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Phenomena Identification and Ranking Table Exercise for Spent Tri-Structural Isotropic Particles in Storage and Transportation

Phenomena Identification and Ranking Table Exercise for Spent Tri-Structural Isotropic Particles in Storage and Transportation

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ABSTRACT

The increasing global focus on advanced reactor designs has significantly heightened interest in TRISO fuel, a high-performance, robust fuel type known for its enhanced safety features and ability to withstand extreme conditions. However, a deeper understanding is needed regarding whether TRISO fuel can be credited as functional containment during transportation and storage conditions.

This topic was the subject of the Phenomena Identification and Ranking Table (PIRT) process summarized in this report. In this work, a panel of world-wide recognized experts evaluated the state of knowledge and the significance of phenomena relevant to TRISO fuel during transportation and storage conditions under different scenarios. As a result of this process, the panel concluded that only one scenario has a high significance ranking: hypothetical accident conditions during transportation ranked high for both the matrix fracture and neutron multiplication phenomena. All other scenarios were ranked as medium or low significance. No regulatory positions are taken in this document by the panel members.

Keywords

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PRIMARY AUDIENCE: Regulators, cask designers, cask users, reactor developers

SECONDARY AUDIENCE: Government and research institutions, fuel fabricators

KEY RESEARCH QUESTION

Existing phenomenon identification and ranking table exercises for TRISO fuel focus on in-core conditions, excluding phenomena relevant during storage and transportation conditions. Considering the current state of knowledge, can TRISO fuel be credited as functional containment during transportation and storage conditions?

RESEARCH OVERVIEW

A panel of experts on TRISO fuel, with experience in thermal performance, criticality, and radiological performance, was assembled to evaluate the current state of knowledge and evaluate if TRISO fuel can be credited as functional containment during storage and transportation scenarios. This report documents the deliberations of the expert panel using the PIRT process to evaluate the knowledgebase and determine the safety significance of phenomena associated with TRISO fuel during storage and transportation. These phenomena were evaluated in six scenarios related to storage and two scenarios related to transportation.

KEY FINDINGS

- The TRISO barriers to radiological release and dose consequence can be credited, but the extent to which they can be credited needs to take the design of the storage and transportation packages into consideration.
- The existing practices used for storage and transportation of commercial LWR SNF (i.e., leak-tight cask, providing containment/confinement in all scenarios) are compatible with TRISO fuels. Additional analytical and/or experimental work is likely required to evaluate TRISO performance under transportation accident conditions.
- TRISO properties (e.g., mechanical properties and thermal management) may enable novel storage and transportation designs:
    - Confinement requirements may be achieved in a different manner for storage.
    - The lower energy density allows design requirements for thermal management during transportation to be reconsidered.
- Guidance for spent fuel could be updated to reflect the unique attributes of TRISO fuel.

WHY THIS MATTERS

The ability to credit TRISO fuel as functional containment during transportation and storage scenarios may allow for novel transportation and storage planning and system design.

HOW TO APPLY RESULTS

Reactor designs and cask designs may use this report as a basis to enable novel storage and transportation designs. Additionally, regulators may use this report when updating existing spent fuel guidance to include TRISO fuel.

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ACRONYMS

ANL	Argonne National Laboratory
AGC	Advanced graphite creep
AGR	Advanced gas reactor
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
AVR	Arbeitsgemeinschaft Versuchsreaktor
BLH	Boltzmann-enhanced Langmuir-Hinshelwood (model)
CFR	Code of Federal Regulations
CoC	Certificate of Compliance
DOE	Department of Energy
dpa	Displacements per atom
EPRI	Electric Power Research Institute
FP	fission products
HAC	hypothetical accident conditions
HALEU	high assay low enriched uranium
HTGR	high temperature gas-cooled reactor
HTR	high temperature reactor
IAEA	International Atomic Energy Agency
INL	Idaho National Laboratory
IPyC	inner pyrolytic carbon
ISG	Interim Staff Guidance
LBL	leach-burn-leach
LH	Langmuir-Hinshelwood (model)
LMNT	LMNT Consulting

LT	long term (storage)
LWR	light water reactor
MCNP	Monte Carlo N-particle transport
MOOSE	object-oriented simulation environment
MPR	MPR Associates
MSR	molten salt reactor
NCT	normal conditions of transport
NEI	Nuclear Energy Institute
NGNP	Next Generation Nuclear Plant
NP	nuclear plant
NRC	U.S. Nuclear Regulatory Commission
NUREG	U.S. Nuclear Regulatory Commission technical report designation
OE	operating experience
ON	off-normal
OPyC	outer pyrolytic carbon
ORNL	Oak Ridge National Laboratory
PARFUME	PARticle FUel ModEl
PBMM	pebble bed micro model
PBMR	pebble bed modular reactor
PBR	pebble bed reactor
PIE	post-irradiation examination
PIRT	phenomena identification and ranking table
PNNL	Pacific Northwest National Laboratory
PWR	pressurized water reactor
PyC	pyrolytic carbon
QC	quality control
SiC	silicon carbide
SME	subject matter experts
SNF	spent nuclear fuel
SNL	Sandia National Laboratories
SSC	structures, systems, and components

ST	short-term (storage)
TC	thermocouple
TECDOC	technical document produced by the IAEA
THTR	thorium high temperature reactor
TRISO	tristructural isotropic
UCO	uranium oxycarbide
US/USA	United States/United States of America
VHTR	very high temperature reactor
VHTRC	very high temperature reactor critical assembly

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1
INTRODUCTION

This report presents the results of a Phenomena Identification and Ranking Table (PIRT) exercise that took place in 2023 to evaluate the possibility of crediting Tristructural Isotropic (TRISO) particles as barriers to radiological release and dose consequence during spent fuel storage and transportation activities. This exercise is intended to build upon a previous EPRI-led topical report on TRISO particle performance from 2019, presented in Reference [1] and evaluated by the NRC in Reference [2], by considering transportation and storage scenarios. Meetings related to this PIRT were held in the EPRI office, located in Washington, DC, and at the MPR Associates, Inc. office located in Alexandria, Virginia, facilitated by EPRI and MPR Associates, Inc.

Background

TRISO fuel is a form of nuclear fuel known for its exceptional retention of radionuclides and structural integrity at high temperatures in nuclear reactors. The acronym TRISO refers to the three layers of coatings (TRi-structural ISOtropic) that encapsulate the uranium-based spherical fuel kernel with a porous carbon buffer region; a dense inner layer of pyrolytic carbon (IPyC), a layer of silicon carbide (SiC), and an outer layer of PyC (OPyC). These coated fuel kernels, or TRISO particles, are generally embedded in a graphite or carbonaceous matrix formed in the shape of a sphere or cylinder, also known as a pebble or compact. Compacts are then set in larger prismatic graphite blocks. An emerging variation on this fuel form uses SiC as the matrix material for cylindrical pellets. The structure of TRISO particles is shown in Figure 1-1, along with cylindrical and spherical fuel forms.

One of the primary advantages of TRISO fuel is the ability of the robust layers of carbon and silicon carbide to act as a barrier to radiological release through prevention of the release of fission products. Additionally, TRISO fuel has a very high temperature tolerance, as the SiC layer can withstand temperatures up to 1600°C [3]. This providing considerable safety margins and improves reactor efficiency. This stability is instrumental in preventing fuel particle failure, especially during transient events or accidents. This inherent safety feature makes TRISO fuel a compelling choice for advanced reactor designs, particularly high-temperature reactors that employ passive safety features. This includes both high temperature gas reactors (HTGRs) and Fluoride-Salt-Cooled High-Temperature reactors (FHRs).

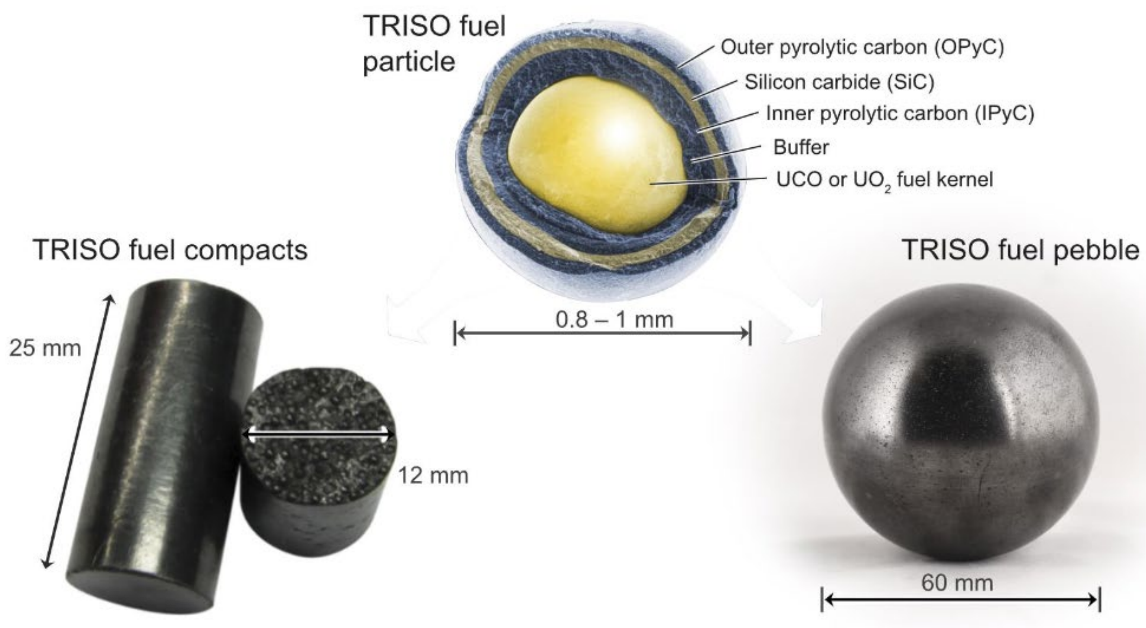

Figure 1-1
Structure of TRISO particles, fuel compacts, and fuel pebbles from Reference [4]

Summary of Previous Relevant Work

Previous PIRT

The NRC previously conducted a PIRT in 2004 [5], focused on TRISO fuel, with the following objectives:

- Identify key attributes of gas reactor fuel manufacture which may require regulatory oversight,
- Provide a valuable reference for the review of gas-cooled reactor fuel qualification plans,
- Provide insights for developing plans for fuel safety margin testing,
- Assist in defining test data needs for the development of fuel performance and fission product transport models,
- Inform decisions regarding the development of the NRC's independent gas-cooled reactor fuel performance code and fission product transport models,
- Support the development of the NRC's independent models for source term calculations,
- Provide insight for the review of vendor gas-cooled fuel safety analysis.

That PIRT focused on six areas: manufacturing, operations, depressurized heat-up accidents, reactivity accidents, depressurization accidents with water ingress, and depressurization accidents with air ingress. Phenomena associated with air and water interactions and their effects on fission products, reaction kinetics, and temperature distributions were generally ranked highly, and more research was recommended. Transport and storage were not discussed.

Topical Report

The Electric Power Research Institute (EPRI) published a topical report [1] on Uranium Oxycarbide (UCO) TRISO particle performance, which provides the technical bases for functional performance of these particles based on the advanced gas reactor (AGR) experimental campaign. Reference [1] used data from the AGR-1 and AGR-2 experiments, the first two of the four irradiation campaigns completed as part of the AGR program. The AGR program studied five major elements: fuel fabrication, fuel and material irradiation, fuel post-irradiation examination (PIE) and safety testing, fuel performance modeling, and fission product transport and source term development.

Reference [1] had three main conclusions:

1. Testing of UCO TRISO fuel particles conducted in the AGR-1 and AGR-2 experiments demonstrate performance of these particle designs over a range of normal operating temperatures and off-normal accident conditions. Therefore, the testing provides a foundational basis for use of these particles in fuel elements of TRISO-fueled HTGR designs.
2. The kernels and coatings of UCO TRISO particles tested in AGR-1 and AGR-2 exhibited a variation in properties, as they were fabricated under different conditions and at different scales. However, they had similar excellent performance in irradiation and accident conditions. Therefore, UCO TRISO fuel particles that satisfy the parameter envelope defined by the particle properties measured in the AGR program can be relied upon to provide satisfactory performance.
3. Aggregate AGR-1 and AGR-2 data on fission product release and fuel failure fractions can be used to support licensing of reactors employing UCO TRISO fuel particles that satisfy the parameter envelope defined by measured particle layer properties from AGR-1 and AGR-2.

The conclusions of this topical report were accepted by the NRC with certain conditions and limitations as noted in its safety evaluation report [2].

Relevant Regulations and Guidance

Regulations

Regulations for the transportation and storage of spent nuclear fuel are documented in 10 CFR 71 and 10 CFR 72, respectively. The following sections provide a high-level overview of these regulations.

10 CFR 71

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 hypothetical accident conditions in 10 CFR Part 71.73.

NRC regulations found in 10 CFR 71 are largely compatible with the International Atomic Energy Agency (IAEA) regulations [6] This assessment was conducted on a prior revision of the regulation**.** A more current compatibility assessment has not been conducted, but would be a valuable exercise.

10 CFR 72

10 CFR Part 72, titled “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, to provide reasonable assurance that cooling and subcriticality are maintained. Additionally, 10 CFR Part 72.122(l) requires that spent nuclear fuel (SNF) be designed to allow ready retrieval of spent fuel without risk to the public for further processing or disposal.

NRC Safety Review Guidance

NUREG-2216 “Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material”, Reference [7]

NUREG-2216 provides guidance to the NRC staff for reviewing an application for package approval issued under 10 CFR 71. This document identifies acceptable approaches to meeting regulatory requirements and possible evaluation findings that can be used in a safety evaluation report, including structural, thermal, containment, shielding, criticality, and materials evaluations. This document also includes a description of the NRC review procedure and approach for issuing a certificate of compliance (CoC).

NUREG-2215 “Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities”, Reference [8]

NUREG-2215 provides guidance to the NRC staff for reviewing safety analysis reports to issue a CoC for a dry storage system or a license for a dry storage facility. As with NUREG-2216, this document identifies acceptable approaches to meeting regulatory requirements and possible evaluation findings that can be used in a safety evaluation report, including structural, thermal, containment, shielding, criticality, and materials evaluations. This document does not apply to any wet storage facilities.

PIRT Process

A PIRT process is divided into nine steps according to the NRC [9], which are the following:

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 of 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 required to be documented in the PIRT report 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.

Objective and Report Organization

The objective of this PIRT is to evaluate if the barriers to radiological release and dose consequence of a TRISO particle can be credited for storage and transportation activities. This objective has been 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 level of understanding in terms of available data, and 3) the quality of the data available.

The expert panel members are listed in Table 1-1 and their resumes are provided in Appendix A.

Table 1-1
TRISO performance expert panel

Name	Affiliation
Andrew Barto	Nuclear Regulatory Commission (NRC)
Jason Piotter	Nuclear Regulatory Commission (NRC)
Harold Adkins	Pacific Northwest National Laboratory (PNNL)
Gordon Petersen	Idaho National Laboratory (INL)
Paul Demkowicz	Idaho National Laboratory (INL)
Jim Kinsey	Idaho National Laboratory (INL)
Steve Nesbit	LMNT Consulting (LMNT)
Finis Southworth	Consultant

The following information in this report is organized following the NRC PIRT process description discussed above.

Section 2 discusses the PIRT bases, including scenarios, evaluation criterion, and phenomena considered for this exercise (Steps 1 through 4 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 5 through 9 of the PIRT process).

Section 4 provides the results of the process, summarizes the current state of knowledge, and presents future opportunities for changes in guidance.

2
PIRT BASES

This chapter discusses Steps 1 through 4 of the NRC PIRT process, including the definition of the problem statement, evaluated hardware, initial conditions, evaluation criterion, and assessed scenarios.

PIRT Problem Statement

There is growing interest among advanced reactor developers in crediting the ‘functional containment’ of a TRISO particle for storage and transportation activities. Functional containment under reactor operating conditions has been addressed in documents such as the EPRI TRISO Topical Report 30021015750 [1], which received an NRC safety evaluation in 2021. Extension of functional containment to storage and transportation might enable optimization of those technologies, potentially leading to reduced costs. This PIRT assesses the gaps in crediting TRISO functional containment in transportation and storage scenarios.

Definition of Evaluated Hardware

The evaluation conducted in this PIRT process focused on the following hardware:

- TRISO particles, including all constituent layers defined in Section 1.
- Matrix material surrounding the TRISO particles, which may be graphitic or silicon carbide.
- Prismatic graphite blocks (also referred to as elements), which contain TRISO particle compacts (described in Section 1). These are not applicable for all reactor designs and specific types or grades of graphite were not considered.

In addition, the interactions of these items with spent fuel casks were considered using existing cask designs as a reference. Detailed transportation and storage plans, including the design of future spent fuel casks, were unknown for the purposes of this PIRT. As these plans advance, these interactions should be revisited to ensure the evaluated phenomena (described below) are not affected by new designs.

Evaluation Criterion

The evaluation criterion (i.e., the specific goal to be achieved in analyzing each scenario) selected by the panel was to prevent unacceptable radiological release of radionuclides from the spent TRISO fuel in conjunction with the storage and transportation system. When evaluating the different phenomena against the evaluation criterion, the panel considered the maintenance of subcriticality, shielding, containment/confinement, and thermal-management performance.

The panel noted that, due to the nature of the TRISO manufacturing process, there will be a very small fraction of defective TRISO particles before irradiation and additional particles that fail after irradiation. This is expected and is distinct from the phenomena evaluated in this effort.

The following assumptions were made as part of the evaluation process:

- Activities to reduce the volume of waste, referred to throughout this report as waste volume reduction, such as removing spent fuel compacts from the graphite blocks, were not considered.
- Cask integrity was assumed to be maintained throughout its lifespan via an aging management program.
- Fuel was handled by the plant's standard means of operation, using the plant's standard methodology and equipment. Fuel that could not be handled by these standard means was not considered (similar to current LWR fuel assemblies with gross rupture).

Definition of Scenarios

Scenarios were used to evaluate variations in the fuel system's 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, two scenarios for loading and unloading activities, and two scenarios for transportation, as described below:

- **Short-term loading (ST-Loading) activities:** incudes all loading, transfer, and container closure activities up to placement of container in the expected storage configuration. For light water reactors these activities typically take a few days for each container.
- **Storage - Normal (Storage-Normal):** includes normal conditions of dry storage with a duration of up to 60 years in dry storage systems.
- **Storage - Long-Term (Storage-LT):** considers normal conditions of dry storage for a duration of between 60 and 100 years in one of the dry storage systems. This considers the fact that aging management programs could be revised with time to reflect storage system needs.
- **Storage - Off-Normal (Storage-ON):** considers events that occur infrequently during dry storage, although a specific limit on the number of occurrences is not identified. The panel assumed a duration of the ON condition of up to 72 hours.
- **Storage- Accident** (**Storage-Accident**)**:** accident conditions during storage as listed in Chapter 16 of [8], including storage container tip over, storage container drop, flood, fire and explosion, and earthquake.
- **Unloading activities (Unloading):** considers the opening of a storage system/transportation package and the removal of prismatic blocks or pebbles. A time limit for when this operation could happen was not applied, meaning that it could happen many years after storage. It is assumed that the SNF is not being dropped during handling activities.

- **Normal Conditions of Transport (NCT):** considers the conditions defined in 10 CFR Part 71.71. In this scenario, the package is considered to be in its transportation configuration (i.e., with impact limiters installed). A 0.3 m (1 ft) drop is an example of a condition evaluated for NCT.
- **Hypothetical Accident Conditions (HAC):** hypothetical accident conditions during transportation as defined in 10 CFR 71.73. These consist of free drop, crush, puncture, thermal, and immersion.

Ranking Rules

During this PIRT exercise, the panel utilized the following ranking scale to determine the importance of each phenomenon and the ability to satisfy the evaluation criterion:

- **Operability:** Does the phenomenon occur or do the effects of the phenomenon impact fuel system performance in the scenario considered? The answer is yes or no (Y/N). Phenomena that were deemed non-operable were not ranked on subsequent criteria.
- **Knowledge:** Is data on the phenomenon available, and is it relevant? The ranking is low, medium, or high (L, M, H), with low assigned when a minimal amount of data is available, and high assigned when a satisfactory amount of data is available.
- **Confidence:** What is the quality of the existing data and models (e.g., is it consistent, can the data be modeled, verified, or 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. When the knowledge was ranked low, the confidence was not ranked.
- **Significance:** To what extent does the phenomenon contribute to a release of radionuclides that exceeds acceptance criterion? The ranking is low, medium, or high (L, M, H), with low assigned when the phenomenon has a low likelihood of radiological release that exceeds acceptance criterion and high when the event has high likelihood of radiological release that exceeds acceptance criterion.

Phenomena Considered During the PIRT

The phenomena the panel evaluated that could potentially affect the integrity of the spent TRISO fuel in this PIRT are listed in Table 2-1. Detailed definitions for each phenomenon are provided in Section 3.

Table 2-1
Phenomena considered within this PIRT exercise

Phenomena	Section
Matrix fracture	3.1
Non-fuel block fracture	3.2
Abrasive wear	3.3
TRISO particle layer fracture	3.4
PyC creep	3.5
SiC corrosion	3.6
Particle, block, and matrix oxidation	3.7
Helium pressurization (alpha decay)	3.8
Fission product leaching	3.9
Fission product diffusion	3.10
Neutron multiplication	3.11
Decay heat	3.12

3 DISCUSSION AND RESULTS

This section documents steps 5 through 9 of the PIRT process. The panelists evaluated phenomena that could affect the integrity of TRISO particles in a spent fuel storage system/transportation package. Below are the panel's ranking and rationale for each phenomenon. If a phenomenon was deemed to not be operable for a scenario, no rankings were performed for that scenario.

Matrix Fracture

Phenomenon Description

This phenomenon refers to the fracture of matrix material that maintains the geometry of the fuel form containing the TRISO particles, described in Section 1. TRISO particles are robust barriers to the fission products (FP) that are produced in the kernel. However, radionuclides are expected to be present outside of the TRISO particles in very small quantities as a result of:

1. Tramp uranium
2. Manufacturing defects or radiation-induced failure of the SiC layer
3. Diffusion of fission products (such as Ag) through intact coating layers, and/or
4. Activation of the matrix material

Therefore, a fracture in the matrix material may provide a pathway for these radionuclides to escape the matrix. In addition, matrix fracture may lead to exposed TRISO particles or material relocation, which when combined with other phenomena, may challenge the evaluation criterion defined in Section 2. No constraints were applied to the size of the fracture.

Available Data and Quality

There is enough data to model graphite matrix behavior in the conditions that are relevant to this PIRT for some grades of graphite. Reference [4] documents mechanical models for TRISO fuel with graphite matrix material. This model is based on a Weibull fit of mean stress at failure from publicly available data. Using this model, the probability of failure of the matrix can be calculated based on the applied stresses. Individual TRISO particles are significantly stronger than the surrounding graphite matrix, therefore the matrix is the more likely failure point. The authors claim that using this data, the probability of failure can be significantly mitigated via design to lessen the likelihood and credibility of a fracture.

The literature base for SiC matrix material is sparse and detailed fracture properties are not available. Therefore, the rankings below only apply to fuels with graphitic matrices.

With respect to previous operational experience (OE), there are no records from the Fort. St. Vrain fuel or the German AVR and THTR reactors that indicate degradation or damage to the fuel pebbles or blocks occurring during SNF storage and transportation activities [10].

Ranking

The summary of the matrix fracture ranking is presented in Table 3-1. Operability for this phenomenon was determined by considering if there was the potential for a significant mechanical shock to fracture the matrix during each scenario. Thus, storage, long-term storage, and storage off-normal conditions were ranked as not operable. The Storage-Accident scenario was marked operable due to earthquake tip over, but the panel noted that this scenario could be mitigated by design.

The knowledge was ranked as medium for ST-Loading, Storage-Accident, and Unloading scenarios due to a combination of available data and previous OE for similar, but not identical, systems. The knowledge was ranked as low for NCT and HAC based on uncertainties in the cask design and transportation plans.

The significance was ranked as low for ST-Loading, Storage-Accident, and Unloading scenarios because the in-reactor conditions experienced by the fuel would likely bound those found in these scenarios. The significance for the NCT scenario was ranked as medium because of potential cumulative effects (e.g., vibration due to transportation and water introduction), but the panel noted that in isolation, the significance was likely low. The significance was ranked high for the HAC scenario because of potential for additional fuel handling due to repackaging combined with the uncertainty in cask design.

Note that the initial rankings had multiple “L/M” in the knowledge category and “M/L” in the significance category. The low knowledge and medium significance were for fuel compacts, while the medium knowledge and low significance were for all other items considered within the scope of the analysis (i.e., pebbles). Subsequent discussions led to the assumption that waste volume reduction activities were not considered. Therefore, the rankings for fuel compacts are not included in Table 3-1. because it is assumed that fuel compacts would only be handled while inside a fuel block.

The panel recommended performing 30-foot drop testing of casks and/or gathering additional post-irradiation data to better understand performance and to potentially reduce the significance of the HAC scenario.

Table 3-1
Matrix fracture ranking

Scenario	Operable	Knowledge	Confidence	Significance
ST – Loading	Y	M	M	L
Storage – Normal	N			
Storage – LT	N			
Storage – ON	N			
Storage – Accident	Y	M	M	L
Unloading	Y	M	M	L
NCT	Y	L		M
HAC	Y	L		H

Non-fuel Block Fracture

Phenomenon Description

This phenomenon refers to the fracture of graphite blocks (such as prismatic blocks) which contain the fuel compacts. As a small amount of radioactive material may be present within blocks due to release from the fuel or from activation of carbon or impurities in the block itself, fracture of the block may cause fuel relocation and challenge the evaluation criterion. This phenomenon is not applicable to graphite pebble designs, because the current designs assume that the pebbles are made entirely of matrix material, therefore fracture of pebbles is grouped into Section 3.1.

Available Data and Quality

The mechanical properties of graphite and the changes due to irradiation have been studied in the Advanced Graphite Creep (AGC) experiment campaign at INL [References 11 and 12]. Reference [11] documents PIE data from graphite in the AGC-2 experiment (irradiated at 600°C to approximately 5 dpa) and Reference [12] documents data from the AGC-3 experiment (irradiated at 820°C to approximately 3.7 dpa). These reports include data for the density, elastic modulus, and thermal properties of the tested graphite. The AGC-2 experiment contained 16 graphite grades and the AGC-3 experiment contained 11 graphite grades. In both experiments, the elastic modulus of all graphite grades increased between 30% and 100% compared to the unirradiated samples. Destructive testing of the samples was planned to generate more mechanical property data, but these samples have been repurposed for the High Dose Graphite (HDG) experiment campaign.

With respect to operating experience, the existing records for both Fort. St. Vrain fuel and the German THTR reactors do not indicate degradation or damage to the fuel blocks [10].

While supporting data exist to inform calculation of failure probability during transportation and storage scenarios, it is difficult to consider the likelihood of block fracture without more detailed storage and transportation plans and cask designs.

Ranking

The summary of the non-fuel block fracture ranking is presented in Table 3-2. Operability for this phenomenon was determined by considering if there was the potential for a significant enough mechanical shock during each scenario, noting the relative strength of the blocks. Thus, ST-Loading, Unloading, Storage-Accident, and HAC were ranked as operable. The Storage-Accident scenario was marked operable due to the possibility of earthquake tip over, but the panel noted that this scenario could be mitigated by design.

The knowledge was ranked as medium for all operable scenarios due to a combination of available data and previous OE for similar, but not identical, systems. The confidence was ranked as medium for all operable scenarios, except for HAC which was ranked as low due to the higher stresses occurring in this scenario combined with the uncertainties in the cask design.

The significance was ranked as low for ST-Loading, Storage-Accident, and Unloading scenarios because fracture will likely be mitigated by design and the amount of radioactive material exposed by block fracture is expected to be low. The significance was ranked medium for the HAC scenario because of the potential for additional fuel handling due to repackaging combined with the uncertainty in cask design.

To increase knowledge and confidence and potentially decrease significance of the HAC scenario, the panel recommended performing 30 ft drop testing of the storage /transportation system and/or gathering additional post-irradiation data to better understand performance.

Table 3-2
Non-fuel block fracture ranking

Scenario	Operable	Knowledge	Confidence	Significance
ST – Loading	Y	M	M	L
Storage – Normal	N			
Storage – LT	N			
Storage – ON	N			
Storage – Accident	Y	M	M	L
Unloading	Y	M	M	L
NCT	N			
HAC	Y	M	L	M

Abrasive Wear

Phenomenon Description

The phenomenon refers to wear of TRISO particles and pebbles caused by friction due to interactions between pebbles and other hardware that might come into contact with pebbles, such as storage casks or transportation package components. Wear of the pebbles may expose TRISO particles, compromise integrity of the pebbles, and lead to graphite dust formation, which is more mobile and therefore could be more easily released.

Available Data and Quality

There is a significant amount of literature available on abrasive wear for graphite, with an emphasis on graphite dust production [13, 14, 15, 16]. These include comparisons of information gathered from modeling efforts to data gathered from experiments. However, analysis conducted on all reviewed literature focused on in-core conditions, which are generally not applicable to transportation and storage scenarios. Nevertheless, the extent of abrasive wear caused in the core is expected to be bounding compared to that in storage and transportation scenarios. In addition, abrasive wear is difficult to predict during the scenarios considered in this PIRT without a better understanding of storage and transportation plans, including actual cask and package designs.

Ranking

The summary of the abrasive wear ranking is presented in Table 3-3. Operability for this phenomenon was determined by considering the potential for high frequency vibrations during a scenario. Therefore, only ST-Loading, Unloading and NCT were considered operable.

For the scenarios analyzed, the knowledge was ranked as low, as there is no experimental data on wear in storage and transportation conditions. Additionally, there is significant uncertainty with cask and package design and how storage and transportation activities will be conducted to determine where vibrations may occur.

The significance was also ranked as low for the scenarios analyzed because transportation and storage conditions are expected to be bounded by in-core conditions.

Table 3-3
Abrasive wear ranking

Scenario	Operable	Knowledge	Confidence	Significance
ST – Loading	Y	L		L
Storage – Normal	N			
Storage – LT	N			
Storage – ON	N			
Storage – Accident	N			
Unloading	Y	L		L
NCT	Y	L		L
HAC	N			

TRISO Particle Layer Fracture

Phenomenon Description

This phenomenon refers to the fracture of the three layers (IPyC, SiC, and OPyC) in the TRISO particle simultaneously due to a short duration mechanical shock. Assuming all layers fail simultaneously is more conservative than considering the fracture of individual layers. Time-dependent degradation of the layers is included in subsequent phenomena. The matrix material is assumed to not retain any radionuclides that would be exposed in the fracture of the TRISO particle, which may cause radionuclide dispersal, challenging evaluation criterion.

Available Data and Quality

Data in the literature show that cracking of the TRISO particle layers is not likely under anticipated loading during storage and transport conditions. Reference [17] documents research on recycling of TRISO particles to inform the resistance of the TRISO particle layers to fracture. This research found that the published work associated with the reprocessing of TRISO particles is sparse and there appears to be no quantitative data on the susceptibility of TRISO particles in a fuel form to fracture.

Reference [18] documents preparation for a series of leach-burn-leach (LBL) tests performed at ORNL. The experiment seen in Figure 3-1 below was developed to crack individual TRISO particles suspended in epoxy. Multiple rounds of testing determined that radial through cracks could be formed by dropping a 4.9 g weight from a height of 2.5–4 cm. While not directly applicable to transport and storage scenarios, this reference provides quantitative information regarding TRISO layer cracking, which is generally lacking in literature.

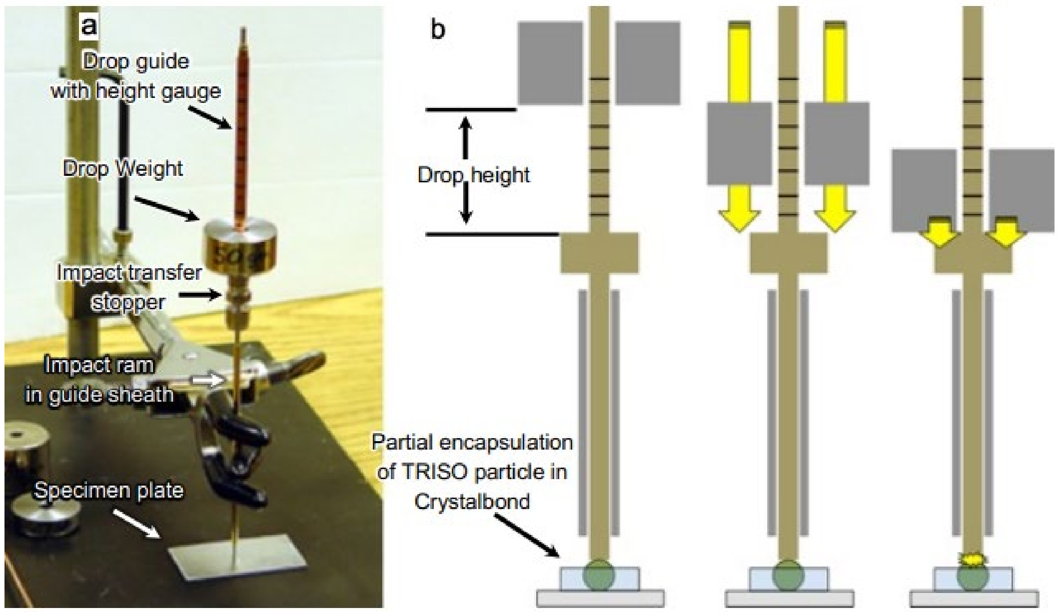

Figure 3-1
Single TRISO particle fracture test set up [18]

Reference [19] documents studies on the fission gas retention properties of TRISO particles in a 3D-printed SiC matrix. It is generally expected that cracks would be deflected by the OPyC layer and would not impact the integrity of the SiC layer because of the relative difference in strength between the two layers. This behavior was reflected in the majority of test results obtained; however, cracks were able to propagate through the particles in some rare instances. This cracking is likely due to strong mechanical interlocking at layer interfaces caused by localized increases in matrix density.

The studies above are not directly applicable to the conditions experienced in transportation and storage scenarios. However, these studies do provide insight that was used by the panel when considering TRISO performance during transportation and storage scenarios.

Rankings

The summary of the TRISO particle layer fracture ranking is presented in Table 3-4. Operability for this phenomenon was determined by considering the potential for a significant mechanical shock in each scenario. Only the ST-Loading, Unloading, and HAC scenarios were ranked as operable.

The knowledge was ranked as low for the operable scenarios due to the general lack of quantitative data and the large amount of uncertainty associated with knowledge of the forces experienced by the TRISO particles, caused by the lack of information on cask design as well as storage and transportation activities.

The significance was ranked as low because the panel determined it to be unlikely that a significant enough fraction of particles would crack to challenge the evaluation criterion.

Table 3-4
TRISO particle layer fracture ranking

Scenario	Operable	Knowledge	Confidence	Significance
ST – Loading	Y	L		L
Storage – Normal	N			
Storage – LT	N			
Storage – ON	N			
Storage – Accident	N			
Unloading	Y	L		L
NCT	N			
HAC	Y	L		L

PyC Creep

Phenomenon Description

This phenomenon refers to thermal creep of the outer pyrolytic carbon layer. The PyC layer imparts a compressive stress on the SiC. When exposed to elevated temperatures for sufficient time, creep may remove this compressive stress, putting the SiC layer into tension due to the development of fission gas. The SiC layer is mechanically much weaker in tension, leading to potential failure and loss of radionuclide retention by the particles.

Available Data and Quality

The available data on PyC creep is generally sparse, and non-existent specifically for conditions relevant to transportation and storage.

PARFUME (PARticle FUel ModEl) is a code that models TRISO fuel performance in advanced reactors. Reference [20] documents the theory and underlying models used to develop PARFUME. In the code, creep strain is not considered for the kernel and the SiC layer, but is considered for the buffer, IPyC, and OPyC layers. All creep is treated as secondary creep (creep strain rate is proportional to stress). Notably, the PARFUME material models are limited to a temperature range of 600°C to 1300°C. These temperatures far exceed the expected temperatures in storage and transportation scenarios.

The BISON code, a finite element nuclear fuel code developed by INL, also contains a PyC creep model. However, this model appears to be based on the PARFUME code and has the same temperature restrictions [21].

Both of these models are only applicable under the presence of irradiation. Graphitic material such as the PyC layer is not expected to creep through thermal mechanism alone.

Ranking

The summary of the PyC Creep ranking is presented in Table 3-5. The phenomenon was not considered to be operable in any scenario because no scenarios have irradiation present to cause creep to occur.

Table 3-5
PyC creep ranking

Scenario	Operable	Knowledge	Confidence	Significance
ST – Loading	N			
Storage – Normal	N			
Storage – LT	N			
Storage – ON	N			
Storage – Accident	N			
Unloading	N			
NCT	N			
HAC	N			

SiC Corrosion

Phenomenon Description

This phenomenon refers to the degradation of the SiC layer via exposure to corrosive chemical agents, including transition metals. This exposure reduces the SiC layer's capabilities to retain fission products. Corrosion could potentially come from inside the SiC layer as well as from outside. From inside, fission products might diffuse through the other layers and attack SiC. However, the TRISO particle's temperature during transportation and storage is expected to be low enough to prevent significant diffusion. Corrosion from outside the layer could be due to the presence of residual molten salts or other metals present in the system as first row transition metals (Cr, Mn, Fe, Co, Ni), or oxygen and water. Corrosion reduces the layer thickness, decreasing the SiC effective tensile strength and increasing the probability of fracture, potentially degrading the stronger barrier to fission products in the TRISO particle.

Available Data and Quality

Data is generally available for corrosion of SiC in different environments. Reference [22] provides a summary of the initial observations of the AGR 5/6/7 Capsule 1 experiments, with Section 4.1 describing the concerns of nickel attack of the SiC layer. Post-irradiation examination (PIE) of capsule 1 found nickel rich deposits. Nickel is suspected to have migrated from thermocouples (TCs). The AGR 5/6/7 included TCs with a Ni sheath surrounded by Nb protective sleeves. A few particle failures in the experiment were attributed to Ni attack.

Reference [23] documents a series of performance tests on TRISO fuel in repository rock conditions. This includes a study of SiC irradiated at the Petten facility at full power for a total of 270 days at a temperature of 270–300 °C. Key results of this experiment are shown below in Table 3-7. The composition of the aqueous solutions is detailed in

Table 3-6
Composition of aqueous solutions used in Petten study [23]

	Granite Water	Q-brine	Clay Pore*
NaCl	199.77 mg/l	350 g/l	44 mg/l
Na_2SO_4	402.73 mg/l	3.04 g/l	1.5 mg/l
$NaHCO_3$	392.38 mg		1250 mg/l
Na_2CO_3	6.22 mg/l		
$MgCl_2$	2.53 mg/l†		
$MgSO_4$		2.25 g/l†	12.0 mg/l
$CaCl_2$	12.3 kg/l	3.12 g/l†	
$CaSO_4$	42.36 mg/l		
$B(OH)_3$	3.4 mg/l		
KCl			20 mg/l
NaF			8 mg/l

*pH was adjusted with NaOH or nitric acid
† mass as the hydrate form

Table 3-7
SiC degradation rates in aqueous solutions (Adapted from Reference [23])

Aqueous Solution	Temperature (C)	R (g m-2/day)	Expected Lifetime (years)
Granite water	90	9.64E−06	2.70E+03
Q-brine	180	4.09E−05	6.39E+03
Clay pore pH 9	90	2.30E−05	1.14E+04
Clay pore pH 12	90	3.70E−05	7.07E+03
Clay pore pH 3	90	1.09E−05	2.39E+04

The conditions selected in this experiment are expected to be representative of potential storage conditions in a repository. Assuming uniform corrosion of a 30 µm thick SiC layer, the expected lifetime of the SiC layer under all conditions was calculated to significantly exceed the timescales of all scenarios considered in this exercise.

Ranking

Operability for this phenomenon was determined by considering which scenarios could include a significant enough amount of corrosive agent to challenge the integrity of the SiC layer, in addition to a combination of time and temperature to drive the reaction kinetics. The panel determined that no scenario contained these parameters, therefore the phenomenon was ranked as inoperable for all scenarios.

Particle, Matrix, and Block Oxidation

Phenomenon Description

This phenomenon refers to oxidation of the PyC in the particle, the graphite and pyrolyzed resin in the matrix, and/or the graphite block from exposure to air or moisture. Oxidation may adversely affect mechanical properties, which could reduce the integrity of the block or matrix, impacting the ability to maintain geometry. Oxidation of the graphite in one of the components considered may impact the ability to retain fission products. These outcomes would challenge the evaluation criterion.

Available Data and Quality

Relevant operating experience is available from the storage of the Fort St. Vrain fuel in a non-inert (an air) environment, where the maximum allowable storage temperature was limited to 400°C in air due to graphite oxidation [24]. The expected temperature for spent TRISO fuel in transportation and storage conditions is expected to be below this historical limit. While the temperature will be system design dependent, thermal modeling of a hypothetical canister-based system estimated a maximum temperature for TRISO pebbles to be approximately 150°C evaluated one year after discharge [25].

Oxidation data for graphite is generally available. Reference [26] documents a study to test the oxidation performance of graphite matrix material in water vapor based on the AGR 5/6/7 tests. Studies were conducted over a wide range of conditions, including kinetics and diffusion regimes for graphite. Data collected was then fit to two established models, the Langmuir-Hinshelwood (LH) and Boltzmann-enhanced Langmuir-Hinshelwood (BLH) models. Both models fit reasonably well, with the LH model performing more favorably at lower temperatures and oxidant partial pressures, which confirms results found in other studies. Data is available at temperatures between 850°C and 1500°C, which is higher than temperatures expected in storage.

The oxidation of matrix graphite in air is studied in Reference [27] between 500°C and 900°C. This work developed an empirical correlation for the oxidation rate as a function of temperature. The impacts of oxidation at various temperatures on the compressive strength of the graphite was also studied. At 10% weight loss, a 77% reduction in compressive strength was observed. The degree of reduction depended on the oxidation temperature, lower temperatures being more severe. At the lower oxidation temperature, oxygen atoms are more able to penetrate the bulk of the graphite while at higher the oxidation temperature oxidation is a surface reaction.

Overall, data is available to quantify the oxidation of graphite over a wide temperature range. However, the temperatures experienced during storage and transportation may be below the range where oxidation becomes significant. The lower temperatures will likely limit the oxidation rate. Additionally, previous experience with the Fort St. Vrain fuel, and predictions for the thermal loading of spent fuel casks, provide a robust knowledge base for this phenomenon.

Ranking

The summary of the Particle, Matrix, and Block Oxidation ranking is presented in Table 3-8. The phenomenon was determined to be operable for all scenarios, with the assumption that air and moisture may always be present. More detailed designs, such as those that include inert environments, may change this ranking.

The knowledge was ranked as high, largely due to existing operational experience, experimental data, and models. The significance of this phenomenon was ranked as low for all scenarios, with the exception of Storage-Accident and HAC, because the temperature will be insufficient to drive oxidation. The Storage-Accident and HAC scenarios were ranked as medium because certain accident scenarios may have higher temperatures due to fire, leading to increased oxidation.

Table 3-8
Particle, matrix, and block oxidation ranking

Scenario	Operable	Knowledge	Confidence	Significance
ST – Loading	Y	H	M	L
Storage – Normal	Y	H	M	L
Storage – LT	Y	H	M	L
Storage – ON	Y	H	M	L
Storage – Accident	Y	H	M	M
Unloading	Y	H	M	L
NCT	Y	H	M	L
HAC	Y	H	M	M

Helium Over-pressurization from Alpha Decay

Phenomenon Description

This phenomenon refers to the buildup of helium pressure in the buffer layer of a TRISO particle due to alpha decay. During operation, fission gases are produced in the kernel and are retained by the three layers, which act as pressure vessel. During transportation and storage, the main pressure increase will be due to helium generated by radioactive decay of alpha-emitting nuclides. Over-pressurization may cause the tensile stresses in the SiC layer to exceed its structural capacity. As SiC is considered the primary barrier, when it ruptures the TRISO particle is considered failed, potentially allowing for unacceptable radionuclide release.

Available Data and Quality

Reference [28] documents a deterministic performance assessment of high burnup HTGR fuel in the proposed Yucca Mountain repository. This assessment includes helium pressurization due to alpha decay using application of the ideal gas law. Reference [28] found that the internal pressure of a TRISO particle does not significantly increase over the first 100 years, which is the maximum timeframe considered within this PIRT. According to the calculations performed in the literature, internal pressure rises by approximately 5MPa over the given timeframe, assuming ideal gas behavior and room temperature. The pressure increase is not expected to challenge the integrity of the SiC layer for at least 7,000 years, with the ultimate timescale depending on the assumptions made for environmental conditions [28]. These timeframes exceed those considered within this PIRT.

Ranking

The summary of the helium over-pressurization (alpha decay) ranking is presented in Table 3-9. Operability was determined by considering the length of a given scenario. No scenario considered allows for a significant buildup of pressure within the particles. All scenarios are ranked as inoperable.

Table 3-9
Helium over-pressurization (alpha decay) ranking

Scenario	Operable	Knowledge	Confidence	Significance
ST – Loading	N			
Storage – Normal	N			
Storage – LT	N			
Storage – ON	N			
Storage – Accident	N			
Unloading	N			
NCT	N			
HAC	N			

Fission Product Leaching

Phenomenon Description

This phenomenon refers to the chemical leaching and mobilization of fission products outside of the intact SiC layer due to other species that act as external drivers for the fission products to leach outside of the TRISO particle.

Available Data and Quality

Reference [18] describes the preparation for a LBL experiment at ORNL, while Reference [29] describes the results of those experiments. These tests involved approximately 10,000 surrogate TRISO particles with simulated impurities, and up to four uranium-bearing TRISO particles with simulated defects. The methodology for LBL testing is shown in Figure 3-2. The leaching steps were performed using 70% concentrated nitric acid at near boiling temperature (~120°C), conditions that are far more severe than any potential transportation and storage scenario.

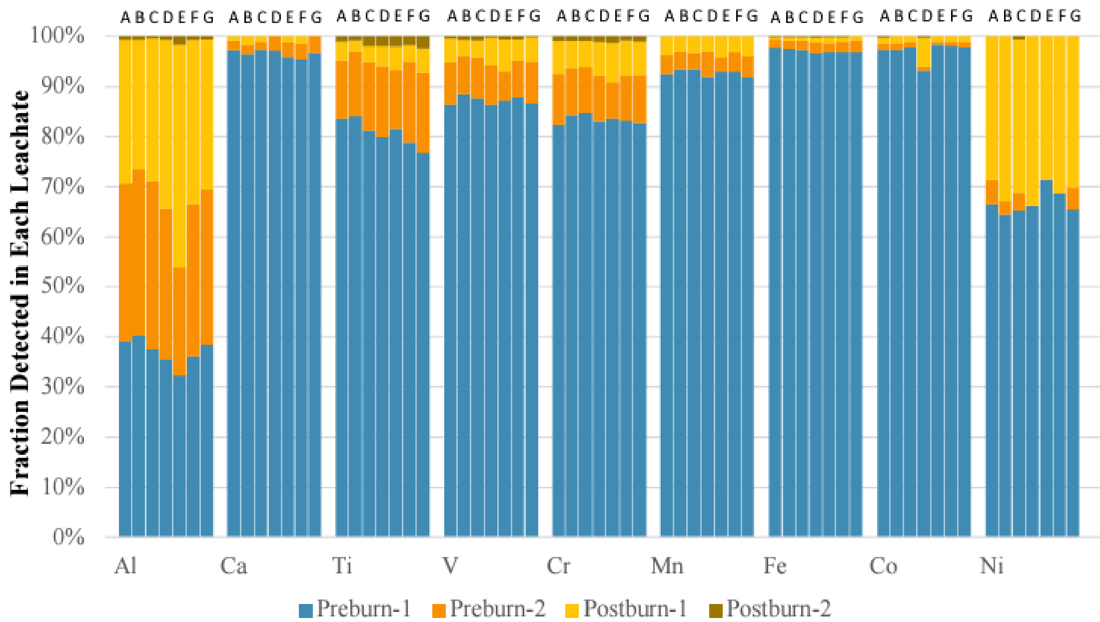

Figure 3-2

Figure 3-3 and Figure 3-4 show the fraction of various elements detected in the leaching agent at different steps in the process. Elements such as calcium, iron, cobalt, strontium, and thorium were almost completely leached before the first burn phase. This experiment demonstrates the extreme conditions required to achieve significant fission product leaching, which far exceed any possible condition experienced during transportation and storage scenarios.

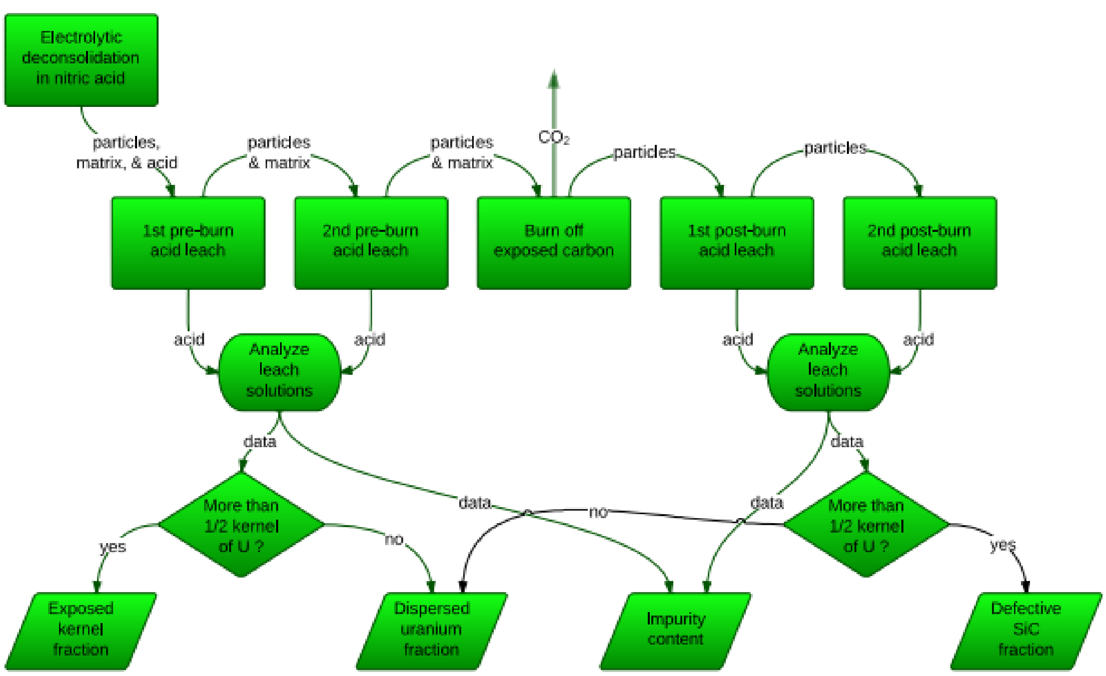

Figure 3-3
Flowchart of the LBL process demonstrating the Individual phases in the analysis and determination of particle defect properties [29].

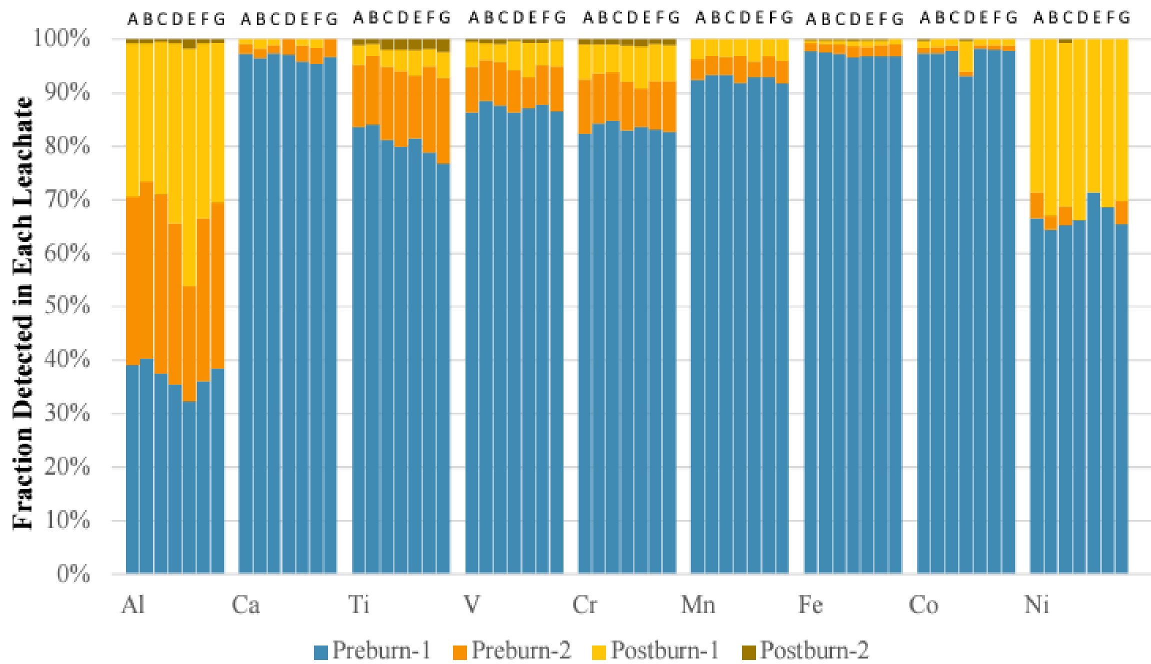

Figure 3-4
Fraction per leach for QC-relevant impurities measured in simulated samples [29]

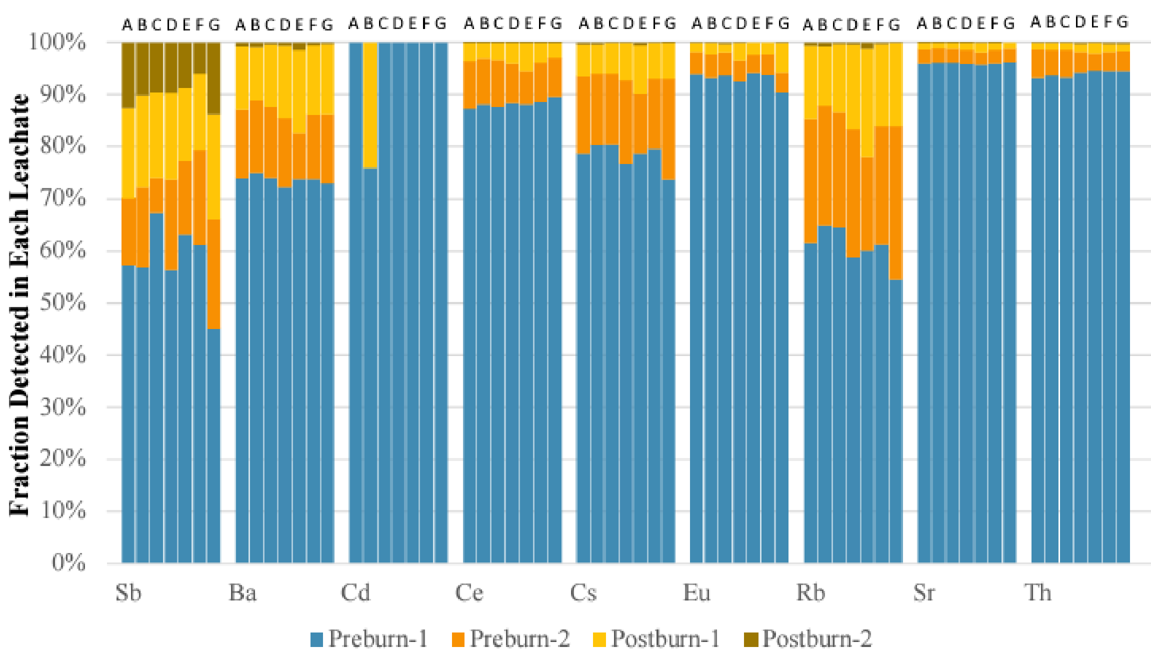

Figure 3-5
Fraction per leach for other targeted impurities measured simulated samples [29]

Ranking

The summary of the fission product leaching ranking is presented in Table 3-10. This phenomenon was predicted to be operable only in accident scenarios (Storage-Accident and HAC,) as leachants should not be present in any other scenario. The panel noted that a cask design should prevent this phenomenon from occurring, however, more information is needed to determine if this is the case.

The knowledge was ranked as medium. The understanding of fission product solubility is high, but the understanding of storage configurations is low. Combination of these two aspects led to a medium ranking overall.

The significance was ranked as low because the amount of fission products outside of an intact SiC should not allow for significant release of radionuclides.

Table 3-10
Fission product leaching ranking

Scenario	Operable	Knowledge	Confidence	Significance
ST – Loading	N			
Storage – Normal	N			
Storage – LT	N			
Storage – ON	N			
Storage – Accident	Y	M	H	L
Unloading	N			
NCT	N			
HAC	Y	M	H	L

Fission Product Diffusion

Phenomenon Description

This phenomenon refers to the movement of fission products under temperature or concentration gradients from inside the kernel through the TRISO particles. During transportation and storage scenarios, there is a temperature gradient through the TRISO particles and surrounding materials. There is also a concentration gradient of fission products. Both gradients can lead to diffusion of isotopes, such as Ag-110, through the particles. The time at temperature increases the extent of the diffusion. This diffusion may lead to an unacceptable release of radionuclides.

Available Data and Quality

There is a significant amount of literature available on the thermal diffusion of fission products for TRISO fuel. Diffusion coefficients of the main fission products are available. These data were used in analytical models built into the existing BISON code, and the models were validated against experimental results from the AGR tests [30]. However, these results are focused on in-core conditions and conditions experienced by the fuel during post-irradiation examination testing. There is a gap in the available data for fission product diffusion in the temperature ranges relevant to transportation and storage scenarios, although diffusion data for in-reactor conditions bound storage due to the higher temperatures. However, it is unclear how relevant the models described in Reference [30] are to the conditions of interest in this PIRT.

Ranking

The summary of the fission product diffusion ranking is presented in Table 3-11. Scenarios with either the potential for high temperatures or that encompassed long periods of time were considered to be operable, which include Storage-Normal, Storage-LT, Storage-ON, Storage-Accident, and HAC.

The knowledge was ranked as medium for all scenarios, as well-validated models exist for fission product diffusion at higher temperatures. However, it is unclear how these models will extrapolate to lower temperatures.

The significance was ranked as low because none of the scenarios had sufficient time at an elevated temperature to achieve significant diffusion. In addition, this phenomenon will likely be bounded by the conditions TRISO fuel experiences while in the reactor based on the exponential dependence of diffusion on temperature versus a linear dependence on time.

Table 3-11
Fission product diffusion ranking

Scenario	Operable	Knowledge	Confidence	Significance
ST – Loading	N			
Storage – Normal	Y	M	M	L
Storage – LT	Y	M	M	L
Storage – ON	Y	M	M	L
Storage – Accident	Y	M	M	L
Unloading	N			
NCT	N			
HAC	Y	M	M	L

Neutron Multiplication

Phenomenon Description

This phenomenon refers to the extent to which fissions in fissile material cause unacceptable heat generation and dose. This could be due to reorientation and reconfiguration of the spent fuel causing local criticality and heat generation. An example precursor event could be a non-fuel block fracture.

Available Data and Quality

References [31] and [32] document criticality benchmark analysis of multiple HTGRs, including the PBMR-400, PBMM, GT-MHR, HTR-10, and VHTR Critical Assembly. Multiple analysis codes were considered within these analyses, including SCALE, Serpent, and MCNP. Many parameters were found to impact the accuracy of the benchmark models compared to the experimental data, including modeling decisions and cross-section libraries. However, in these benchmarks, reasonable agreement was achieved between the experimental and computational data. One example from Reference [32] is shown in Figure 3-5.

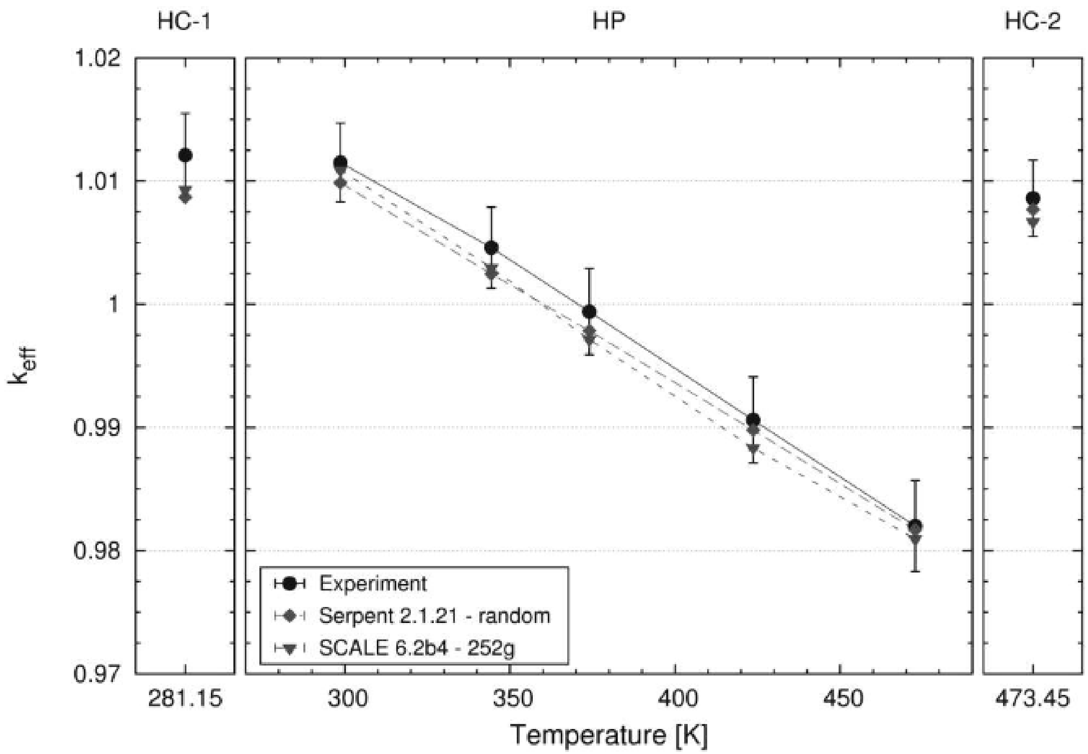

Figure 3-6
Comparison of the SCALE/KENO-VI MH and Serpent VHTRC multiplication factors with ENDF/B-VII.0 cross-sections [32].

Despite the existence of the benchmarks discussed above, the overall suite of benchmarks that include key features of advanced designs, such as graphite and HALEU fuel, are generally sparse. The NRC and the DOE have an ongoing program that intends to increase the number of relevant benchmarks [33]. These additional benchmarks will give further confidence in the modeling approach for TRISO fueled reactors.

Ranking

The summary of the neutron multiplication ranking is presented in Table 3-12. This phenomenon was deemed operable during all scenarios.

For all scenarios, the knowledge was ranked as high for modeling tools and capabilities, but low for model validation. This low ranking comes from gaps in the number of critical experiments for TRISO fuel and potential gaps in the cross-section for key materials, particularly graphite.

The significance was ranked as medium for all scenarios, except for HAC. The medium ranking is driven primarily by uncertainties in validation and cask design. Significance of the HAC scenario was ranked high because it is most likely to challenge maintenance of the criticality limit.

To account for the lack of validation data, the panel recommends that designers perform criticality studies or address uncertainty through design margins.

Table 3-12
Neutron multiplication ranking

Scenario	Operable	Knowledge[1]	Confidence	Significance
ST – Loading	Y	L/H	L	M
Storage – Normal	Y	L/H	L	M
Storage – LT	Y	L/H	L	M
Storage – ON	Y	L/H	L	M
Storage – Accident	Y	L/H	L	M
Unloading	Y	L/H	L	M
NCT	Y	L/H	L	M
HAC	Y	L/H	L	H

Notes: 1. Low for validation, high for models.

Decay Heat

Phenomenon Description

This phenomenon refers to heat produced by the radioactive decay of isotopes present in the spent fuel. This phenomenon only considers the decay heat in isolation. The decay heat can lead to degradation through other phenomena, such as oxidation or diffusion; however, these effects are considered in other phenomenon evaluations.

Available Data and Quality

Decay heat and source term predictions for TRISO SNF are calculated from an estimated isotopic inventory. This inventory depends on many parameters, including the neutron spectrum and temperature. Multiple analytical tools have been developed to calculate this inventory, including the SCALE package from ORNL and the Griffin package within INL's Multiphysics Object-Oriented Simulation Environment (MOOSE). These codes have been applied to many advanced reactor designs, including pebble bed reactors [34, 35, and 36] and prismatic high temperature gas reactors [36 and 37]. Additionally, work has been done to quantify the uncertainty within the calculated inventories [38].

The temperature experienced by the fuel from the decay heat will vary based on design parameters, including fuel type and burnup. As an example, Reference [25] calculated that the maximum temperature of spent TRISO pebbles was approximately 150°C in a hypothetical storage container one year after discharge. This temperature is significantly below that seen in traditional LWR fuel. The lower temperatures are caused by relatively lower power density of TRISO fuel designs.

Ranking

The summary of the decay heat ranking is presented in Table 3-13. This phenomenon was operable for all scenarios, as the spent fuel will inherently produce heat.

For all scenarios, the knowledge and confidence were ranked as low for pebbles and high for prismatic designs. This discrepancy is caused by the uncertainties in the power profile used for calculating isotopic inventory for pebbles, which are not present in other designs.

The significance was ranked as low because the lower heat loading presumed for transportation and storage scenarios is not expected to challenge evaluation criterion. This assumption should be revisited if future designs have heat loadings that are higher than anticipated.

Table 3-13
Decay heat ranking

Scenario	Operable	Knowledge[1]	Confidence[1]	Significance
ST – Loading	Y	L/H	L/H	L
Storage – Normal	Y	L/H	L/H	L
Storage – LT	Y	L/H	L/H	L
Storage – ON	Y	L/H	L/H	L
Storage – Accident	Y	L/H	L/H	L
Unloading	Y	L/H	L/H	L
NCT	Y	L/H	L/H	L
HAC	Y	L/H	L/H	L

Notes: 1. Low for pebbles, high for prismatic fuels.

4

CONCLUSIONS, OBSERVATIONS, AND RECOMMENDATIONS

Summary

A panel of recognized TRISO fuel experts was assembled to evaluate the current state of knowledge of spent TRISO fuel during transportation and storage conditions. Among the phenomena ranked, only two were evaluated as high significance, meaning that they could lead to unacceptable radiological release of radionuclides: matrix fracture and neutron multiplication. Both high rankings occurred for the HAC scenario and the rankings were affected by the lack of transportation and storage system designs.

Conclusions

The PIRT panel came to the following key conclusions:

- The TRISO barriers to radiological release and dose consequence can be credited, but the extent to which they can be credited needs to take the design of the storage and transportation packages into consideration.
- The existing practices used for storage and transportation of commercial LWR SNF (i.e., leak-tight cask, providing containment/confinement in all scenarios) are compatible with TRISO fuels. Additional analytical and/or experimental work is likely required to evaluate TRISO under transportation accident conditions.
- TRISO properties (e.g., mechanical properties and thermal management) may enable novel storage and transportation designs, but additional data is required.
  - Confinement requirements may be achieved in a different manner for storage.
  - The lower energy density allows design requirements for thermal management during transportation to be reconsidered.
- Guidance for spent fuel could be updated to reflect unique attributes of TRISO fuel.

Observations

The PIRT panel made the following observations:

- A designer cannot evaluate spent TRISO particles in isolation in transportation and storage. The entire system (see section 2.2) must be considered.
- The lack of TRISO-specific designs for storage and transportation systems makes it challenging to fully evaluate against storage and transportation requirements.

- Volume reduction activities were not considered and would affect the conclusions of this PIRT.
- Most analyses cited in this PIRT were applicable to the AGR program and the fuel that was examined as part of it. The panel believes that uncertainty regarding SNF behavior increases as fuels diverge from the parameters tested in the AGR program.

Recommendations

The PIRT panel made the following recommendations:

- Continue the ongoing program to establish criticality benchmarks and nuclear data [33], and to ensure appropriate industry input.
- Assess the desirability of alternative storage technologies that leverage TRISO characteristics to meet regulatory requirements as cost-efficiently as possible.
- Collect additional data to evaluate the effects of TRISO layer fracture and matrix fracture on source term and criticality.
- Determine a definition for fuel failure in TRISO fuel systems.

5 REFERENCES

1. Marciulescu C., and A. Sowder, *Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance*, Topical Report EPRI-AR-1(NP), EPRI, Palo Alto, California, 2019.
2. "*Uranium-Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance: Topical Report EPRI-AR-1(NP), Docket NO. 99902021*," NRC Final Safety Evaluation, Document ML20216A453, U.S. Nuclear Regulatory Commission, 2021.
3. Wells, BE., et al., *TRISO Fuel: Properties and Failure Modes*, Pacific Northwest National Laboratory, Report PNNL-31427, Washington, 2021.
4. Geelhood, K., Pacific Northwest National Laboratory, "TRISO Compact Matrix Strength," Letter, Addressed to Adkins, Harold, Advisor at Pacific Northwest National Laboratory, October 13, 2023.
5. US Nuclear Regulatory Commission, *TRISO-Coated Particle Fuel Phenomenon Identification and Ranking Tables (PIRTs) for Fission Product Transport Due to Manufacturing, Operations, and Accidents,* NUREG/CR-6844, Vol. 1. Washington, D.C. 2004.
6. Bajwa, C. S., et al., "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*, Chicago, Illinois, July 27–31, 2008, Vol. 7: Operations, Applications and Components.
7. Borowski, J., et al., *Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material*, NUREG-2216, U.S. Nuclear Regulatory Commission, Washington, D.C., 2020.
8. *Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities*, NUREG-2215, U.S. Nuclear Regulatory Commission, Washington, D.C., 2020.
9. Xing, J. and Morrow, S., *White Paper: Practical Insights and Lessons Learned on Implementing Expert Elicitation*, Report, U.S. Nuclear Regulatory Commission, Washington, D.C., October 13, 2016.
10. Hall, N. et al., *Transportation Experience and Potential Challenges with Transportation of Spent (Irradiated) Advanced Reactor Fuel Types,* Center for Nuclear Waste Regulatory Analyses, San Antonio, TX, January 2020.
11. Windes, W., et al., *AGC-2 Specimen Post-Irradiation Data Package Report*, INL/EXT-15-36244, Rev. 0, Idaho National Laboratory, Idaho Falls, ID, August 2015.
12. Windes, W., et al., *AGC-3 Specimen Post-Irradiation Examination Data Package Report*, INL/EXT-17-43823, Rev. 0, Idaho National Laboratory, Idaho Falls, ID, December 2017.

13. Rostamian, M., et al., “Computational and Experimental Prediction of Dust Production in Pebble Bed Reactors – Part I,” *Nuclear Engineering and Design*, Vol. 263, October 2013, pp. 500-508.
14. Hiruta, M., et al., “Computational and Experimental Prediction of Dust Production in Pebble Bed Reactors – Part II,” *Nuclear Engineering and Design*, Vol. 263, October 2013, pp. 509-514.
15. Stempniewicz, M. M., et al., “Analysis of Dust and Fission Products in a Pebble Bed NGNP,” *Nuclear Engineering and Design*, Vol. 251, October 2012, pp. 433-442.
16. Cogliati, J. and Ougouag, A., “Pebble Bed Reactor Dust Production Model,” INL/CON-08-14226, *ASME 4th International Topical Meeting on High Temperature Reactor Technology Preprint*, Idaho National Laboratory, Idaho Falls, Idaho, September 2008.
17. Nesbit, S. P., “Summary of Information on Recycling TRISO Fuel,” *Meeting notes*, November 25, 2023.
18. Gerczak T., et al., *Preparation of Simulated LBL Defects for Round Robin Experiment*, ORNL/TM-2015/722-R3, Rev. 3, Oak Ridge National Laboratory, Oak Ridge, TN, September 2022.
19. Petrie, C. et al., “Fission gas retention of densely packed uranium carbonitride tristructural-isotropic fuel particles in a 3D printed SiC matrix,” Journal of Nuclear Materials, Vol. 580, July 2023, Article 154419.
20. Miller, G. K. et al., *PARFUME Theory and Model Basis Report*, INL/EXT-08-14497, Idaho National Laboratory, Idaho Falls, ID, September 2009.
21. Idaho National Laboratory, *BISON*, computer program, <https://mooseframework.inl.gov/bison/source/materials/solid_mechanics/PyCCEGACreep.html
22. Stempien, J. et al., *Initial Observations from Advanced Gas Reactor (AGR)-5/6/7 Capsule 1*, INL/RPT-22-66720, Rev. 0, Idaho National Laboratory, Idaho Falls, ID, April 2022.
23. Fachinger, J., et al., “Behavior of Spent HTR Fuel Elements in Aquatic Phases of Repository Host Rock Formations”, *Nuclear Engineering and Design*, Volume 236, Issues 5-6, March 2006, pp. 543-554.
24. U.S. Nuclear Regulatory Commission, Analysis Report for the Fort St. Vrain IFSI, Section 3.3.2.2.3, “Spent Fuel Elements” (NRC ADAMS Accession No. 102380351).
25. Petersen, G. et al., “Preliminary Analysis of Advanced Reactor Spent Nuclear Fuel Storage, Transportation, and Disposal”, INL/CON-24-76902-Rev000, *WM2024 Conference*, Phoenix, Arizona, March 10–14, 2024.
26. Gerczak, T. et al., *Oxidation of Matrix Material in Helium with Varied Moisture Content*, ORNL/TM-2019/1341, Oak Ridge National Laboratory, Oak Ridge, TN, November 2019.
27. Zhou, X. et al. “Oxidation Behavior of Matrix Graphite and Its Effect on Compressive Strength”, Science and Technology of Nuclear Installations, Volume 2017.

28. van den Akker B. P., and Ahn J., *Performance Assessment for Geological Disposal of Graphite Waste Containing TRISO Particles*, Nuclear Technology, 181:3, 408-426, DOI: 10.13182/NT11-103, February 21, 2012.
29. Hunn, J. et al., *ORNL Analysis of Leach-Burn-Leach Round-Robin Test Samples,* ORNL/TM-2023/2951, Oak Ridge National Laboratory, Oak Ridge, TN, June 2023.
30. Jiang, W., et al., *Fission Product Transport in TRISO Particles and Pebbles*, INL/EXT-21-63549, Idaho National Laboratory, Computational Mechanics and Materials Department, Idaho Falls, Idaho, 2021.
31. *Evaluation of High Temperature Gas Cooled Reactor Performance: Benchmark Analysis Related to the PBMR-400, PBMM, GT-MHR, HTR-10, and the ASTRA Critical Facility*, IAEA-TECDOC-1694, IAEA, Vienna, Austria, April 2013.
32. Bostelmann, F., et al., “Criticality calculations of the Very High Temperature Reactor Critical Assembly benchmark with Serpent and SCALE/KENO-VI,” Annals of Nuclear Energy, Vol. 90, April 2016, pp. 343-352.
33. U.S. Department of Energy, Office of Nuclear Energy, “Criticality Benchmarking,” https://www.energy.gov/ne/criticality-benchmarking, accessed on July 9, 2024.
34. Wing, J., et al., “Examining the Criticality and Dose Rate Aspects of Spent PBR Fuels in Storage and Transportation”, University of Tennessee, Waste Management Symposium, 2023.
35. Hanophy, O., and Stewart, B., Idaho National Laboratory, “Verification of equilibrium core solution and decay heat modeling for the GPBR200”, INL/RPT-23-03822, Idaho Falls, ID, 2023.
36. Ivanusa, P. et al., “Comparison of Irradiated TRISO Fuel Radioactivity from Multiple Advanced Reactor Designs”, Nuclear Technology, Vol. 208, 2022.
37. Stewart, B. and Schunert, R., Idaho National Laboratory, “Improved Decay Heat Prediction During HTGR Transients Scenarios”, INL/RPT-22-68845, Idaho Falls, ID, 2022.
38. D. Hartanto, et al., "Uncertainty Quantification of Pebble's Discharge Burnup and Isotopic Inventory Using SCALE," *Proceedings of the ANS Winter Meeting*, Phoenix, Arizona, November 13–17, 2022.

A
PANELIST RESUMES

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HAROLD E. ADKINS, JR., Pacific Northwest National Laboratory

CLASSIFICATION
Chief Research Engineer, Nuclear and Defense Systems

FIELD OF EXPERTISE
Nuclear power generation, storage, transport, disposition of used nuclear fuel, and weapons lifetime extension programs. Experimental process evaluation platform development and commissioning. Thermal hydraulic predictive modeling of combined mass, convective, and radiative transfer. Fluid Dynamics, Structural Mechanics, Heat Transfer, System Engineering and Development, Numerical Analysis, and Fast and Advanced Nuclear Energy Systems

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

EXPERIENCE
Pacific Northwest National Laboratory, Advisor (2000 – present) – Technical leader, principal investigator, and program manager on multiple NRC, DOE, and NNSA programs relating to nuclear power generation, storage, transport, disposition of used nuclear fuel, and weapons lifetime extension programs. Also specialize in experimental process evaluation platform development and commissioning.

Q-Metrics Consulting Services, Inc., Principal Engineer (1997 – 2000) – Thermal hydraulic predictive modeling of combined mass, convective, and radiative transfer for a diverse and broad-spectrum suite of commercial/private sector clients.

Westinghouse Hanford/Waste Management Federal Services, Inc., Senior Mechanical Engineer (1993 – 1997) – Lead analyst providing engineering and analytical support in the design and verification of Type B packaging systems.

SELECTED PUBLICATIONS
G.A. Coles P.P. Lowry T.A. Ikenberry C.A. Condon S.M. Short S.J. Maheras M.S. Taylor J.R. Phillips H.E. Adkins, Jr. 2024. Development and Demonstration of a Risk Assessment Approach for Approval of a Transportation Package of a Transportable Nuclear Power Plant for Domestic Highway Shipment, PNNL-36380, Rev 1. Richland, WA: Pacific Northwest National Laboratory.

Adkins H.E., S.J. Maheras, S.M. Short, and G.A. Coles. 12/14/2021. "Microreactor Transportation Probabilistic Risk Assessment Framework." Abstract submitted to International Conference on the Safe and Secure Transport of Nuclear Radioactive Materials, Vienna, Austria. PNNL-SA-161964.

Coles G.A., S.M. Short, S.J. Maheras, and H.E. Adkins. 2021. Proposed Risk-Informed Regulatory Framework for Approval of Microreactor Transportation Packages. PNNL-31867. Richland, WA: Pacific Northwest National Laboratory.

Eidelpes E., B.M. Hom, R. Hall, H.E. Adkins, and J. Jarrell. 2021. "A High-Assay Low-Enriched Uranium Fuel Transportation Concept." Nuclear Science and Engineering 195, no. 3:279-299. PNNL-SA-156769.

Adkins H.E., C.A. Borrelli, J.J. Mcguire, D.J. Richmond, S.M. Short, R. Hanson, and z. martin. 2021. Tritium Sustainment Program (TSP) Strategic Efficiency Enhancement Study (SEES) Revision 1. PNNL-29256. Richland, WA: Pacific Northwest National Laboratory.

Maheras S.J., and H.E. Adkins. 03/08/2021. "Microreactor Transportability Challenges." Abstract submitted to Waste Management 2021, Phoenix, Arizona. PNNL-SA-155618.

Adkins H.E., C.A. Borrelli, J.J. Mcguire, D.J. Richmond, S.M. Short, R. Hanson, and z. martin. 2019. Tritium Sustainment Program (TSP) Strategic Efficiency Enhancement Study (SEES). PNNL-29256. Richland, WA: Pacific Northwest National Laboratory.

Adkins H.E., and S.J. Maheras. 2020. Concept of Operations for Microreactor Transportation. PNNL-30166 Revision 1. Richland, WA: Pacific Northwest National Laboratory.

Barrett N.P., and H.E. Adkins. 07/20/2020. "Secondary Impacts and Their Effects On Closure Lid Bolt Stresses In A Generic Spent Nuclear Fuel Shipment Cask." Abstract submitted to ASME Pressure Vessels and Piping Conference, Minneapolis, Minnesota. PNNL-SA-148911.

Hom B.M., and H.E. Adkins. 2020. HALEU Structural, Thermal and Operational Evaluation. PNNL-29676. Richland, WA: Pacific Northwest National Laboratory.

Barrett N.P., and H.E. Adkins. 2020. "SECONDARY IMPACTS AND THEIR EFFECTS ON CLOSURE LID BOLT STRESSES IN A GENERIC SPENT NUCLEAR FUEL SHIPMENT CASK." In ASME 2020 Pressure Vessels & Piping Conference. PNNL-SA-150608.

Maheras S.J., and H.E. Adkins. 2020. Complete Documentation of Applicable Regulations and Regulatory Authority for Transportation. PNNL-29718. Richland, WA: Pacific Northwest National Laboratory.

Maheras S.J., and H.E. Adkins. 2020. Concept of Operations for Microreactor Transportation. PNNL-30166. Richland, WA: Pacific Northwest National Laboratory.

Maheras S.J., and H.E. Adkins. 03/08/2021. "Microreactor Transportability Challenges." Abstract submitted to Waste Management 2021, Phoenix, Arizona. PNNL-SA-155618.

Adkins HE, PC Ferrell, BJ Koeppel, JA Fort, AV Savino, and LM Hay. 2018. Packaging and Transportation of Radioactive Material to Address Hanford Onsite and Offsite Needs. PNNL SA 134302. Richland, WA: Pacific Northwest National Laboratory.

Adkins HE. 2018. Multi-Phase Transport Evaluation Loop Capabilities. PNNL-SA-134317. Richland, WA: Pacific Northwest National Laboratory.

Adkins HE, SA Bryan, KM Denslow, TG Levitskaia, and JM Schwantes. 2017. "Forecasting and Preventing WTP Process Upsets While Reducing Sampling with Real-Time Online Monitoring & Prognostics." WM2017 Conference, Phoenix, Arizona. PNNL-SA-120718.

Enderlin CW, HE Adkins, J Jenks, and JK Engeman. 2017. "Comparison of Exhibited Wear in Tefzel®, Kynar®, and UHMWPE Valve Seats Exposed to Cyclic Operation with Abrasive Slurry." Presented by Carl Enderlin at Waste Management Symposia, Phoenix, Arizona on March 8, 2017. PNNL-SA-124313.

Michener TE, DR Rector, JM Cuta, and HE Adkins. 2017. COBRA-SFS: A Thermal-Hydraulic Analysis Code for Spent Fuel Storage and Transportation Casks Cycle 4a. PNNL-24841 Rev. 1. Richland, WA: Pacific Northwest National Laboratory.

Enderlin CW, HE Adkins, EJ Berglin, RC Daniel, MS Fountain, PA Gauglitz, and LF Pease, et al. 2017. Development, Testing, and Verification in Support of Waste Management, Mobilization, Mixing, Retrieval, Transfer, and Processing. PNNL-SA-126823. Richland, WA: Pacific Northwest National Laboratory.

Adkins HE, JR Hilliard, and DM Wellman. 2016. "Conceptual Evaluation of Storage and Disposal Options for the Hanford Waste Encapsulation and Storage Facility Cesium Chloride and Strontium Fluoride Capsules." Waste Management, Phoenix, AZ. PNNL-SA-112323.

Adkins HE, and JM Cuta. 2015. "Technical Considerations for Thermal Analysis." Presented by Harold Adkins, Jr at 2015 Joint DOE-NNSA Packaging and Transportation Technical Exchange Meeting, Las Vegas, Nevada on May 12, 2015. PNNL-SA-110197.

Klymyshyn NA, HE Adkins, J Piotter, and C Bajwa. 2015. "Closure Bolt Modeling for Seal Evaluation Under Extreme Thermal Loads." In Proceedings of the ASME 2015 Pressure Vessels & Piping Conference. PNNL-SA-109678.

Fort JA, JM Cuta, and HE Adkins. 2015. A Compendium of Spent Fuel Transportation Package Response Analyses to Severe Fire Accident Scenarios. PNNL-24792. Richland, WA: Pacific Northwest National Laboratory.

Adkins HE, Jr, JM Cuta, NA Klymyshyn, SR Suffield, B Chris, KB McGrattan, CE Beyer, and A Sotomayor-Rivera. 2015. Spent Fuel Transportation Package Response to the MacArthur Maze Fire Scenario. PNNL-23701, Pacific Northwest National Laboratory, Richland, WA.

Adkins HE, Jr, JM Cuta, NA Klymyshyn, SR Suffield, KB McGrattan, and B Chris. 2015. Spent Fuel Transportation Package Response to the Newhall Pass Tunnel Fire Scenario. PNNL-23708, Pacific Northwest National Laboratory, Richland, WA.

Cuta JM, and HE Adkins, Jr. 2014. Thermal Modeling of HI-STORM 100 Storage Modules at Diablo Canyon Power Plant ISFSI. PNNL-23298, Pacific Northwest National Laboratory, Richland, WA.

Adkins HE, Jr, RL Howard, and S Marschman. 2013. Used Nuclear Fuel Loading and Structural Performance Under Normal Conditions of Transport - Method and Approach. PNNL-22221, Pacific Northwest National Laboratory, Richland, WA.

HE Adkins, Enderlin CW, JJ Jenks, CA Burns, and RC Daniel. 2013. Hanford Tank Farms Double Valve Isolation (DVI) Cycle Test Report. PNNL-22803, Pacific Northwest National Laboratory, Richland, WA.

Cuta JM, and HE Adkins, Jr. 2013. Thermal Modeling of HI-Storm 100S-218 Version B Storage Modules at Hope Creek Nuclear Power Station ISFSI. PNNL-22552; FCRD-UFD-2013-000297, Pacific Northwest National Laboratory, Richland, WA.

Adkins HE, Jr, TE Michener, and JM Cuta. 2013. "PNNL Experience Modeling Damaged Fuel and Spent Fuel Containers." Presented by Harold E. Adkins (Invited Speaker) at meeting with Japan Atomic Energy Agency representatives, Richland, WA on February 1, 2013. PNNL-SA-93193.

Cuta JM, and HE Adkins, Jr. 2013. Thermal Modeling of Hi-Storm 100S-218 Version B Storage Modules at Hope Creek Nuclear Power Station ISFSI. PNNL-22552 Rev. 1, Pacific Northwest National Laboratory, Richland, WA.

Adkins HE, Jr, KJ Geelhood, BJ Koeppel, NA Klymyshyn, and SE Sanborn. 2012. Used Nuclear Fuel Loading and Structural Performance Under Normal Conditions of Transport – Demonstration of Approach and Results on Used Fuel Performance Characterization. PNNL-22825, Pacific Northwest National Laboratory, Richland, WA.

Adkins HE, Jr, JA Fort, SR Suffield, JM Cuta, and BA Collins. 2012. "Thermal Modeling Studies for Active Storage Modules in the Calvert Cliffs ISFSI." In International High-Level Radioactive Waste Management Conference. PNNL-SA-92645, Pacific Northwest National Laboratory, Richland, WA.

Adkins HE, Jr. 2012. "CCNPS Canister Site Inspection Thermal Results." Presented by Adkins, Harold (Invited Speaker) at DOE UFDC Update Inspection Results, Charlotte, NC on November 26, 2012. PNNL SA-92639.

Adkins HE, Jr. 2012. "Modeling Used Fuel Storage Temperatures." Presented by Adkins, Harold (Invited Speaker) at Nuclear Waste Technical Review Board, Idaho Falls, ID on October 17, 2012. PNNL SA 91381.

NAME, AFFILIATION
Andrew Barto, U.S. Nuclear Regulatory Commission

CLASSIFICATION
Nuclear Engineer

FIELD OF EXPERTISE
Criticality safety and radiation shielding aspects of package designs for the transportation of radioactive material licensed under 10 CFR Part 71, and of storage system designs for spent nuclear fuel licensed under 10 CFR Part 72. Regulatory analysis of radioactive materials transportation package and spent fuel storage facility designs. NRC and international guidance on technical issues related to the transportation of radioactive material and storage of spent nuclear fuel.

EDUCATION
Bachelor of Science (Nuclear Engineering – 1997), University of Maryland; College Park

EXPERIENCE

U.S. Nuclear Regulatory Commission (1997 – Present) – Review and evaluate the criticality safety and radiation shielding aspects of package designs for the transportation of radioactive material licensed under 10 CFR Part 71, and of storage system designs for spent nuclear fuel licensed under 10 CFR Part 72. Determine the technical adequacy of radioactive materials transportation package and spent fuel storage facility designs to comply with regulatory standards. Prepare requests to the applicant in order to obtain additional information or clarification which is relevant to determining that the transportation package of spent fuel storage facility design meets NRC regulations. Document the results and conclusions of transportation package and spent fuel storage facility reviews and evaluations in safety evaluation reports.

Represent NRC and participate in meetings concerning criteria and standards for transportation packages and spent fuel storage facilities, to provide technical information and guidance relating to NRC requirements. Develop and revise NRC and international regulations governing the transportation of radioactive and fissile material and the storage of spent nuclear fuel. Develop NRC and international guidance documents on technical issues related to the transportation of radioactive material and storage of spent nuclear fuel. Assist on NRC inspections of facilities involved in radioactive and fissile materials transportation under 10 CFR Part 71 or spent nuclear fuel storage under 10 CFR Part 72.

SELECTED PUBLICATIONS

Barto, A; Rahimi, M; Burnup Credit in Spent Fuel Storage and Transportation: Regulatory Perspective on Current Status and Future Developments; PATRAM 2013 Conference; San Francisco, California; August 2013.

Barto, A; Parks, C; Bowen, D; Review of Fissile Material Exemptions and General Licenses in 10 CFR Part 71; PATRAM 2016 Conference; Kobe, Japan; September 2016.

Barto, A; Parks, C; Bowen, D; NUREG/CR-7239, "Review of Fissile Material Exemptions and General Licenses in 10 CFR Part 71," and Proposed Revisions to Harmonize with IAEA SSR-6 Fissile Material Requirements; PATRAM 2019 Conference; New Orleans, Louisiana; August 2019.

NAME, AFFILIATION

Paul Demkowicz, Ph.D., Idaho National Laboratory

CLASSIFICATION

Nuclear Science and Technology Directorate Fellow

FIELD OF EXPERTISE

TRISO fuel performance testing; nuclear fuels irradiation testing and post-irradiation examination; study of materials behavior in extreme environments

EDUCATION

Ph.D., Materials Science and Engineering, University of Florida
M.S., Materials Science and Engineering, University of Florida
B.S., Ceramic Engineering, University of Washington

EXPERIENCE

Technical Director for the U.S. Department of Energy Advanced Gas Reactor Fuel Development and Qualification Program, Idaho National Laboratory (2016 – Present) – Dr. Demkowicz has leads a diverse team of engineers and scientists in developing and testing TRISO fuel for high-temperature reactors.

Post-irradiation Examination Technical Lead, Idaho National Laboratory (2006 – 2016) – Developed methods and equipment for PIE of TRISO fuels. Led a team in performing a wide range of exams to evaluate the in-pile and high-temperature performance of TRISO fuel.

Senior Staff Engineer, Idaho National Laboratory (2003 – 2008) – Led projects to study material behavior in extreme environments, evaluate innovative fuel particle coating materials, and develop a Pb coolant test facility at INL.

SELECTED PUBLICATIONS

Paul A. Demkowicz, Blaise P. Collin, David A. Petti, Grant L. Hawkes, Binh T. Pham, Dawn M. Scates, Douglas E. Stacey, James W. Sterbentz, "AGR-2, the performance demonstration irradiation experiment of the Advanced Gas Reactor Fuel Development and Qualification Program" Annals of Nuclear Energy 150 (2021) 107833

Paul A. Demkowicz, John D. Hunn, "Two-Decade DOE Investment Lays Foundation for TRISO-Fueled Reactors in the US," Nuclear News, Vol. 63, August 2020

P.A. Demkowicz, D.A. Petti, K. Sawa, J.T. Maki, R.R. Hobbins, TRISO-Coated Particle Fuel Fabrication and Performance, In: Konings, Rudy JM and Stoller Roger E (eds.). Comprehensive Nuclear Materials 2nd edition, vol. 5 (2020) pp. 256-333. Oxford: Elsevier

P.A. Demkowicz, B. Liu, J.D. Hunn, "Coated particle fuel: Historical perspectives and current progress," J. Nucl. Mater., 515 (2019) 434-450 https://doi.org/10.1016/j.jnucmat.2018.09.044

Paul A. Demkowicz, John D. Hunn, David A. Petti, Robert N. Morris, "Key results from irradiation and post-irradiation examination of AGR-1 UCO TRISO fuel," Nuc. Eng. Des. 329 (2018) 102-109 https://doi.org/10.1016/j.nucengdes.2017.09.005

Robert N. Morris, Paul A. Demkowicz, John D. Hunn, Charles A. Baldwin, Edward L. Reber, "Performance of AGR-1 high temperature reactor fuel during post-irradiation heating tests," Nuc. Eng. Des. 306 (2016) 24-35

Paul Demkowicz, Karen Wright, Jian Gan, David Petti, "High temperature interface reactions of TiC, TiN, and SiC with palladium and rhodium," Solid State Ionics, 179 (2008) 2313-2321

Paul A. Demkowicz, James L. Jerden, Jr., Noriko Shibuya, Ronald Baney, James C. Cunnane, and James Tulenko, "Aqueous dissolution behavior of urania-thoria nuclear fuels," Nuclear Technology, 147, 2004, 157-170

NAME, AFFILIATION

James C. Kinsey, P.E., Idaho National Laboratory

CLASSIFICATION

Director of Regulatory Affairs

FIELD OF EXPERTISE
Development and implementation of key licensing strategies of light water and advanced (non-light water) reactors, including management of the DOE portion of the industry-led Licensing Modernization Project.

EDUCATION
B.S., Nuclear Engineering, University of Cincinnati, 1982. Course emphasis in commercial nuclear power thermal/fluids systems.

EXPERIENCE
Idaho National Laboratory, Regulatory Affairs and Program Development (7/08 – Present) – Originally joined the INL as the Regulatory Affairs Director for the NGNP Project, with responsibility for establishing and implementing the regulatory and commercial licensing strategy for this high-temperature gas cooled reactor technology. Those activities included the development and implementation of the project licensing strategy, management of NRC interfaces, and coordination of all project licensing activities being performed by the reactor designers.

Following NGNP, named by DOE as the National Technical Director (NTD) for the multi-laboratory regulatory affairs efforts being conducted within the Advanced Reactor Technology Program. In this role, responsible for establishing advanced reactor regulatory development strategies, increasing engagement and coordination with industry stakeholders, and implementing the resulting strategies through management of regulatory development activities at multiple DOE national laboratories. These efforts were primarily focused on resolving long-standing Nuclear Regulatory Commission policy and technical issues that have been restraining the development and deployment of advanced reactor technologies. Following a DOE restructuring of Office of Nuclear Energy programs, continued in this NTD role within the currently ongoing Regulatory Development portion of DOE's Advanced Reactor Demonstration Program (ARDP). A sampling of key accomplishments and successes from these NTD efforts that directly support the near-term deployment of advanced reactor technologies includes:

- Managed the DOE portion of the industry-led Licensing Modernization Project's risk-informed approach to advanced reactor design and licensing, which has been formally reviewed and endorsed by the NRC as an acceptable stakeholder approach to commercial licensing and deployment, resolving a decades-old Commission policy issue.
- Managed the development of regulatory proposals for "right-sizing" the Emergency Planning Zone requirements for advanced reactor technologies that are now being reflected in pending updates to NRC's regulatory requirements, which will resolve this longstanding Commission policy issue.
- Developed and managed a first-ever joint initiative between INL and the Electric Power Research Institute (EPRI) for the development and NRC submittal of a topical report regarding key research and fuel qualification approaches being developed within DOE's Advanced Gas Reactor (AGR) Program. These efforts resulted in formal NRC approval and endorsement of the approach, significantly reducing regulatory uncertainty for industry stakeholders planning to utilize the tristructural isotropic (TRISO) particle fuel form.

- Managed the development of proposals for alternatives to NRC's historical leaktight reactor containment requirements, resulting in NRC's endorsement of a more performance-based and right-sized "functional containment" alternative for advanced reactors, resolving another decades-old Commission policy issue.
- Nominated by industry to serve as the representative for DOE's national laboratories as a member of the Nuclear Energy Institute's New Reactor Regulatory Working Group, which is made up of senior management representatives from industry and is focused on developing strategic guidance on key generic licensing and regulatory issues that meets the industry's cost, timing and predictability needs.
- Managed the DOE portion of a multi-laboratory joint DOE-NRC initiative that established NRC-endorsed guidance for design criteria development for advanced reactors.

In addition to the above NTD responsibilities and associated accomplishments, also providing regulatory support for DOE's Gateway for Accelerated Innovation in Nuclear (GAIN) initiative, including familiarizing industry with the NRC's commercial licensing processes, and supporting public outreach efforts associated with new reactor deployments. Also leading a series of INL tasks being implemented in direct support of the NRC in their efforts to establish the requirements and associated regulatory guidance for an updated regulatory framework that addresses advanced reactor technologies, while also implementing Congressional direction provided in the Nuclear Energy Innovation and Modernization Act. These tasks include support for:

- Part 53 rule text and implementing guidance development
- Advanced reactor license application content development
- Advanced reactor construction inspection guidance development

GE-Hitachi Nuclear Energy, Vice President, ESBWR Licensing (2/05 – 7/08) –
Responsible for all licensing activities associated with the ESBWR design certification, as well as development of portions of the associated Combined License applications for NRC Staff review that provide for commercial deployment of the ESBWR technology. These activities included development and implementation of regulatory strategy, managing all direct interactions with the NRC Staff and the Advisory Committee on Reactor Safeguards, and issuance of ESBWR licensing submittals. These activities resulted in NRC approval and certification of the ESBWR design, which was one of only three successfully completed NRC design certification efforts in the past two decades.

Piedmont Management & Technical Services, Project Manager and Management Consultant (2/97 – 2/05) – Various Utility Clients

Nebraska Public Power District, Cooper Station (6/02 – 2/05) – Team lead for the performance of a multi-system internal assessment utilizing guidance contained in NRC inspection procedures covering "Safety System Design and Performance Capability" reviews. This assessment included a detailed review of major electrical systems and containment systems/components. Provided primary interface with the NRC regarding the assessment findings and their resolution.

Project manager for the multi-phase Service Water System Improvement Project, which included construction of major upgrades to the plant's river water supply systems and the Intake Structure facility. Responsibilities included development of the project plan and schedule, development of hardware technical requirements, contract development and management, technical oversight and interface with major equipment suppliers, and management of project implementation through the station's work control process.

American Electric Power, D.C. Cook Nuclear Station (3/99 – 5/02) – Managed selected senior management initiatives to improve the reliability of critical plant equipment and support future plant life extension initiatives, in accordance with INPO guidance documents. As Engineering Programs Manager, developed and implemented plans for the recovery and implementation of critical programs supporting plant restart associated with increased NRC oversight. This effort included the review and validation of plant modifications, procedures, specifications, event analyses, and associated licensing basis documents to confirm that selected systems can perform their safety and accident mitigation functions.

Illinois Power Co., Clinton Power Station (9/98 – 3/99) – Managed senior engineering staff in support of the Clinton Power Station's NRC "Watch List" recovery and restart effort, while working closely with the station's regulatory affairs and system engineering departments. Managed tasks included completion of portions of the System Design and Functional Verification project with a specific focus on the Emergency Core Cooling Systems, which entailed validation of system performance consistent with the design and licensing basis. Mentored plant staff in preparation for NRC inspections of the Engineering area and coordinated the successful response to previously identified inspection issues.

Detroit Edison Co., Fermi 2 Nuclear Station (10/97 – 9/98) – As project manager in support of the Licensing Department, developed the project plan and procedures for performing a successful validation of all design and licensing bases contained in the Updated Final Safety Analysis Report (UFSAR) in support of Detroit Edison commitments made in response to the NRC's 10CFR50.54(f) request. Provided direction, technical review, and project management oversight of the NSSS supplier (General Electric) performing the validation, including review of plant equipment modifications, construction activities, and historical licensing documents.

Commonwealth Edison Co., LaSalle County Station (2/97 – 10/97) – Project manager for the performance of the System Functional Performance Review at the LaSalle County Generating Station in support of NRC "Watch List" recovery to confirm that various Nuclear Steam Supply Systems were configured and tested consistent with their design and licensing bases. Served as the Licensing member of the multidiscipline Senior Independent Review Group that evaluated the impact and priority of review findings, and made plant restart recommendations to senior utility management, based on completed corrective actions.

Carolina Power & Light Co., Brunswick Nuclear Plant, Emergency Core Cooling System Sub-Unit Manager (8/95 – 2/97) – Supervised a team of system engineers and technicians responsible for the aggressive management of ECCS performance and reliability. Also managed all ASME Section XI programs, 10CFR50 Appendix J inspection/testing programs, and Maintenance Rule interfaces, including both program administration and implementation. Coordinated and completed a design and licensing basis reconstitution program for selected systems in support of unit power uprate. Assigned as Test Manager in charge of the Engineering and Control Room interface during initial power uprate testing. Served as a Maintenance Rule Expert Panel member.

IES Utilities, Duane Arnold Energy Center, Nuclear Licensing Supervisor (7/91 – 8/95) – Directly supervised department staff responsible for maintaining licensable operation of the plant. This included interpretation of NRC regulations, development of Technical Specification and Operating License changes, and the evaluation/resolution of generic issues. These responsibilities included regulatory issues (NRC Generic Letters, Bulletins, etc.) and direct oversight of industry organization interfaces (BWR Owners' Group, NEI, INPO). Coordinated department efforts in support of NRC Resident Inspector activities and NRC Region-based inspections. Served as the company spokesperson for primary interfaces with regulatory agencies and the local press.

Vermont Yankee Nuclear Power Corporation, Operations and Regulatory Support Engineer (11/85 – 7/91) – Operations and Regulatory Support Department: Served as primary point of contact for the NRC staff, including both region-based inspectors and headquarters personnel. Developed numerous evaluations and resolutions of licensing and technical issues including Technical Specification Proposed Changes, Licensee Event Reports, responses to NRC Inspection Reports, and technical support of corporate legal counsel. Acted as liaison with the State of Vermont nuclear oversight organization.

Duke Power Company, Catawba Nuclear Station, Operations Support Engineer (3/84 – 10/85) – Provided engineering support for the construction, pre-operational testing, and Operations department turnover of major plant systems in support of plant startup and initial operation.

The Babcock and Wilcox Company, Field Services, Startup and Test Engineer (7/82 – 3/84) – Principally responsible for installation and modifications to the Steam Generators, Pressurizers, and portions of the Reactor Coolant System piping during initial plant construction at the Midland Nuclear Plant. Developed and implemented initial startup testing procedures for the evaluation of the above systems.

NAME, AFFILIATION
Steven P. Nesbit, LMNT Consulting

CLASSIFICATION
Consultant

FIELD OF EXPERTISE
Nuclear fuel performance, thermal-hydraulic analysis, spent fuel management, storage and transportation technology, licensing.

EDUCATION
Bachelor of Science Nuclear Engineering, University of Virginia, 1980.
Master of Engineering, Nuclear Engineering, University of Virginia, 1982.

EXPERIENCE
LMNT Consulting, Charlotte, NC (2019 – present) – Founder and president. Providing support to a variety of clients in the areas of advanced nuclear energy systems, spent fuel management, and nuclear nonproliferation.

Duke Energy and Affiliates, Charlotte, NC (1982 – 2018) – In addition to extensive commercial nuclear utility experience, worked as a contractor on several high-visibility DOE projects, including the New Production Reactor Project, the Yucca Mountain repository project, the centralized interim storage project, and the surplus plutonium disposition project.

SELECTED PUBLICATIONS

EPRI-AR-1(NP)-A, *Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO)-Coated Particle Fuel Performance,* Product ID 3002019978, November 20, 2020.

Rod McCullum, Tom Brookmire, John Kessler, Suzanne Leblang, Adam Levin, Zita Martin, Steven Nesbit, Marc Nichol, and Terry Pickens, "Demonstrating the Safety of Long-Term Dry Storage," Waste Management 2013, Phoenix, AZ, February 2013.

Marc Nichol, Rodney McCullum, John Kessler, Keith Waldrop, Tom Brookmire, Paul Murray, and Steven Nesbit, "Concept Plan for a High Burn-up Fuel Storage and Transportation Confirmatory Data Project," ANS International High-Level Radioactive Waste Management Conference, Albuquerque, NM, April 2013.

Steven Nesbit, "Centralized Interim Storage – Does It Make Sense Today?" ANS International High-Level Radioactive Waste Management Conference, Albuquerque, NM, April 2

NAME, AFFILIATION

Gordon Petersen, Idaho National Laboratory

CLASSIFICATION

Spent Fuel Analyst

FIELD OF EXPERTISE

Packaging, transportation, and disposition of reactor waste streams.

EDUCATION

B.S. Nuclear Engineering, University of Tennessee, Knoxville, May 2014
M.S. Nuclear Engineering, University of Tennessee, Knoxville, May 2015
Ph.D. Nuclear Engineering, University of Tennessee, Knoxville, December 2016

EXPERIENCE

Spent Fuel Analyst at Idaho National Laboratory Idaho Falls, ID (2018 – Present)

- Coordinated and led multi-national laboratory effort on characterizing and assessing packaging options for advanced reactor waste streams
- Defined and articulated scope for $5M advanced reactor waste stream budget
- Lead author on Key M2 Milestone in the Spent Fuel Waste Disposition Campaign
- Control Account Manager for Fuel within the Research and Innovation Campaign within the Office of Spent Nuclear Fuel and High-Level Waste Disposition
- Managing research and development to advance knowledge and reduce liability in managing spent nuclear fuel.

- Control Account Manager for International Engagement for Consent Based Siting Campaign within the Office of Spent Nuclear Fuel and High-Level Waste Disposition.
- Managing efforts to coordinate and integrate international studies within the Spent Fuel and High-Level Waste Disposition campaign.
- Contributor to Back-End Management of Advanced Reactors (BEMAR)
- Subcommittee lead on cost estimating
- Manage the technical coordination to assure the demonstration spent nuclear fuel (SNF) packages loaded at Idaho National Lab can eventually be transported and dispositioned
- Coordinate between Environmental Management (EM) HQ and Idaho National Lab on road-ready dry storage
- Managing eleven work packages in system analysis, advanced reactors, and transportation of SNF
- Led two graduate research assistants at the University of Tennessee in projects involving advanced reactor and DOE-managed SNF
- Awarded “Superior” paper at waste management symposium
- Awarded “best graduate presentation” in fuel cycle, waste management, and decommissioning area at American Nuclear Society Student Conference.
- LDRD on developing software to model advanced reactor SNF in MOOSE
- Spent fuel packaging and disposal subject matter expert
- Developed a response for the Spent Nuclear Fuel Working Group (SNFWG) for the Nuclear Waste Technical Review Board’s (NWTRB) 2017 report on DOE-managed SNF
- Lead author working directly with EM SNFWG co-chair
- Analyzed DOE Standard Canister, the advanced neutron absorber (ANA), and the Idaho Spent Fuel Packaging Facility
- Summary reports on past work
- Path forward for DOE Standard Canister
- Criticality evaluations
- Analyzed the effects in transitioning Navy SNF from highly enriched to lower enriched uranium
- Mentored five interns on projects related to SNF

CTO at Babel Inc., (2016 – 2018)

- Developed web, Android, and IOS application for online real-estate
- Raised funds from investors
- Ran the day-to-day operations of the company including managing funds and drafting contracts.

Graduate Research Assistant at Oak Ridge National Laboratory (2014 – 2016)

- Performed cost benefit analysis on use of Standardized Canisters (STADs), the benefits of a consolidated interim storage facility, and different allocation strategies.
- Compared different optimization techniques for cost in order to develop an allocation strategy that benefited all reactors in terms of pick-up dates and minimized the total system cost.

SELECTED PUBLICATIONS

G. Petersen, U. Carvajal, A. Clark, B. Hanson, R. Torres, R. Cumberland, M. Billone, E. Mateo, L. Price, D. Sassani, Preliminary Analysis of Advanced Reactors Storage, Transportation, and Disposal, WM2024, March 2024.

G. Petersen, U. Carvajal, E. Mateo, L. Price, D. Sassani, R.Torres, B. Hanson, Advanced Reactors Spent Fuel & Waste Science and Technology Program, WM2024, March, 2024.

U. Carvajal, G. Petersen, R. Torres, M. Billone, B. Hanson, R. Cumberland, Bounding Pressure and Flammability Evaluations of Aluminum-Clad Spent Nuclear Fuel Department of Energy Standard Canisters, WM2024, March 2024.

E. Eidelpes, G. Petersen, Examining the Criticality and Dose Rate Aspect Associated with Storage and Concept of Operations for Advanced Reactor Spent Nuclear Fuel Management, WM2024, March 2024.

G. Petersen, et al. Storage, Transportation, and Disposal of Advanced Reactor Spent Nuclear Fuel and High-Level Waste, INL/RPT-23-76421, September 2023. DRAFT

G. Petersen, et. al, Scenario Summary Report FY23, August 2023.

J.Jarrell, G. Petersen, R. Fanning, A. Orrell, R. Howard, M. Nutt, J. Carter, J. Clarity, Updated Categorization of the Spent Nuclear Fuel Inventory, INL/RPT-22-65571 -Rev 1, June 2023.

S. Arm, G. Petersen, Concepts for Managing Advanced Reactor Spent Nuclear Fuel SNFWG Presentation, Presentation, Spent Nuclear Fuel Working Group, May 2023.

E. Eidelpes, G. Petersen, Bounding Pressure and Flammability Evaluations for a DOE Standard Canisters Loaded with ASNF, NL/RPT-22-68212 Rev: 002, May 2023

R. Joseph, J. Jarrell, G. Petersen, R. Cumberland, R. Howard, M. Nutt, T. Cotton, Consolidated Interim Storage Advantages and Disadvantages from Prior Reports and Studies, 2023 ANS Summer Meeting, June 2023.

J. Wing, G. Petersen, R. Joseph, G. Maldonado, Excess Criticality Mitigation for Pebble Bed Reactor Fuels by Employing Burnup Credit, INL/CON-23-7902, 2023 ANS Summer Meeting, June 2023.

L. Nguyen, I. Maldonado, G. Petersen, R. Joseph, In-Package Criticality Evaluation for Packages Containing Graphite Spent Nuclear Fuel in DOE Standard Canisters, ANS Summer Meeting, June 2023.

G. Petersen, et. al, Considerations for Managing DOE Standard Canisters within an Over-canister as Part of an Integrated Waste Management System, WM2023, March 2023.

G. Petersen, SFWD Seminar Series Advanced Reactor Overview – Status of Advanced Reactors and Their Waste Streams, Presentation Virtual, INL/MIS-23-71131, February 2023.

G. Petersen, Comparing Legacy Waste Management to Advanced Reactor Waste Management, Presentation, Management of Spent Fuel, Radioactive Waste, and Decommissioning in Small Modular Reactor/Advanced Reactor Technologies, Ottawa, Canada. November 2022.

G. Petersen, E. Eidelpes, Bounding Pressure and Flammability Evaluations for a DOE Standard Canisters Loaded with ASNF, Presentation, August 2022 NWTRB Meeting, virtual, August 2022

J. Wing, G. Petersen, R. Joseph, G. Maldonado, Examining the Criticality and Dose Rate Aspect Associated with Storage and Transportation of Spent Nuclear Fuel from Pebble Bed Reactors, WM2023, March 2023.

C. Boutros, S. Arm, H. Gadey, O. Garaburu, P. Ivanusa, R. Torres, M. Atz, R. Joseph, G. Petersen, E. Kitcher, R. Belles, D. Fairchild, R. Pierce, Advanced Reactor / Fuel Cycle Waste Management System Concepts – Fiscal Year 2022 Status, February 2023.

G. Petersen, R. Joseph, D. Thomas, S. Trost, C. Chandler, Examination of Requirements for Transport Consolidated Storage, and Disposal of the DOE Standard Canister in an Integrated System, INL/RPT-22-68585, August 2022.

P. Ivanusa, S. Arm, K. Kadooka, N. Kucinski, P. Stefanovic, G. Petersen, Initial assessment of Accident-Tolerant Fuel and its Effects on Storage and Transportation. August 2022.

G. Petersen, et. al, Scenario Summary Report FY22, August 2022.

G. Petersen, R. Joseph, Recommendations for IWM Advanced Reactor Fuel Cycle Back End Priorities INL/RPT-22-67998, 2022.

G. Petersen, L. Ward, B. Hartman, R.Elmetti Relating DOE's Programmatic Progress to the NWTRB Recommendations for DOE-managed Spent Nuclear Fuel, SNFWG-22-001, 2022.

R. Joseph, G. Petersen, K. Banerjee, B. Craig, Overview of System Integration Analysis Activities, International High-Level Radioactive Waste Management Conference (IHRLWM), November 2022.

G. Petersen, E. Eidelpes, A Review of N-stamping Requirement in Connection to the DOE Standard Canister, Proc. WM2022, Phoenix, AZ, March 2022.

P. Ivanusa, H. Gadey, P. Jensen, J. Carter, R. Howard, M. Nutt, B. Feng, M. Atz, R. Joseph, G. Petersen, E. Kitcher, R. Belles, W. Poore, Advanced Reactor / Fuel Cycle Waste Management System Concepts – Fiscal Year 2021 Status Report, January 2022.

K. Banerjee, P. Jensen, N. Klimyshyn, D. Richmond, A. Rigato, J. Clarity, N. Kucinski, P. Miller, P. Stefanovic, G. Petersen, Incorporation of Accident Tolerant Fuel into the Back End of the Fuel Cycle, August 2021.

G. Petersen, L. Vander Wal, Expanded System Analysis Capabilities for Non-Commercial SNF, Proc. WM2021, Phoenix, AZ, March 2021.

G. Petersen, K. Bulmahn, Preliminary Evaluation of the Pressure, Temperature, Flammability, and Reactivity of Peach Bottom Unit 1 and Fort Saint Vrain Spent Nuclear Fuel in a DOE Standard Canister, Proc. WM2021, Phoenix, AZ, March 2021.

E. Eidelpes, E. Kitcher, G. Petersen, Structural, Criticality, and Radiation Dose Calculations to Support SNF Loading into a DOE Standard Canister, WM2021, Virtual Meeting, March 2021.

G. Petersen, NGSAM Validation and Verification Report: Non-Commercial, INL/EXT-21-62849, May 2021.

R. Joseph, G. Petersen, J. Jarrell, FY21 Reference Interim Storage Facility Scenario for Systems Analyses, INL/EXT-21-64146, August 2021.

R. Joseph, R. Cumberland, G. Petersen, B. Craig, C. Olson, L. Vander Wal, Updated Waste Management System Analysis Simulation Tool Requirements, March 2021.

R. Joseph, R. Cumberland, G. Petersen, D. Sullivan, R. Howard, L. Vander Wal, Report Summarizing NGSAM Testing, January 2021.

G. Petersen, Idaho Spent Fuel Facility Summary Report, INL/EXT-20-58439, May 2020.

G. Petersen, Codisposal Waste Package Loading Options for DOE SNF and HLW, Proc. WM2020, Phoenix, AZ, March 2020.

G. Petersen, Damaged Spent Nuclear Fuel in the United States, Proc. WM2020, Phoenix, AZ, March 2020.

G. Petersen, Proposed NGSAM Updates for Expansion of DOE-Owned SNF Capabilities, INL/EXT-19-54119, June 2019.

G. Petersen, Damaged Fuel Management Practices at U.S. Reactor Sites, INL/EXT-19-54642, August 2019.

G. Petersen, Documenting Data and Requirements for DOE SNF and HLW, INL/EXT-19-55527, September 2019.

G. Petersen, Reevaluation of Post-Shutdown, At-Reactor Costs, INL/EXT-19-53388, April 2019.

G. Petersen, Evaluation of Neutron Absorbers in the DOE Standardized SNF Canister, INL/EXT-19-53193, Idaho Falls, ID, September 2019.

G. Petersen, S. Birk, K. Bulmahn, B. Carlsen, D. Daubaras, L. Montierth, R. Smith, History and Status of DOE's Standardized Canister, Proc. WM2019, Phoenix, AZ, March 2019.

G. Petersen, B. Carlsen, J. Jarrell, Neutron Absorber Considerations for the DOE Standardized Canister, Proc. ANS IHLRWM 2019, Knoxville, TN, April 2019.

J. Jarrell, B. Carlsen, G. Petersen, C. Shelton-Davis, P. Winston, A Plan to Prepare DOE-Managed Spent Fuel for Long-term Storage, Transportation, and Disposal, Proc. ANS IHLRWM 2019, Knoxville, TN, April 2019.

G. Petersen, S. Skutnik, J. Ostrowski & R. Joseph III, Determining Optimal Used Fuel Allocation Strategies, Nuclear Technology, 200:3, 208-224, DOI: 10.1080/00295450.2017.1377509 2017.

R. Joseph III, J. Jarrell, R. Cumberland, E. Kalinina, G. Petersen, R. Howard, W. Nutt, Standardized Canisters for Spent Nuclear Fuel: Their Potential Impact and a Proposed Path Forward, International Journal of Integrated Waste Management, Science and Technology, pg. 1 – 25, Vol. 2, No. 3, July 2017.

J. Jarrell, R. Joseph III, R. Cumberland, G. Petersen, E. Kalinina, Standardized Canisters and Their Potential Impact on an Integrated Waste Management System, Proc. ANS IHLRWM 2017, Charlotte, NC, April 9 – 13, 2017.

J. Jarrell, R. Joseph III, R. Howard, R. Cumberland, G. Petersen, W. M. Nutt, J. Carter, T. Cotton, Potential Cost Implications of an Interim Storage Facility for Commercial SNF, Proc. WM2017, Phoenix, AZ, March 5-9, 2017.

J. Jarrell, R. Joseph III, Riley Cumberland, G. Petersen, J. Fortner, E. Kalinina, T. Severynse, An Evaluation of Standardized Canisters in the Waste Management System, Proc. WM2016, Phoenix, AZ, March 6-10, 2016.

J. Jarrell, R. Joseph, R. Howard, G. Petersen, R. Cumberland, W. Nutt, J. Carter, T. Cotton, "Cost Implications of an Interim Storage Facility in the Waste Management System", FCRD-NFST-2015-000648 Rev. 1, ORNL/TM-2015/18, September 2016.

R. Joseph III, R. Hale, G. Petersen, R. Howard, M. Nutt, "Process Flow Diagrams and Node Descriptions for the UNF Waste Management System", Institute of Nuclear Materials Annual Meeting (July 2014).

NAME, AFFILIATION
Jason M. Piotter, U.S. Nuclear Regulatory Commission

CLASSIFICATION
Team Leader, NMSS Program Lead for Accident Tolerant and Advanced Reactor Fuels

FIELD OF EXPERTISE
Readiness and implementation of advanced nuclear fuels, criticality, shielding, and risk assessment of fuel facilities, fresh and spent fuel transport packages, spent fuel storage casks.

EDUCATION
Ph.D. Candidate – dissertation pending, Structural Engineering, Virginia Polytechnic Institute and State University
M.S., Structural Engineering, Virginia Polytechnic Institute and State University, 2001
B.S., Civil Engineering (Structures), University of Iowa, 1999
B.S., Exercise Science, University of Iowa, 1999

EXPERIENCE
Team Leader (formerly Senior Project Manager) - NMSS Program Lead for Accident Tolerant and Advanced Reactor Fuels, Division of Fuel Management, USNRC (June 2021 – October 2022, May 2023 – Present) – includes 9-month rotation to Fuel Facilities Licensing Branch

- Develop readiness and implementation plan to enable the safe use of new fuel technologies

- Maintained awareness of advanced fuels initiatives and technologies development including external stakeholders' plans for fuel enrichment, fabrication, transportation, and storage of advanced fuels.
- Developed recommendations as appropriate to Division Director and Office Director related to advanced and accident tolerant fuels.
- Managed the overall progress of advanced and accident tolerant fuel activities, facilitated team meetings, and conducted information gathering
- Communicated project progress to all stakeholders
- Supported development of budget input for advanced and accident tolerant fuels
- Coordinated preapplication activities for new advanced reactor fuel facilities

Acting Branch Chief, Nuclear Analysis and Risk Assessment Branch, Division of Fuel Management USNRC (October 2022 – May 2023)

- Responsible for supervising criticality, shielding and risk assessment reviewers of fuel facilities, fresh fuel transport packages, spent nuclear fuel storage casks and spent fuel transport packages
- Responsible for budget formulation and execution
- Planned and executed work plan for three Nuclear Regulatory Apprenticeship Network participants
- Performed interviews and selections to fill engineering vacancies

Senior Mechanical Engineer, Division of Fuel Management, USNRC (September 2013 – February 2016, November 2016 – May 2021)

- Responsible for thermal and containment technical reviews of spent nuclear fuel storage casks and transport packages
- Interfaced with industry groups on technical issues (NEI, EPRI)
- Technical monitor and COR for $2.5 million technical support contract at Pacific Northwest National Lab
- Mentored junior staff

Acting Branch Chief, Thermal and Containment Branch, Division of Fuel Management USNRC (February 2016 – November 2016)

- Responsible for supervising thermal and containment reviewers of spent nuclear fuel storage casks and transport packages
- Responsible for budget formulation and execution
- Responsible for guidance development and publication (Standard Review Plan for Storage, Fire Studies Compendium)

Senior Structural Engineer, (November 2008 – September 2013), Structural Engineer, Division of Spent Fuel Storage and Transportation, USNRC (January 2004 – November 2008)

- Staff expert in structural mechanics and finite element modeling
- Responsible for structural review of spent nuclear fuel storage casks and transport packages
- Technical Advisor – Private Fuel Storage hearings on aircraft impact
- Interface with industry groups (NEI, EPRI)
- NRC Representative to American Society of Mechanical Engineers
- Responsible for maintenance of classified and sensitive unclassified high and low fidelity computing facilities
- Technical monitor and COR for $4.2 million technical support contract at Pacific Northwest National Lab
- Mentor to junior staff

Structural Engineer (Rotation to Office of Research), Computational Modeling, USNRC (5/2006 – 10/2006)

- Developed Finite Element explicit dynamics model of transportation package
- Developed Finite Element explicit dynamics model of aircraft impact into containment structures

SELECTED PUBLICATIONS

G.S. Bjorkman, J.M. Piotter (2007), “Design Features that Enhance Spent Fuel Canister Integrity Under Drop Impact”, presented at SMiRT-19, Toronto, Canada, August 2007

G.S. Bjorkman, J.M. Piotter (2007), “Finite Element Mesh Considerations for Reduced Integration Elements, Structural Mechanics in Reactor Technology”, presented at SMiRT-19, Toronto, Canada, August 2007

N.A. Klymyshyn, H.E. Adkins, C.S. Bajwa, J.M Piotter (2010), “Package Impact Models as a Precursor to Cladding Analysis”, in Proceedings of ASME 2010 Pressure Vessels and Piping Conference, July 18-22, 2010, Bellevue WA

C.S. Bajwa, J.M. Piotter, J.M. Cuta, H.E. Adkins, N.A. Klymyshyn, J. Fort, S. Suffield (2010), “Best Practices for Finite Element Analysis of Spent Nuclear Fuel Transfer, Storage, and Transportation Systems”, 51st Annual Meeting of the Institute of Nuclear Materials Management, July 11-15, Baltimore MD

N.A. Klymyshyn, H.E. Adkins, J.M. Piotter (2015), “Closure Bolt Modeling for Seal Evaluation Under Extreme Thermal Loads”, in Proceedings of ASME 2015 Pressure Vessels and Piping Conference, July 19-23, 2015, Boston MA

H.E, Adkins, S. Suffield, N.A. Klymyshyn, J.M. Cuta, J.M. Piotter “SNF Transportation Package Response to MacArthur Maze Fire Scenario: Leak Rate Determination After Seal Failure”, in Proceedings of ASME 2015 Pressure Vessels and Piping Conference, July 19–23, 2015, Boston MA

NAME, AFFILIATION
Finis H. Southworth, N/A

CLASSIFICATION
Consultant

FIELD OF EXPERTISE
Nuclear Engineering Sciences, Nuclear Power, High Temperature Gas Cooled Reactors, Power Plant Operations and Maintenance, Fuel, and Applied R&D

EDUCATION
Ph.D. Nuclear Engineering Sciences, University of Florida.

EXPERIENCE
Framatome Inc., Chief Technology Officer (2006–2015)

Idaho National Laboratory (1990–2006) – Group Manager Fuel and Target Development for New Production High Temperature Gas Cooled Reactor (NP-MHTGR) (1990–1994), Manager Plutonium Focus Area (1994–1999), Manager Systems Engineering (1994–2005), Technical Director Generation IV HTGR Roadmap, System Steering Committee, AGR Fuel Program, and Next Generation Nuclear Project (2000–2005), (Director Project Management (2005–2006)

Florida Power & Light Company (1977–1990) – Supv. of Core Design for four nuclear units (1977-1984), Nuclear Operations Rotation to Turkey Point Nuclear Plant—various roles—QC Supv., Maintenance Supt., Technical Supt., Assistant Plant Manager, Plant Manager (1984–1989), Manager System Planning (2989–1990)

University of Illinois, Urbana, Illinois (1974–1977) – Assistant Professor of Nuclear Engineering and member of Fusion Studies Laboratory.

SELECTED PUBLICATIONS
Publications in Fusion supported by EPRI Contract on Advanced Fuel Fusion, Numerous topical reports while at FP&L on reactor physics methods for FP&L nuclear plants, and resolution of Pressurized Thermal Shock concerns with dramatic high energy flux reduction at vessel welds. Dozens of reports and papers on TRISO fuel by subordinates on NP-MHTGE at INL. Preparation of AGR Fuel Program Plan at INL (2003).

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PROGRAMS

Nuclear Power, P41

Advanced Nuclear Technology, P41.08.01

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