sciences, Dartmouth College, 1965

PROFESSIONAL AFFILIATIONS

Member, ASME, ASTM and ANS

Chairperson, Mechanical Properties Experts Group, Ad Hoc Group formed by the U.S. NRC and its international partners (France, Japan and Sweden) to share and compare mechanical-properties test and data-refinement methods for Zr-based cladding alloys, 2000-2005.

AWARDS AND HONORS

Recipient, 2005 University of Chicago Award for Distinguished Performance (June 16, 2005)

PROFESSIONAL EXPERIENCE

Current Laboratory Assignment, Manager: Irradiation Performance Program, Nuclear Science & Engineering Division. Manager of Radiological Facilities used to conduct research with irradiated materials.

Argonne Principal Investigator for DOE-NE Spent Fuel & Waste Disposition Program.

Senior Mechanical Engineer (2008), Argonne National Laboratory, 1998-2015

Principal Investigator (PI) for NRC high-burnup fuel LOCA and SNF programs; coordinated first-of-a-kind experiments with fueled (LOCA) and defueled (LOCA and SNF) cladding samples; and expert in light-water-reactor Zr-based cladding alloys. PI for high-burnup fuel cladding embrittlement research sponsored by DOE-NE Used Fuel Disposition Campaign.

Mechanical Engineer, Argonne National Laboratory, 1984-1997

Materials and modeling expert for DOE Fusion Power and International Thermonuclear Experimental Reactor (ITER) Programs for Li-based tritium breeding ceramics, beryllium multipliers, and structural materials.

Fuel and cladding materials and modeling expert for DOE advanced fast-reactor metal fuels [U-Pu-Zr] program; developed fuel-performance codes for fast-reactor fuel rods during normal and off-normal conditions.

Modeled performance of research and test reactor low-enrichment fuel plates.

Modeled performance of thermo-electric response of fuel-cell materials.

Assoc. Prof. of Mechanical and Nuclear Engineering, Northwestern University. 1978-1984

Taught upper-level nuclear and mechanical engineering courses.

Sponsored student research leading to 7 MS and 3 PhD theses.

Continued research efforts in modeling fission and fusion reactor materials.

Mech. Eng., Assist. Mech. Eng., Post-doc, Argonne National Laboratory, 1972-1978

Developed fuel-rod performance codes for fast-reactor mixed-carbide, mixed-nitride fuel (1976-78, Mech. Eng. and Leader of Modeling Group) and mixed-oxide fuel (1974-76, Assist. Mech. Eng.).

Visiting Professor at Northwestern University (1976-77).

SELECTED PUBLICATIONS

1. M.C. Billone, "Ductility of High-Exposure ZIRLO™ Cladding following Drying and Storage, Proc. International High-Level Radioactive Waste Management Conference (IHLRWM) 2019, Knoxville, TN, April 14-18, 2019, pp 536-545.
2. M.C. Billone, Y. Yan, T.A. Burtseva, and R.O. Meyer, *Cladding Behavior during Postulated Loss-of-Coolant Accidents*, NUREG/CR-7219, ANL-16/09, July 2016
3. M.C. Billone and Y. Chen, "Argonne Capabilities for Radioactive Materials Characterization in the Irradiated Materials Laboratory (IML)," Proc. ANS 2015 Winter Meeting, Washington, DC, Nov. 8-12, 2015.
4. M.C. Billone, T.A. Burtseva, and Y.Y. Liu, "Characterization and Effects of Hydrides in High-Burnup PWR Cladding Alloys," Proc. International High-Level Radioactive Waste Management Conference (IHLRWM), April 12-16, 2015, Charleston, SC.
5. M.C. Billone, T.A. Burtseva, and R.E. Einziger, "Ductile-to-brittle transition temperature for high-burnup cladding alloys exposed to simulated drying-storage conditions," J. Nucl. Mater. 433(2013) 431-438.
6. M.C. Billone, *Assessment of Current Test Methods for Post-LOCA Behavior*, NUREG/CR-7139, ANL-11/52 (2012).
7. Y. Yan, T.A. Burtseva, and M.C. Billone, "High-temperature steam-oxidation behavior of Zr-1Nb cladding alloy E110," J. Nucl. Mater. 393 (2009) 433-448.
8. M. Billone, Y. Yan, T. Burtseva, and R. Daum, *Cladding Embrittlement during Postulated Loss-of-Coolant Accidents*, NUREG/CR-6967, ANL-07/04 (2008).
9. Other (available upon request)

Joe Rashid, Ph.D.

AFFILIATION: Structural Integrity Associates (SI), Senior Advisor

FIELD OF EXPERTISE

Nuclear Fuel Behavior, Computational Mechanics

EDUCATION/TRAINING

BS., MS., Ph.D., UC Berkeley, Civil and Environmental Engineering (CEE)

PROFESSIONAL AFFILIATIONS

2015 inductee in UC Berkeley CEE Academy of Distinguished Alumni

2018 recipient of ANS Mishima Award

Fellow Member of ANS and ASME

ASTM Committee 26.13

California License-Nuclear

EXPERIENCE

Dr. Rashid's career in the nuclear industry began in 1965, and during his long professional career he contributed to the technical literature in a wide range of topics, with over 200 publications as journal articles, conference proceedings and topical reports. He served on Expert Elicitation Meetings and NRC PIRT Panels for severe accidents, some of which are listed below:

- Nominated by the NRC to the International Organization for Economic Co-operation and Development (OECD) to serve as Chairman of the Structural Mechanics Peer Review Panel for the TMI Vessel Investigation Project of the Three-Mile Island Accident
- Appointed by the NRC to serve on the Expert Elicitation Panel for NUREG-1150, which Updates the severe accidents report Wash-1400 Report
- Nominated by EPRI to NRC to serve on three Expert Panels for PIRT (Phenomena Identification and Ranking Table):
  - PWR and BWR Loss of Coolant Accident (LOCA)
  - PWR Reactivity Initiated Accident (RIA)
  - BWR Anticipated Transient Without Scram (ATWS)

Dr. Rashid began his career in the gas-cooled reactor industry at General Atomic (GA), where he developed the first application of large-scale three-dimensional finite element computation for High Temperature Gas Cooled Reactors (HTGRs) concrete pressure vessel, TRISO fuel particles and graphite moderator blocks. At GA he developed constitutive models for ceramic (brittle) materials, generalized as a common constitutive property for HTGR materials which included pyro-carbon, silicon carbide, graphite and concrete (pressure vessels). After seven years at GA, Dr. Rashid joined the Nuclear Energy Division of General Electric Company where he developed the industry's first two-dimensional fuel performance code for BWR application to pellet-clad interaction (PCI).

In 1978, Dr. Rashid founded ANATECH where he and his team developed unique capabilities in the analytical simulation of severe damage and failure of critical structures subjected to aircraft and missile impact, and failure analysis of reactor vessels and containments under beyond design basis accidents LOCA. Dr. Rashid received a service Recognition Award from EPRI for work on the Sandia/EPRI joint project on Reactor Containments behavior during severe accidents.

Through long-time support by the Electric Power Research Institute (EPRI), Dr. Rashid and his ANATECH team developed a series of codes for the modeling and simulation of LWR fuel performance throughout the fuel cycle from normal operations to the backend spent fuel storage and transportation. In the fuel performance area, Dr. Rashid is the principal author of the EPRI-Falcon fuel behavior code and its predecessor FREY code, and is a member of DOE-CASL's research team as the PCI Challenge Problem Integrator for the Bison fuel performance code.

In the spent fuel area, Dr. Rashid and his team made significant contributions in the behavioral modeling and analysis of the complex problems of dry storage and transportation, starting in 1990's with DOE's source term report SAND90-2406 for low-burnup fuel, followed by two decades of EPRI-supported research in high-burnup fuel documented in more than two dozen publications, of which a selected set is listed below of relevance to the PIRT process at hand.

SELECTED PUBLICATIONS

1. SANDIA REPORT SAND90-2406 • TTC-1019 • UC-820 • 1992, "A Method for Determining the Spent-Fuel Contribution to Transport Cask Containment Requirements", T. L. Sanders, et.al.
2. Joseph Y. R. Rashid and Albert Machiels, "PNNL Limiting Mechanisms for Long-Term Spent Fuel Dry Storage", 9th International Conference on Nuclear Engineering (ICONE9), April, 2001, Nice, France.
3. Creep Modeling and Analysis Methodology for Spent Fuel in Dry Storage, EPRI Report No. 100, November 2001, by Y. R. Rashid and R. S. Dunham.
4. Hydride Precipitation in Spent Fuel Cladding During Storage, Joe Rashid and Albert Machiels, Proceedings of The 10th International Conference on Environmental Remediation and Radioactive Waste Management, ICEM05-1038, 2005.
5. Results of a Multi-Year Study Aimed at the Resolution of Regulatory Issues Related to the Storage and Transportation of High-Burnup Spent Fuel, Joseph Rashid and Albert Machiels, Proceedings of The 11th International Conference on Environmental Remediation and Radioactive Waste Management, ICEM07-7244, 2007.
6. A Methodology for the Evaluation of Fuel Rod Failures Under Accident Conditions, Y. R. Rashid, A. J. Machiels, RAMTRANS, Vol. 16, No. 3, pp. 211-218 (2005).
7. Spent Fuel Transportation Applications: Global Forces Acting on Spent Fuel Rods and Deformation Patterns Resulting from Transportation Accidents, EPRI Report No. 1011817, December 2005, by Y. R. Rashid, R. S. Dunham, F. Wong, and R. J. James.
8. Spent-Fuel Transportation Applications – Normal Conditions of Transport, EPRI Report No. 1015049, June 2007, by Y. R. Rashid, F. Wong, and R. S. Dunham.
9. Spent-Fuel Transportation Applications: Modeling of Spent-Fuel Rod Transverse Tearing and Rod Breakage Resulting from Transportation Accidents, EPRI Report No. 1013447, October 2006, by Y. R. Rashid and R.S Dunham.

10. Spent Fuel Transportation Applications: Longitudinal Tearing Resulting from Transportation Accidents – A Probabilistic Treatment, EPRI Report No. 1013448, December 2006, by Y. R. Rashid, R. S. Dunham, Y. Zhang, and R. O. Montgomery.
11. Spent Fuel Transportation Applications: Fuel Rod Failure Evaluation Under Simulated Cask Side Drop Conditions, Report No. 1009929, June 2005, by R. J. James, Y. R. Rashid, R. S. Dunham, and L. Zhang.
12. Failure Criteria for Zircaloy Cladding Using a Damage-based Metal/Hydride Mixture Model, Report No. 1009693, December 2004, by Y. R. Rashid, M. Rashid, and R. S. Dunham.
13. Spent Fuel Transportation Applications – Assessment of Cladding Performance, A Synthesis Report, EPRI Report No. 1015048, December 2007, Y R Rashid
14. Development of a Metal/Hydride Mixture Model for Zircaloy Cladding with Mixed Hydride Structure, EPRI Report No. 1009694, June 2004, by Y. R. Rashid, M. M. Rashid, and R. S. Dunham.
15. Spent Fuel Issues - Assessment and Resolution, Proceedings of the 17th International Conference on Nuclear Engineering, ICONE17, July 12-16, 2009, Brussels, Belgium, Joe Rashid and Albert Machiels.
16. Simulation of Spent PWR Fuel Assembly Behavior Under Normal Conditions of Transport, IAEA International Conference on Spent Fuel Management, Vienna, May 28 to June 4, 2010, Joseph Y R Rashid and Albert J Machiels.
17. Joe Rashid and Albert Machiels, “A Sufficiency Criterion For Spent Fuel In Dry Storage Subjected To Transportation Accident Conditions”, International High-Level Radioactive Waste Management, Charleston, Sc, April 12-16, 2015

Ricardo Torres, Ph.D.

AFFILIATION

U.S. Nuclear Regulatory Commission

EDUCATION/TRAINING

Ph.D. Materials Science and Engineering, University of Florida, 2009
M.S. Materials Science and Engineering, University of Florida, 2005
B.S. Polymer Science and Engineering, Case Western Reserve University, 2003

PROFESSIONAL AFFILIATIONS

Member, ACI, ANS, and ASTM

PROFESSIONAL EXPERIENCE

Materials Engineer, U.S. Nuclear Regulatory Commission, 2013-Present

Review designs for domestic transportation packages and dry storage systems of radioactive material for compliance with the regulatory requirements in 10 CFR Part 71 and Part 72. Review designs for international transportation packages for compliance with IAEA requirements and revalidation of approvals from international competent authorities. Coauthor of various NRC technical reports and guidance documents on spent fuel performance and aging management/degradation modes of materials for use in continued dry storage. Lead the planning and implementation efforts on NRC-sponsored research on spent fuel cladding performance, accident tolerant fuel and non-light water reactor fuels

Senior Engineer, Savannah River Nuclear Solutions, 2010-2013

Principal Investigator for DOE-NNSA research projects on non-proliferation sensors and instrument demonstration/qualification for weapons complex support facilities. Provided engineering and research support to Savannah River Site projects on spent fuel recycling, environmental remediation of radioactive materials, process facility improvements for high level waste vitrification, and weld process characterization in aggressive environments.

Post-Doctoral Research Scientist, Savannah River National Laboratory, 2009-2010

Supported efforts to demonstrate viability of cascade impactors for size-separation of aerosolized environmental samples for homeland security applications.

Engineer, General Electric Company, Consumer & Industrial Lighting, 2003

Performed process capability studies and production qualifications of new phosphor materials for fluorescent lighting applications.

SELECTED PUBLICATIONS

1. R. Torres, G. Bjorkman, T. Ahn, F-C. Chang, A. Rigato, W. Reed, D. Tang, B. White, H. Akhavannik, and M. Rahimi, Review Guidance for the Safe Transport and Dry Storage of High Burnup Spent Nuclear Fuel, Proceedings of 2019 IAEA International Conference on the Management of Spent Fuel from Nuclear Power Reactors: Learning from the Past, Enabling the Future, IAEA-CN272-111.

2. J. Wise, R. Torres, K. Banovac, D. Dunn, and M. Davis, An Integrated Approach to Aging Management of Spent Fuel Dry Storage Systems in the United States, Proceedings of 2019 IAEA International Conference on the Management of Spent Fuel from Nuclear Power Reactors: Learning from the Past, Enabling the Future, IAEA-CN272-110.
3. P.K. Shukla, Y. Pan, R. Torres, T. Ahn, P. Raynaud, Assessment of Aging Mechanisms for Zirconium-Based High Burnup Fuel Cladding in Dry Storage Systems, CORROSION 2018 Proceedings, NACE-2017-8939.
4. R.D. Torres, M.J. Martínez-Rodríguez, J.R. Gray, P.S. Korinko, and T.M. Adams, Kinetics of Fission Product Fluorination with Sulfur Hexafluoride, Transactions of the American Nuclear Society, Vol. 107, 281-283 (2012).
5. R.D. Torres, J.R. Gray, M.J. Martínez-Rodríguez, P.S. Korinko, J.M. Becnel, B.L. García-Díaz, A.E. Visser, and T.M. Adams, Sulfur Hexafluoride Treatment of Used Nuclear Fuel to Enhance Separations, FCRD-SWF-2012-000110, DOE-NE Fuel Cycle Research & Development Campaign (2012).
6. R.D. Torres, L.T. Sexton, S.M. Serkiz, S.S. Jurisson, and C.R. Martin, Smart Nanophase Extractors for Tailored Fission Product Sequestration, Transactions of the American Nuclear Society, Vol. 107, 285-287 (2012).
7. J.I. Mickalonis and R.D. Torres, Evaluation and Recommendation of Saltstone Mixer Auger/Paddles Materials of Construction for Improved Wear Resistance, Savannah River Remediation, SRNL-STI-2012-00379 (2012).
8. R.D. Torres, J.R. Gray, P.S. Korinko, M.J. Martínez-Rodríguez, S.R. Sherman, B.L. García-Díaz, G.A. Morgan, A.E. Visser, and T.M. Adams, Reactive Gas Recycle Separation Technology for Used Nuclear Fuel, FCRD-SEPA-2011-000096, DOE-NE Fuel Cycle Research & Development Campaign (2011).
9. R.D. Torres, J.R. Gray, M.J. Martínez-Rodríguez, P.S. Korinko, B,L. García-Díaz, A,E. Visser, and T.M. Adams, Reactive Gas Recycle of Used Nuclear Fuel – Sulfur Hexafluoride, Transactions of the American Nuclear Society, Vol. 105, 217-218 (2011).
10. R.D. Torres, A.E. Méndez-Torres, and P-S Lam, Atomic Force Lithography of Nano/Microfluidic Channels for Verification and Monitoring of Aqueous Solutions, Proceedings of the 52nd Annual Meeting of the Institute of Nuclear Materials Management, SRNL-STI-2011-00351 (2011).
11. R.D. Torres, S.L. Johnson, R.F. Haglund, J. Hwang, and P.H. Holloway, Mechanisms of Resonant Infrared Matrix-Assisted Pulsed Laser Evaporation (RIR-MAPLE), Critical Reviews in Solid State and Materials Sciences, Vol. 36, Issue 1, 16-45 (2011).
12. R.D. Torres, J.S. Wright, B. Peters, Sensing Needs and Technology Assessment for SRS Tritium Facilities, NNSA Savannah River Tritium Enterprise, SRNL-TR-2012-00210 (2010).

PATENTS

R.D. Torres, L.T. Sexton, R.E. Fuentes, J. Cortes-Concepcion, Photonic crystal scintillators and methods of manufacture, Patent No. 9,103,921 (2015).

Albert J. Machiels, Ph.D.

Dr. Albert Machiels retired from the Electric Power Research Institute (EPRI) in June 2017. During his 35-year tenure at EPRI, his responsibilities included the oversight and/or management of several R&D programs in the following technical areas: Nuclear Fuel Industry Research (NFIR); Severe Accident Issue Resolution; Severe Accident Technology; Control and Diagnostics; Advanced Light Water Reactor (ALWR); Primary System Corrosion Research; PWR Materials Reliability Program (MRP); BWR Vessel Integrity Program (VIP); Risk and Reliability; Balance-of-Plant Corrosion; and Used Fuel and HLW Management. From 1996 to 2017, Dr. Machiels was actively engaged in topics related to spent fuel management (storage and transportation) and advanced fuel cycles, areas in which his contributions gained international recognition.

Dr. Machiels served as the EPRI executive liaison to the Nuclear Management and Resources Council (NUMARC, now part of the Nuclear Energy Institute, NEI) from August 1988 to October 1989.

Before joining EPRI, Dr. Machiels was a tenured, associate professor of nuclear engineering at the University of Illinois, Urbana-Champaign.

Prior to coming to the United States in 1970, Dr. Machiels was a lecturer at the University of Liège, Belgium.

Dr. Machiels received Ingénieur Civil Chimiste and Ingénieur en Génie Nucléaire degrees from the University of Liège, in Belgium; and a Ph.D. in Engineering from the University of California, Berkeley.

Dr. Machiels is the author of over 200 publications and technical contributions. He has managed and contributed to the publication of over 100 EPRI reports. He has served on several national and international panels and committees. In April 2012, Dr. Machiels received a Lifetime Achievement Award in recognition of his strategic and technical contributions to the nuclear generation of electricity. He was only the eighth individual (and the first one from the Nuclear Generation Sector) to receive such an award over the 40+ years of EPRI's existence.

Presently, Dr. Machiels is a part-time consultant on topics related to the management of spent nuclear fuel.

Storm Kauffman, P.E.

SUMMARY

Storm Kauffman heads MPR's Nuclear Technology area. He has over 45 years of multi-disciplinary experience in nuclear power, including engineering, design, construction, alternatives analysis, safety, operations, training, program and personnel management, emergency planning, decommissioning, security, and NRC design and licensing requirements. Mr. Kauffman joined MPR in mid-2008 after retiring as a Senior Executive from the DOE and Navy, having headed Reactor Safety, Reactor Engineering, and a reactor field office for the Naval Nuclear Propulsion Program (NNPP). Degrees: EMBA (Naval Postgraduate School) & MS in Nuclear Engineering and BS in Mechanical Engineering (MIT). While at MIT, he worked as an operator at the MIT Reactor and assisted with its conversion from a heavy water moderated and cooled design to its current light water moderated and cooled, heavy water reflected configuration.

EDUCATION/TRAINING

Naval Postgraduate School, Monterey, CA (distance learning) – EMBA, 2005

Massachusetts Institute of Technology, Cambridge, MA – MS, Nuclear Engineering, 1975

Massachusetts Institute of Technology, Cambridge, MA – BS, Mechanical Engineering, 1975

PROFESSIONAL AFFILIATIONS

Professional Engineer, Nuclear, VA (current)

Reactor Operator, MITR-I (expired)

EXPERIENCE

Safety Analysis

Directed Reactor Safety & Analysis for NNPP. Established requirements for safety for reactor operations, reactor refueling, and nuclear fuel facilities. Responsible for multimillion dollar test program for loss of coolant transients. Managed severe accident analysis and test program. Reviewed options for testing of fission product release from U-10Mo fuel. Managed update of ABWR Design Certification renewal probabilistic risk assessment, and justified activities in meetings with the NRC. Improved PRA techniques and conduct of PRAs.

Fuel & Core Design

Headed NNPP Reactor Engineering: provided technical & project management of fuel design, fabrication, qualification, & manufacturing, reactor thermal-hydraulic & structural design, and operation. Assessed design, qualification, and transient response of fuel for nuclear thermal propulsion and for HTGRs. Original lead for NuScale CRDM. Led R&D of advanced technology (methods, analysis, testing) in thermal-hydraulics, structural mechanics, advanced plant concepts (e.g., gas cycles), design by simulation, computational fluid dynamics, & computer code automation. Responsible for NRC-licensed Category I fuel facilities, including technical criteria and assessing safety/security regulations.

Alternatives Analysis & Advanced Reactors

Knowledgeable of various reactor technologies (current PWR & BWR, AP1000, ABWR, PBMR, MSR, HTGR, 4S). Developed for EPRI an advanced reactor guidance document. Prepared study of HTGR fuel qualification experience and needs. Assessed relative merits of high temperature reactors for process heat, including licensability, technical challenges, and project cost and schedule certainty. Directed down-select of advanced concepts, enabling initial R&D of supercritical CO2 plant design. Identified safety requirements for a Brayton cycle, fast reactor for NASA use for unmanned spacecraft propulsion.

Nuclear Regulation

Led and presented engineering/licensing response on key NRC issues for design certification for APR1400 and NuScale. Developed licensing strategy to deviate from certain General Design Criteria. Licensing advisor to HTGR and medical isotope production reactor start-ups. Assessed NRC safety requirements' applicability to naval designs and to a sodium fast reactor. Managed preparation of Safety Analysis Reports (SARs). Led Navy effort for NRC technical review of new shipboard reactor and developed review strategy for next design. Developed methodology for independent peer review for Canadian Nuclear Safety Commission (CNSC). Led international safety exchange requiring working knowledge of British reactor safety criteria. Testified to Defense Nuclear Facilities Safety Board. Developed regulatory and safety strategy for space reactor.

Deactivation and Decommissioning

As head of DOE field office at a training reactor site, oversaw safety of operations and adequacy of training. Planned and drove start of D&D work that led to site free release. Monitored radiological, environmental, and cost performance. Member of independent reviews for DOE waste project.

Criticality Analysis

Assessed criticality control for reprocessing of high-enriched fuel. Led team performing new fuel shipping package analysis. Directed criticality analysis for new/spent reactor fuel handling, disposal of fuel in a geologic repository, and lab work. Established criticality safety requirements for new activities and audited compliance.

Yucca Mountain License Application (LA)

Managed creation of NNPP supplement to LA for pre- and post-closure and briefed NRC staff. Reviewed and critiqued DOE LA. Responsible for both technical content and presentation.

Fuel/Radioactive Material Shipping Licensing

Developed and led criticality safety approach for new fuel package. Managed preparation, submittal, & comment resolution for reactor compartment disposal Safety Analysis Reports for Packaging (SARPs) and for naval spent fuel Technical Support Document for disposal in the geologic repository. Assessed plans for Independent Spent Fuel Storage Installation for San Onofre. Performed independent review of criticality, thermal, shielding, and structural analyses for new and spent fuel containers for storage and shipment. Safety and security lead for naval fuel shipment and storage; performed physical security inspections.

Fuel Handling

Evaluated design of equipment and procedures for fuel load, refueling, spent fuel storage, loading for dry storage, and transport. Advised Yucca Mt. project regarding suitability of design of surface facilities to be used for repacking of spent fuel. Managed seismic analysis of fuel storage racks.

Probabilistic Risk Assessment (PRA)

Managed reconstitution/update of ABWR Design Certification renewal PRA, and justified activities in meetings with the NRC. Improved PRA techniques and conduct of PRAs. Independently reviewed foreign three level PRA. Established safety related valve reliability statistics. Led application of PRA methods to Yucca Mt. post-closure criticality analysis. Assisted in probabilistic assessment of safety, operational reliability, and performance. Attended MIT Risk-Informed Operational Decision Management & NRC PRA courses and ANS PRA topical meetings. Taught in-house classes on PRA. Developed plan to assess risk of inadvertent safety feature actuation, reliability of crossing site power lines, and means to reduce risk importance of specific components.

SELECTED PUBLICATIONS

1. S. Kauffman, M. Charrouf, J. Yoon, S. Lee, T. Kim, “Evaluation Of Potential for High Energy Line Break Jet Dynamic Loading Including Feedback Amplification and Resonance Effects,” ICONE 27, May 2019.
2. M. Charrouf, S. Kauffman, J. Yoon, S. Lee, T. Kim, “Blast Wave Analysis of High Energy Line Break,” ICONE 27, May 2019.
3. J. S. Newman, S. Kauffman, A. Ford, “Naval Reactors Safety Assurance,” NASA/Navy Benchmarking Exchange (NNBE), Volume II, July 15, 2003.

Valentina Angelici Avincola, Ph.D.

SUMMARY

Dr. Angelici Avincola joined MPR Associates in 2018. Prior to joining MPR, Angelici Avincola developed expertise in experimental work and data acquisition testing materials related to nuclear applications under high-temperature conditions. During her time at MPR, she has participated in a variety of projects for the power and health science industries. Her experience includes performing material evaluations and third-party reviews, thermal-hydraulic analyses of piping systems, and thermal analysis.

EDUCATION/TRAINING

Karlsruhe Institute of Technology Institute for Applied Materials, Ph.D.,
Mechanical Engineering, 2016

Sapienza University of Rome Department of Nuclear Engineering, M.S.,
Nuclear Engineering, 2012

Sapienza University of Rome Department of Energy Engineering, B.S.,
Energy Engineering, 2009

PROFESSIONAL AFFILIATIONS

American Nuclear Society (ANS), Member since 2012

International Youth Nuclear Congress (IYNC), Member of the board of directors since 2016

EXPERIENCE

Accident Tolerant Fuels Performance, Karlsruhe Institute of Technology / Massachusetts Institute of Technology

Performed research on accident tolerant fuel cladding (metallic and ceramic materials) for nuclear power plants using both experimental and computational techniques, assessing the performance of materials under accident conditions. Led a research group to develop a protocol to test advanced reactors materials and developed technical procedures for test programs performed by third-party labs.

Materials Testing and Analysis

Gained significant experience with testing and analysis of engineering materials, including ceramics composites, alloy steel, graphite, specialty alloys. Works performed at MPR include the evaluation of the corrosion rate of venting piping affected by galvanic and aqueous corrosion and the evaluation of the performance of materials selected for the main turbine in a supercritical CO_2 plant. Results were used to assess the conditions of degraded components during inspection.

Spent Nuclear Fuel

Performed project related to nuclear waste canisters, contributing to a white paper focused on FEA modeling of canisters scratches, and evaluating the feasibility of overlay repair for mitigation of stress corrosion cracking on canisters.

Participated in a phenomena identification ranking table (PIRT) process on fuel cladding performance organized by EPRI.

Advanced Nuclear Reactors

Contributed to the preparation of the Owner-Operator Requirement Guide (ORG) for Advanced Reactors published by the Electric Power Research Institute (EPRI) to provide guidance to advanced reactor owner-operators, vendors, regulators, and other stakeholders.

Prepared an EPRI report on advanced security technologies available to improve physical security of small modular reactors.

Characterization of Coatings for Superalloys, University of Virginia

Developed the set-up of a low-velocity burner rig for testing superalloys and coatings in corrosive environments. She developed approaches to test and characterize different materials to advance critical testing protocols needed for validation of gas turbine materials and prepared reports for submittal to the US Navy describing the test programs. Results were used to support material evaluations to achieve longer hot section lives and higher service operating temperatures in future marine operations.

Microscopy Techniques, Massachusetts Institute of Technology / University of Virginia

Performed extensive materials characterization using scanning electron microscopy (SEM), energy dispersive spectroscopy (EDS), electron backscatter diffraction (EBSD), Raman spectroscopy, X-ray photoelectron spectroscopy (XPS), X-ray powder diffraction (XRD), and transmission electron microscopy (TEM).

SELECTED PUBLICATIONS

1. V. Angelici Avincola, E. Opila, High-temperature oxidation of BN-coated SiC Sylramic fibers in dry and wet atmospheres, Materials Science and Technology Conference 2017, Pittsburgh, PA, USA, October 8-12, 2017
2. V. Angelici Avincola, K. Fitzgerald, D. Shepherd, D. Kinay, C. Sauder, M. Steinbrück, High-temperature tests of silicon carbide composite cladding under GFR conditions, Energy Procedia, 127, pp320-328, 2016
3. V. Angelici Avincola, P. Guenoun, K. Shirvan, Mechanical performance of SiC multi-layer cladding in PWRs, Nuclear Engineering and Design, 35, pp 280-294, 2016
4. V. Angelici Avincola, D. Cupid, H.J. Seifert, Thermodynamic modeling of the silica volatilization in steam related to silicon carbide oxidation, Journal of the European Ceramic Society, 35(14), pp. 3809-3818, 2015
5. V. Angelici Avincola, M. Grosse, M. Steinbrück, Oxidation at high temperatures in steam atmosphere and quench of silicon carbide composites for nuclear application, Nuclear Engineering and Design, 295, pp. 468-478, 2015
6. V. Angelici Avincola, U. Stegmaier, M. Steinbrück, H.J. Seifert, Oxidation of tantalum at high temperatures in steam atmosphere, Oxidation of Metals, 5(85), pp 459-487, 2015

Cécile Dame, Ph.D.

SUMMARY

Dr. Dame has over 15 years of experience with the nuclear industry, as a researcher, an engineer and a project manager. Prior to joining MPR in 2019, she performed in-depth research into corrosion, primary and secondary water chemistry, fuel integrity, decontamination, and radiation generation issues specific to the nuclear industry. Dr. Dame is also an expert in major component maintenance activities and has a proven track record of managing complex projects for the nuclear industry.

EDUCATION/TRAINING

2006 Ph.D. in Chemical Engineering, Commissariat à l'Energie Atomique (CEA) – Université de Marseille III

2002 M.S. in Process Engineering, Université Montpellier II / Ecole Nationale Supérieure de Chimie de Montpellier (France)

2001 B.S. in Physical Chemistry, Université Montpellier II (France)

EXPERIENCE

Pressurized Water Reactor primary and secondary water chemistry management

- Evaluated primary and secondary operations related to chemistry issues
- Performed in-depth research related to new additives and contaminants
- Modeled low flow and obstructed regions chemistry
- Developed training materials

Corrosion

- Modeled coolant system corrosion and corrosion products deposition in nuclear systems
- Studied the corrosive effects of steam generator deposits
- Proposed new additives to mitigate stress corrosion cracking

Fuel integrity

- Provided site specific consulting on fuel degradation mechanisms
- Participated in the revision of the EPRI Fuel Cladding Corrosion and Crud guidelines
- Evaluated plant operation impacts in terms of fuel integrity

Decontamination and decommissioning

- Developed a decontamination process using foam
- Provided site specific consulting
- Evaluated decontamination processes
- Advised for decommissioning preparation

SELECTED PUBLICATIONS

1. K. Fruzzetti, D. Wells, D. Morgan, J. Giannelli, C.R. Marks, C. Dame, Optimization of Fuel Deposit Properties for Reducing Plant Radiation Fields an Assessment, in “NPC 2014 - Nuclear Plant Chemistry Conference 2014 Proceedings,” held in Sapporo, Japan, Oct 26-31, 2014.
2. R. Reid, K. Fruzzetti, A. Ahluwalia, A. Summe, C. Dame, K.P. Schmitt, Modelling of the Local Chemistry in Stagnant Areas in the PWR Primary Circuit, in “NPC 2014 - Nuclear Plant Chemistry Conference 2014 Proceedings,” held in Sapporo, Japan, Oct 26-31, 2014.
3. C. Dame, C.R. Marks, A.R. (Jenks) Olender, J. Farias, Evaluation of Chemical Surface Treatment Methods for Mitigation of PWSCC, in “Contribution of Materials Investigations & Operating Experience to LWRs’ Safety, Performance, & Reliability. Proceedings of the International Symposium, Fontevraud 8,” held in Avignon, France, September 15-18, 2014.
4. C. Dame, Chemical Cleaning Experience in France Using the EPRI/SGOG Process; Expectations and Performances, 2nd SGMP 2010 Steam Generator Secondary Side Management Conference, March 2-4, 2010, in San Antonio, Texas.
5. C. Dame, S. Faure, F. Boué, A. Steinchen, Foam stability investigation through three – scale study and related parameters analysis, EUFOAM 2006 – 6th European Conference on Foams, Emulsions and Applications, July 3rd-6th, 2006, Postdam (Germany). (Prize for top scientific research)
6. C. Dame, J.L. Alcaraz, S. Faure, Viscosified Nuclear Decontamination Foams, ICEM’05 - 10th International Conference on Environmental Remediation and Radioactive Waste Management, September 4-8, 2005, Glasgow (Scotland).
7. *Minimization of Pressurized Water Reactor Radiation Fields through Fuel Deposit Engineering: Deposit Property Evaluation and Optimization*. EPRI, Palo Alto, CA. 2013. 3002000584.
8. *An Assessment of PWR Water Chemistry Control in Advanced Light Water Reactors: APR1400: Volume 1: PWR Primary Water Chemistry*. EPRI, Palo Alto, CA. 2012. 1026540.

Export Control Restrictions

Access to and use of this EPRI product is granted with the specific understanding and requirement that responsibility for ensuring full compliance with all applicable U.S. and foreign export laws and regulations is being undertaken by you and your company. This includes an obligation to ensure that any individual receiving access hereunder who is not a U.S. citizen or U.S. permanent resident is permitted access under applicable U.S. and foreign export laws and regulations.

In the event you are uncertain whether you or your company may lawfully obtain access to this EPRI product, you acknowledge that it is your obligation to consult with your company's legal counsel to determine whether this access is lawful. Although EPRI may make available on a case by case basis an informal assessment of the applicable U.S. export classification for specific EPRI products, you and your company acknowledge that this assessment is solely for informational purposes and not for reliance purposes.

Your obligations regarding U.S. export control requirements apply during and after you and your company's engagement with EPRI. To be clear, the obligations continue after your retirement or other departure from your company, and include any knowledge retained after gaining access to EPRI products.

You and your company understand and acknowledge your obligations to make a prompt report to EPRI and the appropriate authorities regarding any access to or use of this EPRI product hereunder that may be in violation of applicable U.S. or foreign export laws or regulations.

The Electric Power Research Institute, Inc. (EPRI, www.epri.com) conducts research and development relating to the generation, delivery and use of electricity for the benefit of the public. An independent, nonprofit organization, EPRI brings together its scientists and engineers as well as experts from academia and industry to help address challenges in electricity, including reliability, efficiency, affordability, health, safety and the environment. EPRI also provides technology, policy and economic analyses to drive long-range research and development planning, and supports research in emerging technologies. EPRI members represent 90% of the electricity generated and delivered in the United States with international participation extending to nearly 40 countries. EPRI's principal offices and laboratories are located in Palo Alto, Calif.; Charlotte, N.C.; Knoxville, Tenn.; Dallas, Texas; Lenox, Mass.; and Washington, D.C.

Together...Shaping the Future of Electricity

Programs:

Nuclear Power

Used Fuel and High-Level Waste

3002018439

Electric Power Research Institute
3420 Hillview Avenue, Palo Alto, California 94304-1338 • PO Box 10412, Palo Alto, California 94303-0813 USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com