United States Nuclear Regulatory Commission
Protecting People and the Environment

RIL 2025-04

Proceedings of the 2024 Workshop on Storage and Transportation of TRISO and Metal Spent Nuclear Fuels

Virtual Workshop
December 3–5, 2024

Date Published: March 2025

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EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) held the 2024 Workshop on Storage and Transportation of TRISO and Metal Spent Nuclear Fuels as a virtual event on December 3 to 5, 2024. The workshop was held in coordination with the U.S. Department of Energy (DOE) Office of Nuclear Energy and the Electric Power Research Institute (EPRI), with assistance from the Center for Nuclear Waste Regulatory Analyses (CNWRA®).

The views and information presented in the workshop were provided and captured in this document for information exchange and do not establish or modify any regulatory guidance or positions of the U.S. NRC.

The workshop was focused on research on technical and regulatory considerations for spent nuclear fuel (SNF) management pertaining to advanced nuclear reactors—specifically focusing on tri-structural isotropic (TRISO) fuels and metal fuels. To prepare for potential future licensing and certification reviews, the NRC staff heard from industry experts and other researchers on their understanding and approaches to storage and transportation of these advanced reactor fuels. The NRC staff also conveyed regulatory process information to better prepare industry.

The three-day workshop was composed of 10 technical sessions with 36 presenters from a wide range of organizations, including universities, national laboratories, government agencies, nuclear vendors, nuclear industry, and advanced reactor developers. With 302 unique participants from across the globe, the workshop provided a forum for nuclear industry and advanced reactor stakeholders to discuss technical and regulatory issues related to storage and transportation of TRISO and metal advanced reactor fuels. Sessions provided an opportunity for NRC staff to gather information to help NRC assess the need for additional regulatory guidance updates and for attendees, including members of the public, to gain information and ask questions.

Sessions were arranged by technical areas, with each session including an opening presentation followed by expert presentations and discussions. The workshop sessions covered the following topics for storage and transportation of spent TRISO and metal fuels: SNF Structural Integrity; SNF Materials Performance; SNF Nuclear Physics/Neutronics; Experience and Projections; and Regulations, Guidance, and Crosscutting Topics.

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1 INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) held the 2024 Workshop on Storage and Transportation of TRISO and Metal Spent Nuclear Fuels as a virtual event on December 3 to 5, 2024. The workshop was held in coordination with the U.S. Department of Energy (DOE) Office of Nuclear Energy and Electric Power Research Institute (EPRI), with assistance from the Center for Nuclear Waste Regulatory Analyses (CNWRA®).

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The workshop was focused on research on technical and regulatory considerations for spent nuclear fuel (SNF) management pertaining to advanced nuclear reactors—specifically focusing on tri-structural isotropic (TRISO) and metal fuels.

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Sessions were arranged by technical areas, with each session including an opening presentation followed by expert presentations and discussions. Sessions also provided an opportunity for NRC staff to gather information to help NRC assess the need for additional regulatory guidance updates and provide an opportunity for members of the public to ask questions. The workshop spanned three days. On the first day of the workshop, Tuesday, December 3, 2024, Kathy Brock, Deputy Director, NRC Office of Nuclear Material Safety and Safeguards, opened the workshop with introductory remarks, followed by plenary session presentations from NRC, DOE, EPRI, and Idaho National Laboratory (INL). Tom Boyce from NRC moderated sessions on TRISO SNF structural integrity, and TRISO SNF materials performance. Wendy Reed from NRC moderated an additional session on TRISO SNF materials performance, followed by Hossein Esmaili, NRC, moderating a session on TRISO SNF nuclear physics / neutronics and the public question-and-answer session for the first day. On the second day of the workshop, Wednesday, December 4, 2024, Tekia Govan, NRC, moderated a third technical session on TRISO SNF materials performance followed by Jason Piotter, NRC, who moderated a session on metal SNF nuclear physics/neutronics. Tekia Govan, NRC, moderated a session on metal SNF materials performance and structural integrity and Jesse Carlson, NRC, moderated the end of day public question-and-answer session. On the third day of the workshop, Thursday, December 5, 2024, Laurel Bauer, NRC, moderated a session on experience and projection and Jose Cuadrado, NRC, moderated a session on regulations, guidance, crosscutting topics.

The workshop program is provided in Appendix A of this document. The presentation abstracts are provided as a booklet in Appendix B of this document. The presentations' slides are available in Appendix C of this document.

The following sections summarize the information from the workshop. The individuals listed as Additional Discussion Participants in these sections are based on the program, not based on actual attendance.

2 PLENARY SESSION

2.1 Session 1 Plenary Session

Presenters: Kathy Brock (NRC), Cinthya Roman (NRC), Paul Murray (DOE), Craig Stover (EPRI), Gordon Petersen (INL), Jason Piotter (NRC)

2.1.1 Session Summary

Speakers from NRC, DOE, and EPRI summarized work to prepare for TRISO and metal SNFs. DOE summarized the status of their activities related to SNF and high-level waste disposals. EPRI focused on their advanced reactor roadmap developed in collaboration with NEI. EPRI provided insights on the fuel management actions in the roadmap, including the intent to develop a spent fuel handling and storage strategy for multiple fuel types in the next four years. A summary of an EPRI-sponsored phenomena identification and ranking table (PIRT) on the storage and transportation of TRISO spent nuclear fuel was presented. The NRC staff discussed how its New Fuels Team has been working on developing infrastructure to support the regulation of new fuels activities.

2.1.2 Takeaways

- The NRC, DOE, industry, and national laboratories are each working to prepare for TRISO and metal spent nuclear fuels.

3 TRISO SNF SESSIONS

3.1 Session 2 TRISO SNF Structural Integrity

Presenters: John Stempien (INL),* Eddie Lopez Honorato (Oak Ridge National Laboratory) (ORNL), Tanner Mauseth (INL), Wen Jiang (North Carolina State University) (NCSU)
**Note: John Stempien was not able to attend the workshop, and his slides were presented by Lu Cai (INL).*

Additional Discussion Participants: Blaise Collin (Ultra Safe Nuclear), Benjamin Spencer (INL), Jonathan Wright (Ultra Safe Nuclear)

3.1.1 Session Summary

Presentations in this session focused on TRISO fracturing. Speakers were from INL, DOE, and North Carolina State University. In INL's advanced gas reactor (AGR) experiments, it was observed that accidents during post-irradiation handling (e.g., irradiation experiment disassembly) can occur, causing the fuel compacts to chip and fracture. The fracture path generally runs between the TRISO particles, but not always. John Stempien's presentation noted that fewer than 3.5% of broken particles were in the fracture plane of broken compacts for AGR 5, 6, and 7 at INL. Alternate materials and coatings were discussed that would allow for higher temperatures and higher effective full power days. The talks discussed fracture of the TRISO particles as well as radionuclide release considerations, but the development of material properties through experimentation, modeling methods, and analysis criteria were highlighted as areas where consensus is lacking. Of note was a lack of thorough understanding of the interaction between the matrix and the embedded TRISO particles, especially for a silicon carbide (SiC) matrix.

3.1.2 Presentations in this Session

- *Matrix Structural Integrity – Desirable and Undesirable Features of Matrix Materials for TRISO-based Fuels.* John Stempien, INL. *Note: John Stempien was not able to attend the workshop, and his slides were presented by Lu Cai (INL).*

- *Implications of New Coated Particle Fuels with New Architectures for an Expanded Service Envelope.* Eddie Lopez Honorato, ORNL.

- *Fracture Behavior Considerations for the TRISO Particle Matrix.* Tanner Mauseth, INL.

- *Modeling of TRISO and Matrix Fracture.* Wen Jiang, NCSU.

- *TRISO Particle Fracture – Importance of Strong Matrix and Careful Handling.* John Stempien, INL. *Note: John Stempien was not able to attend the workshop, and his slides were presented by Lu Cai (INL)*.

3.1.3 Notes from Live Discussion

Discussion of TRISO particle fracture in the INL matrix structural integrity presentation clarified that observed chips and fractures were an unintended result of testing. Other research on TRISO layers at ORNL indicates the deflection of cracking is dependent on the microstructure and roughness around the interface between the matrix and particles. ORNL is still in the early stages of these evaluations and noted additional characterization of the pyrolytic carbon (PyC) layer is needed and a clear timeframe for this work has not been established. Further discussion indicated the scale of fracture experiments was an important consideration; for example, there is a need to evaluate fractures in the matrix of TRISO compacts and within the particles in the compact matrix. INL research on fractures in the TRISO matrix indicated fracture properties depend on radiation conditions. A methodology has been developed for evaluating SiC layers and the capability exists for measuring effects. Discussion about TRISO response to drop test conditions (e.g., shock effects on matrix) specified in NRC transportation package approval standards indicated the evaluations of fracture properties in the TRISO particle matrix at INL have been limited to tensile testing. INL are not currently looking into other aspects, including drop test considerations, but have capabilities to do that. During a discussion of the varying conditions for fuel within a reactor relative to transportation and storage conditions, it was noted that the most extreme conditions for fuel would be within a reactor, the TRISO particles are robust, and data from reactor studies can be used in models to evaluate fuel performance under storage and transportation conditions.

3.1.4 Takeaways

- Experience has shown that while accidental damage during post-irradiation handling is possible, it is generally confined to small numbers of TRISO particles in elements composed of friable matrix.

- A large majority of failure mode analyses are focused on the TRISO particle itself. There are fewer studies investigating the mechanical interaction between the matrix and the embedded TRISO particles.

- AGR experiments observed that accidents during interfacility transfers via pneumatic rabbit and other post-irradiation handling during the course of post irradiation examination (PIE) can occur, causing compacts to chip and fracture. The fracture path generally runs between the TRISO particles, but not always.

- The development of material properties through experimentation, modeling methods, and analysis criteria was highlighted as a significant area where consensus is lacking.

- There is a lack of understanding of the interaction between the matrix and the embedded TRISO particles, especially for a SiC matrix.

- Research on the response of TRISO fuel to hypothetical accident conditions specified in NRC transportation package approval standards (e.g., drop test) can provide insights and identify potential challenges.

3.2 Session 3 TRISO SNF Materials Performance Part 1

Presenters: Tanner Mauseth (INL), Haiming Wen (Missouri S&T), John Stempien (INL), Wen Jiang (NCSU)

Additional Discussion Participants: Benjamin Spencer (INL), Steven Muller (NRC), Blaise Collin (Ultra Safe Nuclear), Jonathan Wright (Ultra Safe Nuclear), Jeffery Powers (BWX Technologies, Inc.), Eddie Lopez Honorato (ORNL).

3.2.1 Session Summary

Presentations addressed the performance of TRISO fuel and described evaluations of potential failure modes for TRISO fuel layers that could lead to fracture, creep, and delamination. Factors such as mechanical stress, oxidation, and radiation were evaluated. These presenters found that micro-tensile testing of TRISO fuel showed a decrease in tensile strength in inner pyrolytic carbon (IPyC) and SiC with increasing temperature. At the SiC-IPyC interface, this lower tensile strength (weak points) could lead to delamination or cracking under sufficient stress. The analysis of oxidation found that low pressure produces nonuniform oxide layers on SiC substrates at very high temperatures. Analysis of radiation and temperature effects found that without neutrons and at low temperatures, the multistep failure mode for SiC would not occur. Even with IPyC and SiC failure, outer pyrolytic carbon (OPyC) can remain intact and retain fission gas. It was noted that high temperatures are needed for metallic fission products to migrate out of the SiC layer. Limitations of models in existing fuel performance codes were described. Further development and validation of multi-scale TRISO modeling with the BISON code would enable simulation results that could be useful to reactor vendors. For fuel analysis in the BISON code, time-dependent failure analysis was highlighted as needed to understand long-term storage. Additional work to control crack growth was also recommended.

3.2.2 Presentations in this Session

- *Micro-Tensile Properties of Irradiated AGR-2 TRISO Fuel Pyrolytic Carbon (PyC) and Silicon Carbide (SiC) Coatings.* Tanner Mauseth, INL.

- *Oxidation Behavior of the SiC Coating of TRISO Fuel Particles in Air.* Haiming Wen, Missouri S&T.

- *PyC Creep and SiC Fracture – out-of-pile PyC creep should be zero as should SiC fracture.* John Stempien, INL. *Note: John Stempien was not able to attend the workshop, and his slides were presented by Lu Cai (INL)*.

- *Time-Dependent Weibull Failure Analysis of TRISO Fuel.* Wen Jiang, NCSU.

3.2.3 Notes from Live Discussion

A question was raised as to whether BISON for fracture mechanics applications is sufficiently developed for regulatory applications. Multiple participants agreed that simpler methods and techniques should be used now and BISON can be increasingly used as it develops further in these areas. There was additional discussion related to the presented information that 95% of fission gasses are retained in compacts after irradiation, and dry storage would not result in further degradation. Confirmation of this conclusion was provided, and no significant release from the fuel is seen to occur if the PyC layer is intact. Additionally, the fission gasses diffuse so slowly that they are not expected to diffuse through the PyC layer over the time scale studied.

3.2.4 Takeaways

- Factors associated with failure modes such as mechanical stress, oxidation, and radiation have been evaluated.

- Current research is focused on TRISO fuel performance under in-reactor conditions (e.g., high-temperature and radiation) that are generally more extreme than conditions expected under storage and transportation.

- Further work was recommended to address limitations of models in existing fuel performance codes with research, including development and validation of multi-scale TRISO modeling with the BISON code and time-dependent failure analysis, including control of crack growth, to understand long-term storage.

3.3 Session 4 TRISO SNF Materials Performance Part 2

Presenters: Rebecca E. Smith (INL), Lu Cai (INL), David Arregui-Mena (ORNL)

Additional Discussion Participants: Blaise Collin (Ultra Safe Nuclear), Jonathan Wright (Ultra Safe Nuclear), Benjamin Spencer (INL), Joseph Bass (NRC), Jeffery Powers (BWX Technologies, Inc.), Eddie Lopez Honorato (ORNL), and Public Attendees.

3.3.1 Session Summary

Presentations focused mostly on oxidation of graphite and graphitic matrix material (which can surround TRISO fuel pebbles when they are used in larger fuel elements/blocks). Rebecca Smith described the properties of graphite and noted that oxidation is temperature dependent and positively associated with temperature. She conveyed that mass loss is much lower at 500 ºC versus 750 ºC. Mass loss is also positively associated with irradiation. She further noted that above 1000 ºC carbon monoxide release could be a concern and stressed the importance of focusing on actual conditions to determine mass loss. Lu Cai described studies to characterize the oxidation of matrix graphite. She conveyed that oxidation rates at 500 ºC or below are low and that mass loss due to irradiation is also low. J. David Arregui-Mena described oxidation of graphitic components under accident conditions. He noted that acute oxidation occurs if air gets into the core and described laboratory simulation of accident conditions using a furnace to expose graphite samples to elevated temperatures up to 1600 ºC. The study described by Arregui-Mena showed that oxidation increases with temperature and graphite material becomes more porous as oxidation proceeds.

3.3.2 Presentations in this Session

- *Safety Considerations for Irradiated Graphite.* Rebecca E. Smith, INL.
- *Determining the Oxidation Behavior of Matrix Graphite.* Lu Cai, INL.
- *Oxidation of Graphitic Components Under Accident Conditions.* J. David Arregui-Mena, ORNL.

3.3.3 Notes from Live Discussion

The discussion focused on what might be characterized as the weakest link for TRISO fuel. Experts noted that the matrix would degrade faster than the layers under the same oxidizing conditions, but also that more reliance on TRISO layers for containment is a reason to focus on the layers.

3.3.4 Takeaways

- Oxidation of graphite is slow and can gradually degrade the material properties of the remaining material. Irradiated graphite may oxidize at double or triple the rate of the same grade of unirradiated graphite.
- Matrix graphite materials may experience preferential oxidation of the non-graphitic carbon.
- Experts noted that the matrix would degrade faster than the layers under the same oxidizing conditions but also that more reliance on TRISO layers for containment is a reason to focus on the layers.

3.4 Session 5 TRISO SNF Nuclear Physics / Neutronics

Presenters: Andrew Bielen (NRC), Laura Price (Sandia National Laboratories) (SNL), Gordon Petersen (INL), Andrew Barto (NRC)

Additional Discussion Participants: Taek K. Kim (Argonne National Laboratory) (ANL), Pavlo Ivanusa (Pacific Northwest National Laboratory) (PNNL), Blaise Collin (Ultra Safe Nuclear), Jonathan Wright (Ultra Safe Nuclear), Justin Clarity (PNNL), Jeffery Powers (BWX Technologies, Inc.), Sven Bader (Orano Federal Services LLC), Steven Nesbit (LMNT Consulting), Eddie Lopez Honorato (ORNL).

3.4.1 Session Summary

Presentations discussed decay heat and neutronics considerations for storage and transportation of TRISO fuels. Presentations highlighted that the use of TRISO fuel is not entirely new, and the NRC licensed the storage and transportation of spent TRISO fuel at Fort St. Vrain (FSV) using older codes and data. Uncertainties seem to be understood for reactors using TRISO fuel and can be attributed primarily to uncertainties in nuclear data (such as cross section uncertainties for U-235, U-238, and graphite*)* and modeling parameters, such as uncertainties in irradiation history as a pebble traverses a core, or to model uncertainties from lacking knowledge of proprietary information. It is anticipated that new codes and data may reduce uncertainties and that additional uncertainty margins could be reduced as more code validation becomes available. Modeling of SNF TRISO packages found that the produced decay heat of TRISO fuel may be an order of magnitude lower per package than typical light water reactor (LWR) fuel, and comparison studies that treated TRISO storage similarly to typical LWR fuel seem to indicate that criticality may be a limiting factor in the design of TRISO storage casks and transportation packages relative to decay heat. Finally, the NRC maintains a suite of codes to assist in the assessment of decay heat, criticality and neutron multiplication, and shielding and radiation protection of LWR and non-LWR fuel and have conducted demonstration calculations to support licensing reviews.

3.4.2 Presentations in this Session

- *NRC's Simulation Capabilities Supporting Criticality, Reactor Physics, Decay Heat, and Shielding for TRISO-particle Fueled Non-LWRs.* Andrew Bielen, NRC.

- *TRISO and Metal Spent Nuclear Fuels Decay Heat.* Laura Price, SNL.

- *Modeling Capabilities for TRISO and Metallic SNF. Gordon Petersen, INL.*

- *Licensing Experience with TRISO Spent Fuel – A Historical Perspective: Fort St. Vrain Independent Spent Fuel Storage Installation (ISFSI).* Andrew Barto, NRC.

3.4.3 Notes from Live Discussion

Regarding NRC's simulation capabilities, there was a question about whether the two-dimensional ORIGEN model that the NRC uses could be simplified to study the sensitivity of the modeling to the level of detail. NRC staff indicated a less detailed model is not an option because of the spectral boundary. When asked if MELCOR has been applied to TRISO storage and transportation scenarios, the NRC staff indicated the code was sufficiently flexible to address storage and transportation scenarios. Regarding the analysis of decay heat, a comment was made that similar analyses had been conducted, and criticality was found to be the limiting characteristic for TRISO spent fuel. Discharging spent fuel directly to a storage canister without a cooling period was not addressed in the SNL decay heat study, but was acknowledged as a possibility given that TRISO reactor designs do not include pool storage. Direct discharge to storage would increase the decay heat loading to the storage cask. Additional discussion on the FSV TRISO storage experience focused on storage system design and cooling method, operational details such as radiation doses, whether any transportation challenges were found, and the potential for radiolytic corrosion of canisters during storage. The results of radiation monitoring were expected to be documented, and no specific transportation challenges were noted. It was noted the FSV spent fuel was much cooler than typical LWR spent fuel when it was transported. The radiolytic corrosion issue was associated with conditions specific to Europe that were not applicable to the FSV storage facility.

3.4.4 Takeaways

- The use of TRISO fuel is not entirely new and the NRC licensed the storage and transportation of spent TRISO fuel at FSV using older codes and data.

- A major component of modeling uncertainty can be attributed to uncertainties in nuclear data (such as cross section uncertainties for U-235, U-238, and graphite) and modeling parameters, such as uncertainties in irradiation history as a pebble traverses a core and design uncertainties from the unavailability of proprietary information.

- Modeling of TRISO SNF packages found that the produced decay heat of TRISO fuel may be an order of magnitude lower per package than typical LWR fuel.

- Criticality may be a limiting factor in the design of TRISO storage casks and transportation packages relative to decay heat. The importance of shielding was emphasized. Accounting for burnup would be expected to reduce the estimated reactivity and potential for criticality.

- TRISO SNF can be modeled using existing nuclear codes to assess radiation protection and maintaining subcriticality.

3.5 Session 6 TRISO SNF Materials Performance Part 3

Presenters: James Corson (NRC), Umapathy R Ganjigatte (Indian Institute of Technology Delhi, New Delhi, India & Inter University Accelerator Center, New Delhi, India)

Additional Discussion Participants: Rebecca E. Smith (INL), Lu Cai (INL), J. David Arregui-Mena (ORNL), Blaise Collin (Ultra Safe Nuclear), Jonathan Wright (Ultra Safe Nuclear), Benjamin Spencer (INL), Joseph Bass (NRC), Jeffery Powers (BWX Technologies, Inc.), Wen Jiang (NCSU), Eddie Lopez Honorato (ORNL).

3.5.1 Session Summary

The NRC staff summarized application of the FAST code to non-LWR fuel performance analyses. This included considerations for FAST-TRISO code applicability, development, and analyses. The development effort is considering TRISO-specific processes, including addressing particles and layers, heat transfer, stresses, fission product transport, and failure modes. The challenges in developing FAST-TRISO include applying FAST models that are valid at higher temperatures to the lower temperatures expected during storage and transportation. Developing inputs, including for representative temperatures during storage and transportation, was also emphasized. A presentation about doping and irradiating SiC evaluated doped coatings on TRISO fuel that can result in an oxidation-resistant layer. The evaluation of doped coatings considered melting point, thermal conductivity, thermal expansion, oxidation resistance, and high-temperature strength. Material response to high-energy irradiation was evaluated and post-irradiation defects were noted. Analysis of test samples indicated some post-irradiation defects. Results showed variation in performance characteristics among doping materials.

3.5.2 Presentations in this Session

- *US NRC Modeling for TRISO Material Performance.* James Corson, NRC.

- *Effects of Rare Earth Doping and High-Energy Irradiation in Silicon Carbide for Advanced Nuclear Applications.* Umapathy R Ganjigatte, Indian Institute of Technology Delhi, New Delhi, India & Inter University Accelerator Center, New Delhi, India.

3.5.3 Notes from Live Discussion

Discussions addressed the plans for FAST-TRISO development: including analysis of failure probabilities, design-basis accident (DBA) validation, available validation data in the AGR project at INL, and expanding the code to multiple dimensions. The discussion indicated that FAST could be applied to evaluating research reactor or other small reactor TRISO fuels, as well as design-basis accident conditions. In response to questions, NRC staff provided the location of FAST-TRISO documentation in ADAMS as ML21175A151. One of the challenges in validating FAST-TRISO is the large number of TRISO particles, but a combination of code analyses and experimental data are available to support validation. The importance of considering the initial state of TRISO layers was discussed and finite element modeling was mentioned to estimate residual stresses. Failure rates for TRISO particles have been developed as part of the DOE AGR and related studies. Further discussion of doping TRISO particles noted difficulties associated with thermal expansion of niobium, aluminum, and titanium and clarified that doping could be applied before or after fuel irradiation.

3.5.4 Takeaways

- The NRC staff is applying the FAST code to non-LWR fuel performance analyses.
- Plans for development of the FAST code simulations for TRISO SNF are focused on failure probabilities, fission product diffusion, and code validation.
- Doped outer chemical layers or coatings on TRISO fuel can provide an oxidation-resistant layer.
- Variation in performance was noted for different doping materials.

4 METAL SNF SESSIONS

4.1 <u>Session 7 Metal SNF Nuclear Physics / Neutronics</u>

Presenters: Andrew Barto (NRC), Andrew Bielen (NRC), Gordon Petersen (INL), Laura Price (SNL)

Additional Discussion Participants: Taek K. Kim (ANL), Justin Clarity (PNNL), Sven Bader (Orano Federal Services LLC), Steven Nesbit (LMNT Consulting).

4.1.1 Session Summary

Storage experience of metal SNFs exists from EBR-I, EBR-II, FFTF, and Fermi-1. Untreated spent fuel equates to over 20 metric tons of heavy metal existing in storage. Data are lacking on long-term storage of metallic fuel. NRC staff discussed the limited history of licensing metallic SNF. While there are some historical data on metallic SNF and related activities, the use of metallic fuels was not commercialized and there has thus far been little need for additional data. There have been occasional packages used for transportation of metal fuel, but these packages may not be optimized for scaled operations expected in commercial uses. Data for sodium fast reactor (SFR) fuel are limited, but packages may look like those used for LWR transportation based on comparisons between fuel forms. NRC staff also summarized existing capabilities to conduct analyses of decay heat, neutron multiplication and criticality, and shielding and radiation protection of metal fuels in a storage and transportation context. The results of other package analyses of metal fuel addressing criticality, dose, and decay heat in the context of storage and transportation were presented, as was a separate comparison of canistered metal and LWR fuel decay heats.

4.1.2 Presentations in this Session

- *10 CFR Part 71 - Certification of Transportation Packages for Metal Fuel.* Andrew Barto, NRC.
- *NRC's Simulation Capabilities Supporting Criticality, Reactor Physics, Decay Heat, and Shielding for Metallic Fueled Non-LWRs.* Andrew Bielen, NRC.
- *Modeling Capabilities for TRISO and Metallic SNF.* Gordon Petersen, INL.
- *TRISO and Metal Spent Nuclear Fuels Decay Heat.* Laura Price, SNL.

4.1.3 Notes from Live Discussion

Discussions addressed simulation capabilities related to decay heat, criticality, and radiation protection. NRC staff indicated that libraries are provided with SCALE and ORIGEN codes and that users can also create their own libraries, if desired. Regarding validation of depletion calculations for burnup credit, NRC staff indicated that an approach being explored for metal fuels is the possibility of burnup credit with available supporting validation data, similar to a system that currently exists for LWR fuel. It was noted that supporting data are limited to burnups of 60-100 Gigawatt-day/metric ton uranium currently and experimental data do not support higher burnups. Discussion of current packaging options for metal fuel indicated no knowledge of commercial options. Analysis of metal fuel in LWR-sized packages indicated no issues with decay heat or criticality.

4.1.4 Takeaways

- Storage and transportation experience is limited due to lack of past commercialization.
- Metallic SNF can be modeled using existing nuclear codes to assess radiation protection and maintain subcriticality.
- Available data for validation of depletion calculations for burnup are limited to burnups of 60-100 Gigawatt-day/metric ton uranium.
- Analysis of metallic fuel in LWR-sized packages indicated no issues with decay heat or criticality.

4.2 Session 8 Metal SNF Materials Performance and Structural Integrity

Presenters: James Corson (NRC), Tiankai Yao (INL), Walter Williams (NRC), Stuart Arm (PNNL), Steven D. Herrmann (INL), Jamie Noel (University of Western Ontario)

Additional Discussion Participants: Benjamin Spencer (INL), Sven Bader Orano (Federal Services LLC)

4.2.1 Session Summary

The NRC staff discussed development of the FAST code for metallic fuel performance analyses. It was noted that data for metallic fuel under storage conditions would enable further development of the code in this area. The next presenter discussed fission product diffusion, corrosion of cladding materials, and interactions between metallic cladding and water. A presentation on metal fuel swelling showed that asymmetric fuel swelling can occur based on the state and composition of the uranium. There is a decrease in swelling with increasing plutonium due to fission product phase transitions. Data are not currently available on the geometric dependency of metallic fuel designs. Issues with metallic fuel, including fuel cladding interaction mechanisms (FCMI and FCCI), were described including potential and hypothetical issues around these technical concerns. INL research and testing of treatment options for sodium-bonded metal spent fuel from EBR-II and fuel blankets from Fermi-1 were also described.

4.2.2 Presentations in this Session

- *U.S NRC Modeling Capabilities of Metal Fuel in FAST.* James Corson, NRC.
- *Fission Product Diffusion.* Tiankai Yao, INL.
- *Corrosion of Cladding Materials.* Tiankai Yao, INL.
- *Interactions Between Metallic Fuel and Water.* Tiankai Yao, INL.
- *Fission Product Induced Metal Fuel Swelling.* Walter Williams, NRC.
- *Assessment on Metal Spent Nuclear Fuel Swelling Effects on Structural Integrity.* Walter Williams, NRC.
- *Potential Treatment Options for Sodium-Bonded Metal Fuel.* Stuart Arm, PNNL.
- *Removal and Deactivation of Bond Sodium from Fast Reactor Materials.* Steven D. Herrmann, INL.
- *Materials Interactions Leading to Enhanced Dissolution or Protection of Spent Fuel in Long-Term Storage.* Jamie Noel, University of Western Ontario.

4.2.3 Notes from Live Discussion

Discussions focused on the sodium chemical process and the different ways that sodium can be removed from the fuel. Sodium infiltration into fuel was noted as a factor that complicates processing. Infiltration increases with burnup and infiltration in higher burnup metal fuel relative to EBR-II burnup was noted as a topic that should be investigated. The scale of treatment demonstrations was discussed as small relative to the scale needed to support the deployment of commercial reactors.

4.2.4 Takeaways

- The FAST code can be used for modeling fission gas production and diffusion within metallic fuels.

- Fission product diffusion in metallic fuel directly impacts the fuel constitutional redistribution.

- Interaction between metallic fuel and water is a safety concern. The reaction is highly exothermic with a significant amount of heat being released. The reaction can lead to fuel damage with rapid temperature increase and volumetric expansion.

- Treatment options for removal of sodium from discharged fuel have been demonstrated but not at the scale likely needed to support the deployment of commercial reactors.

- Infiltration of sodium into the fuel is associated with burnup and complicates processing to remove sodium.

5 ADDITIONAL TOPICS SESSIONS

5.1 Session 9 Additional Topics, Part 1: Experience and Projections

Presenters: Ralf Schneider-Eickhoff (BGZ Gesellschaft für Zwischenlagerung mbH), Maik Stuke (BGZ Gesellschaft für Zwischenlagerung mbH), Bret Leslie (U.S. Nuclear Waste Technical Review Board), Taek K. Kim (ANL), Jesse Sloane (Deep Isolation), Steve Sisley (NAC International)

Additional Discussion Participants: Blaise Collin (Ultra Safe Nuclear), Jonathan Wright (Ultra Safe Nuclear), Steven Nesbit (LMNT Consulting), Andrew Barto (NRC), Jason Piotter (NRC), Paul Cantonwine (ORNL), Sven Bader (Orano Federal Services LLC), Matt Featherston (X-Energy, LLC), Stephen Vaughn (X-Energy, LLC), Prakash Narayanan (ORANO TN), Eddie Lopez Honorato (ORNL), Rod McCullum (Nuclear Energy Institute)(NEI), Steven Maheras (PNNL).

5.1.1 Session Summary

Speakers presented on various topics pertaining to storage, transportation and disposal of advanced reactor fuel. One presentation summarized the operating experience in Germany pertaining to the storage of pebble bed reactor fuel, including insights into the identification of damaged fuel. Another presentation addressed the work of the U.S. Nuclear Waste Technical Review Board (NWTRB) in evaluating the technical and scientific validity of DOE activities related to SNF and high level waste (HLW). Information was presented about work being conducted at ANL related to the storage and transportation of TRISO fuel, including microreactors, potential volume challenges relating to the storage of waste, and criticality considerations in the event of a microreactor flooding. Finally, Deep Isolation and NAC provided an overview of their Universal Canister System development and the assessments being conducted under an Advanced Research Projects Agency – Energy (ARPA-E) funded project. This included consideration of the varied waste forms that that may need disposal, and the performance assessment models being developed.

5.1.2 Presentations in this Session

- *Dry Storage of THTR Spent Fuel in Germany.* Ralf Schneider-Eickhoff, Maik Stuke, BGZ Gesellschaft für Zwischenlagerung mbH.

- *Management and Disposal of U.S. Department of Energy's TRISO- and Metallic-based Spent Nuclear Fuel and Preliminary Considerations for Waste Resulting from Advanced Nuclear Reactors.* Bret Leslie, U.S. NWTRB.

- *Projection of TRISO Spent Nuclear Fuels and Related Issues.* Taek K. Kim, ANL.

- *Management of TRISO spent fuel using a Universal Canister System.* Jesse Sloane, Deep Isolation. Steve Sisley, NAC International.

5.1.3 Notes from Live Discussion

Discussion on a variety of topics included an interest in learning more about the technology used in Germany for identifying and separating damaged fuel pebbles prior to storage, the potential need for a new TRISO spent fuel storage cask, potential conditioning of TRISO spent fuel to reduce the volume of waste needing disposal, and the consideration of sodium-bonded fuel disposal from a technical and regulatory standpoint.

5.1.4 Takeaways

- There is some pebble bed spent fuel management experience in Germany for the transportation and storage of THTR fuel (similar but not identical to TRISO fuel), including the loading of steel canisters, use of dual-purpose casks, and storage of fuel in a managed facility for over 30 years.

- The NWTRB has made several findings, conclusions, and recommendations that apply to storage, transportation, and disposal of SNF from advanced reactors.

- Evaluation of potential fuel use scenarios for advanced reactors illustrates a higher volume of SNF per unit electricity generation for TRISO pebbles, resulting in a demand for a larger number of storage and transportation canisters relative to other advanced and LWR fuels.

- A Universal Canister System (UCS) being developed by Deep Isolation and its collaborators with support from DOE's ARPA-E aims to enable the safe storage, transport, and disposal of advanced reactor waste streams, including TRISO spherical pebbles, cylindrical compacts, and full prismatic assemblies, in either conventional mined repositories or deep boreholes.

- Discussion of topics related to "damaged fuel" requirements highlighted a need for further consideration of the concept of "damaged fuel" in the contexts of new fuels.

5.2 Session 10 Additional Topics, Part 2: Regulations, Guidance, Crosscutting Topics

Presenters: Steven Maheras (PNNL), Travis Chapman (BWX Technologies, Inc.), Prakash Narayanan (ORANO TN), Rod McCullum (NEI)

Additional Discussion Participants: Blaise Collin (Ultra Safe Nuclear), Jonathan Wright (Ultra Safe Nuclear), Steven Nesbit (LMNT Consulting), Andrew Barto (NRC), Jason Piotter (NRC), Paul Cantonwine (ORNL), Sven Bader (Orano Federal Services LLC), Matt Featherston (X-Energy, LLC), Stephen Vaughn (X-Energy, LLC), Eddie Lopez Honorato (ORNL).

5.2.1 Session Summary

The last session started with a presentation on microreactor transportation emergency planning challenges. Cross-cutting challenges include hazardous materials in microreactor designs, criticality control during transport, fuel type issues, potential compensatory measures, and emergency response training. It was noted that microreactors shipped within a short time of operation may not meet the 10 millirem per hour at 2 meters standard (49 CFR 173.441 and 10 CFR 71.47). An elevated dose rate from a conveyance could present emergency response issues. External engagement for emergency planning could require 2 to 3 years. DOE is working closely with DoD, the Army Reactor Office, the Army Office of Chief of Engineers, and the National Reactor Innovation Center (NRIC). Principal design criteria (PDC) for transportable reactors were presented, showing how PDCs tie all the domains together for regulatory acceptance and differences between how developers and regulators use the PDCs were highlighted. Challenges associated with developing PDC for transportable reactors include the review approach, curation of “licensed” activity scope, technology approach to storage and transportation activities, and differences in evaluation approaches. The potential use of microreactors as storage systems was discussed. Development efforts indicate defueling may be needed. Designing for long-term storage or disposal of a microreactor without fuel removal may be too ambitious for an engineered system. Additionally, design for direct disposal would add waste volume, and would be another factor for considering defueling. A presentation on system design and safety analysis associated with storage and transportation emphasized differences between LWR and advanced reactor fuels and how to adapt existing storage and transportation technology to these new fuels. Safety criteria related to fission product barriers, criticality control, fuel design, containment/confinement, heat removal, and radiation protection were emphasized as key factors that need to be considered in the context of new fuel characteristics and the need to update or adjust related guidance or regulations. Examples of advanced fuel characteristics that may inform guidance updates included containment/confinement functions addressed by TRISO fuel layers rather than cladding; reduced criticality concerns with low power density fuels; and the related possibility for benchmark validations for burnup credit for new fuel types which may have the potential to reduce the effect of conservative assumptions on capacity optimization.

5.2.2 Presentations in this Session

- *Microreactor Transportation Emergency Planning Challenges.* Steven Maheras, PNNL.

- *Cross-domain Development of Principal Design Criteria for Transportable Reactors.* Travis Chapman, BWX Technologies, Inc.

- *System design and safety analysis associated with storage and transportation.* Prakash Narayanan, ORANO TN.

- *Building on Established Knowledge to Inform the Regulatory Framework for TRISO and Metal Spent Nuclear Fuels.* Rod McCullum, NEI. *Note: This presentation was not presented due to unavailability of the speaker, but the speaker selected to retain the abstract and slides within the proceedings when provided the option, such that the abstract and slides are available to the public.*

5.2.3 Notes from Live Discussion

Live discussion highlighted design challenges for microreactor storage and transportation, including whether to design the reactor for transportation or design transportation packaging that fit the reactor. Some considerations included the increased in-transit safety protections needed at the end of reactor life relative to the beginning (i.e., comparing after irradiation with spent fuel to before irradiation with fresh fuel). The increased protection needed at the end of reactor life may favor approaches that add more robust packaging when it is needed. It was noted that, depending on the source of fresh uranium for the fuel, the front-end transportation of a microreactor may not meet Type A limits (e.g., if recycled uranium with impurities that impart additional radioactivity is used for fuel). Additional discussion on emergency planning considered how emergency responders would have to understand the various unique microreactor designs in order to develop response plans. The merits of vendors providing emergency response capabilities instead of state and local governments were also discussed. It was noted that such an approach would conflict with state and local responsibilities to respond to emergencies. The level of required emergency planning was discussed as something that needed to be further explored and/or clarified. The discussion also noted other sessions had recognized the importance of volume in transportation and, for example, how the difference in limiting safety factors for TRISO transportation may lead to consideration of larger volume transportation and risk-informed approaches to certification. NRC staff noted the complexity of factors addressed in the presentation on system design and safety analysis and recommended the industry provide assessments or white papers along those lines for NRC to consider as it prepares for regulating new and advanced fuels. NRC staff emphasized the benefits of industry input during NRC regulatory preparations for new fuels.

5.2.4 Takeaways

- Cross-cutting challenges for microreactor transportation include hazardous materials in-reactor designs, criticality control during transport, fuel type issues, potential compensatory measures, and emergency response training.

- Challenges associated with developing principal design criteria for transportable reactors include the review approach, curation of “licensed” activity scope for new technologies, new technology approaches to storage and transport activities, and differences in evaluation approaches.

- New fuel characteristics can inform guidance updates, including addressing containment and confinement in the absence of cladding. Closing data gaps (both fundamental and proprietary) can reduce conservatism in criticality analyses for low power density fuels.

- Adapting existing storage and transportation technology to new fuels can be an efficient approach where possible.

- Industry input is important for informing NRC regulatory activities.

6 PUBLIC Q&A SESSIONS

6.1 Public Q&A Session (Day 1 Morning)

This public question-and-answer session included discussion on information sharing and collaboration and the value of operating experience and, in particular, international experience on storage of TRISO and metal fuels. Gaining efficiencies by leveraging this existing knowledge, rather than reinventing the wheel, was suggested.

6.2 Public Q&A Session (Day 1 Afternoon)

This public question-and-answer session included a comment about the possible need for volume reduction in TRISO fuel management and potential challenges in reprocessing TRISO fuels. The potential for corrosion of FSV canisters was raised. An additional question was raised about the potential for nitric acid to form at the surface of storage canisters based on radiolysis of air. NRC staff noted canisters are stainless steel with an aluminum coating to resist corrosion, and that NRC-sponsored analyses have previously considered nitric acid corrosion.

6.3 Public Q&A Session (Day 2 Afternoon)

This public question-and-answer session included questions and comments relating to the definition of damaged fuel, the potential for corrosion of vitrified waste canisters, and some disposal-related topics that were beyond the scope of the workshop. The discussions on the definition of damaged fuel stem from a question about a prior phenomena identification and ranking table (PIRT) applicable to damaged fuel and whether a similar PIRT is needed for TRISO and metal fuels. NRC staff were unaware of similar efforts for advanced fuels but noted that carrying forward the momentum from the recent EPRI TRISO PIRT to other advanced fuels would likely involve consideration of such issues. NRC staff indicated that NRC may have follow-up actions relating to the LWR PIRT for damaged fuel (that is, staff noted there was a possibility that regulatory considerations relating to LWR damaged fuel could change as a result of the LWR PIRT for damaged fuel).

A member of the public recommended consideration of a study that showed corrosion of vitrified waste canisters containing borosilicate glass. One commenter with experience in the field of vitrification noted they did not see evidence of corrosion. NRC staff requested the original commenter email the study reference so they could take a look. A corrosion expert mentioned that, in general, even inert materials in contact with stainless steel can cause crevice corrosion. A commenter asked if the study represented something NRC staff was overlooking. NRC staff clarified the significance of any study would depend on the circumstances but there may be meaningful information to study and that NRC would consider the phenomenon as needed. NRC staff also stressed the importance of context in evaluating individual studies for relevance and that NRC focuses on issues important to safety. A member of the public suggested the use of stainless steel for canisters should be discouraged in favor of alternatives.

6.4 Public Q&A Session (Day 3 Afternoon)

This public question-and-answer session included questions and comments about gas generation in TRISO particles, neutron shielding for microreactors, corrosion of stainless-steel canisters, disposal of SNF, and the safety significance of superficial scratches on SNF storage canisters. Participants described how research on gas generation in TRISO particles indicated that the volume in the particle is sufficient to maintain low pressures over long time periods. Regarding neutrons from microreactors, potential safety issues have been evaluated and considerations for additional shielding during operations and potentially during transportation have been considered to address neutron radiation hazards. Regarding corrosion of canisters, the NRC staff clarified that no corrosion issues have been identified with large, bolted, storage and transportation casks but continues to consider corrosion during its safety reviews. Regarding scratches in welded stainless steel storage canisters, the NRC staff also summarized the known information of scratches on canisters in service that were evaluated and noted that scratches as a result of contact with carbon steel are not desirable but that small scratches were not consequential for aging management. The NRC staff also noted the presence of carbon steel contamination on a storage canister typically would not directly cause corrosion on stainless steel surface but the presence of iron contamination could lead to the formation of an environment where localized corrosion and possibly stress corrosion cracking of the stainless steel could occur; however, because canisters were treated in the instance under discussion, this condition would not be a problem.

7 CLOSING SESSION

7.1 Session 11 Closing

Jason Piotter (NRC), Laura McManniman (EPRI), Jorge Narvaez (DOE)

During the closing session, NRC, EPRI, and DOE staff summarized the information shared in the workshop.

NRC staff closed the workshop by reiterating the workshop was intended to provide an interactive venue where regulators, industry, and researchers discussed technical and regulatory considerations for managing TRISO and metal fuel following its use in advanced nuclear reactors. The NRC’s goal in hosting this workshop was described as furthering the NRC’s regulatory readiness to license and oversee advanced technologies. NRC staff expressed hope that the workshop has been beneficial to all participants, aiding NRC, DOE, and industry.

8 LIST OF PRESENTERS AND DISCUSSION PARTICIPANTS

Name	Affiliation	Presenter
Stuart Arm	PNNL	x
J. David Arregui-Mena	ORNL	x
Sven Bader	Orano Federal Services LLC	
Andrew Barto	NRC	x
Joseph Bass	NRC	
Andrew Bielen	NRC	x
Kathryn Brock	NRC	x
Lu Cai	INL	x
Paul Cantonwine	ORNL	
Travis Chapman	BWX Technologies, Inc.	x
Steve Chengelis	EPRI	
Justin Clarity	PNNL	
Blaise Collin	NRC	
James Corson	NRC	x
Matt Featherston	X-energy LLC	
Umapathy Ganjigatte	1). Indian Institute of Technology Delhi, New Delhi, Indian 2). Inter University Accelerator Center, New Delhi, India	x
Steven D. Herrmann	INL	x
Pavlo Ivanusa	PNNL	
Wen Jiang	NCSU	x
Taek K. Kim	ANL	x
Bret Leslie	NWTRB	x
Eddie Lopez Honorato	ORNL	x
Steven Maheras	PNNL	x
Tanner Mauseth	INL	x
Rod McCullum	NEI	x
Laura McManniman	EPRI	x
Dan Moneghan	EPRI	
Steven Muller	NRC	
Paul Murray	DOE	x
Prakash Narayanan	ORANO TN	x
Jorge Narvaez	DOE	x
Steven Nesbit	LMNT Consulting	
James J. Noel	The University of Western Ontario	x
Gordon Petersen	INL	x
Jason Piotter	NRC	x
Jeffrey Powers	BWXT	
Laura Price	SNL	x
Cinthya Roman	NRC	x

Name	Affiliation	Presenter
Ralf Schneider-Eickhoff	BGZ Gesellschaft für Zwischenlagerung mbH	x
Steve Sisley	NAC International	x
Jesse Sloane	Deep Isolation	x
Rebecca E. Smith	INL	x
Benjamin Spencer	INL	
John Stempien	INL	x
Craig Stover	EPRI	x
Maik Stuke	BGZ Gesellschaft für Zwischenlagerung mbH	x
James B. Tompkins	X-energy, LLC	
Stephen Vaughn	X-energy, LLC	
Haiming Wen	Missouri S&T	x
Walter Williams	NRC	x
Jonathan Wright	Ultra Safe Nuclear Corporation	
Tiankai Yao	INL	x

APPENDIX A

WORKSHOP PROGRAM

U.S.NRC United States Nuclear Regulatory Commission *Protecting People and the Environment* | U.S. DEPARTMENT OF ENERGY | Office of NUCLEAR ENERGY | EPRI | CNWRA

2024 Workshop on

STORAGE AND TRANSPORTATION OF TRISO AND METAL SPENT NUCLEAR FUELS

Program Schedule

December 3-5, 2024

The Nuclear Regulatory Commission (NRC) is holding the 2024 Workshop on Storage and Transportation of TRISO and Metal Spent Nuclear Fuels as a virtual event on December 3rd to 5th, 2024. The workshop is being held in coordination with the DOE Office of Nuclear Energy and EPRI, with assistance from the Center for Nuclear Waste Regulatory Analyses.

The workshop will be focused on research on technical and regulatory considerations for new fuels spent fuel management.

A meeting link will be sent to registered attendees and presenters approximately one week before the event.

AGENDA

Tues, Dec. 3rd, 10:00 AM to 5:00 PM

Time	Topic	Speaker
10:00 – 11:20 am	Intro & Plenary Session	NRC, DOE, EPRI
11:20 – 11:30 am	Q&A	Public
11:30 – 12:30	TRISO SNF Structural Integrity	NRC, DOE, EPRI, and Industry
12:30 – 1:30 pm	Lunch Break	
1:30 – 2:20 pm	TRISO SNF Materials Performance Part 1	NRC, DOE, EPRI, and Industry
2:20 – 3:00 pm	TRISO SNF Materials Performance Part 2	NRC, DOE, EPRI, and Industry
3:20 – 3:20 pm	Break	
3:20 – 4:45 pm	TRISO SNF Nuclear Physics / Neutronics	NRC, DOE, EPRI, and Industry
4:45 – 5:00 pm	Q&A	Public

Wed, Dec. 4th, 10:00 AM to 5:00 PM

Time	Topic	Speaker
10:00 – 11:00 am	TRISO SNF Materials Performance Part 3	NRC, DOE, EPRI, and Industry
11:00 – 12:30	Metal SNF Nuclear Physics / Neutronics	NRC, DOE, EPRI, and Industry
12:30 – 1:30 pm	Lunch Break	
1:30 – 4:30 pm	Metal SNF Materials Performance and Structural Integrity (with one break)	NRC, DOE, EPRI, and Industry
4:30 – 5:00 pm	Q&A	Public

Thurs, Dec. 5th, 10:00 AM to 4:30 PM

Time	Topic	Speaker
10:00 – 12:30	Additional Topics Part 1: Experience and Projections	NRC, DOE, EPRI, and Industry
12:30 – 1:30	Lunch Break	
1:30 – 3:00 pm	Additional Topics Part 2: Regulations, Guidance, Crosscutting Topics	NRC, DOE, EPRI, and Industry
3:00 – 3:30 pm	Break	
3:30 – 4:00 pm	Q&A	Public
4:00 – 4:30 pm	Closing Remarks	NRC

Session 1: Plenary Session – Tuesday, Dec 3, 10:00 AM to 11:30 AM Eastern Standard Time
Public Meeting Statement: Andrea Johnson
Moderator: Raj Iyengar
Scribes: Ashley Smith, Patrick LaPlante

Presenter	Affiliation	Title	Present (minutes)	Discuss (minutes)
Kathy Brock	*U.S. Nuclear Regulatory Commission*	Opening Remarks	5	N/A
Cinthya Roman	*U.S. Nuclear Regulatory Commission*	NRC Vision	15	
Paul Murray	*U.S. Department of Energy*	DOE Vision	15	
Craig Stover	*Electric Power Research Institute*	EPRI Perspective	15	
Gordon Petersen	*Idaho National Laboratory*	TRISO Spent Nuclear Fuel PIRT – Storage and Transportation	15	
Jason Piotter	*U.S. Nuclear Regulatory Commission*	Advancing the Vision of NextGen Fuels	15	
		Q&A	10	

Session 2: TRISO SNF Structural Integrity – Tuesday, Dec 3, 11:30 AM to 12:30 PM Eastern Standard Time.
Subtopics: *Matrix Fracture, Non-fuel Block Fracture, TRISO Particle Layer Fracture*
Moderator: Tom Boyce
Scribes: Joseph Bass, Curtis Lurvey, Hector Mendoza

Presenter	Affiliation	Title	Present (minutes)	Discuss (minutes)
John Stempien	*Idaho National Laboratory*	Matrix Structural Integrity – desirable and undesirable features of matrix materials for TRISO-based fuels	5	5
Eddie Lopez Honorato	*Oak Ridge National Laboratory*	Implications of new coated particle fuels with new architectures for an expanded service envelope	5	5
Tanner Mauseth	*Idaho National Laboratory*	Fracture Behavior Considerations for the TRISO Particle Matrix	5	5
Wen Jiang	*North Carolina State University*	Modeling of TRISO and Matrix Fracture	5	5
John Stempien	*Idaho National Laboratory*	TRISO Particle Fracture – importance of strong matrix and careful handling	5	5
		Open Discussion	10	

Additional Discussion Participants: Blaise Collin (Ultra Safe Nuclear), Jonathan Wright (Ultra Safe Nuclear), Benjamin Spencer (Idaho National Laboratory), and Public Attendees.

Lunch Break – Tuesday, Dec 3, 12:30 PM to 1:30 PM Eastern Standard Time

Session 3: TRISO SNF Materials Performance Part 1 – Tuesday, Dec 3, 1:30 PM to 2:20 PM Eastern Standard Time
Subtopics: *SiC Corrosion, PyC Creep and SIC Fracture*
Moderator: Tom Boyce
Scribes: Ashley Smith, Joseph Bass

Presenter	Affiliation	Title	Present (minutes)	Discuss (minutes)
Tanner Mauseth	*Idaho National Laboratory*	Micro-Tensile Properties of Irradiated AGR-2 TRISO Fuel Pyrolytic Carbon (PyC) and Silicon Carbide (SiC) Coatings	5	5
Haiming Wen	*Missouri University of Science and Technology*	Oxidation Behavior of the SiC Coating of TRISO Fuel Particles in Air	5	5
John Stempien	*Idaho National Laboratory*	PyC Creep and SiC Fracture – out-of-pile PyC creep should be zero as should SiC fracture	5	5
Wen Jiang	*North Carolina State University*	Time-Dependent Weibull Failure Analysis of TRISO Fuel	5	5
		Open Discussion	10	

Additional Discussion Participants: Benjamin Spencer (Idaho National Laboratory), Steven Muller (U.S. Nuclear Regulatory Commission), Blaise Collin (Ultra Safe Nuclear), Jonathan Wright (Ultra Safe Nuclear), Jeffery Powers (BWX Technologies, Inc.), Eddie Lopez Honorato (Oak Ridge National Laboratory, and Public Attendees.

Session 4: TRISO SNF Materials Performance Part 2 – Tuesday, Dec 3, 2:20 PM to 3:00 PM Eastern Standard Time
Subtopics: *Particle, Block, and Matrix Oxidation*
Moderator: Wendy Reed
Scribes: Aditya Savara, Patrick LaPlante

Presenter	Affiliation	Title	Present (minutes)	Discuss (minutes)
Rebecca E. Smith	*Idaho National Laboratory*	Safety Considerations for Irradiated Graphite	5	5
Lu Cai	*Idaho National Laboratory*	Determining the Oxidation Behavior of Matrix Graphite	5	5
J. David Arregui-Mena	*Oak Ridge National Laboratory*	Oxidation of graphitic components under accident conditions	5	5
		Open Discussion	10	

Additional Discussion Participants: Blaise Collin (Ultra Safe Nuclear), Jonathan Wright (Ultra Safe Nuclear), Benjamin Spencer (Idaho National Laboratory), Joseph Bass (U.S. Nuclear Regulatory Commission), Jeffery Powers (BWX Technologies, Inc.), Eddie Lopez Honorato (Oak Ridge National Laboratory), and Public Attendees.

Break – Tuesday, Dec 3, 3:00 PM to 3:20 PM Eastern Standard Time

Session 5: TRISO SNF Nuclear Physics / Neutronics – Tuesday, Dec 3, 3:20 PM to 4:45 PM Eastern Standard Time
Subtopics: *Decay Heat, Neutron Multiplication and Criticality, Shielding and Radiation Protection*
Moderator: Hossein Esmaili
Scribes: Trey Hathaway, Ellie Cohn

Presenter	Affiliation	Title	Present (minutes)	Discuss (minutes)
Andrew Bielen	*U.S. Nuclear Regulatory Commission*	NRC's simulation capabilities supporting criticality, reactor physics, decay heat, and shielding for TRISO-particle fueled non-LWRs	20	5
Laura Price	*Sandia National Laboratories*	TRISO and Metal Spent Nuclear Fuels Decay Heat	5	5
Gordon Petersen	*Idaho National Laboratory*	Modeling Capabilities for TRISO and Metallic SNF	5	5
Andrew Barto	*U.S. Nuclear Regulatory Commission*	Licensing Experience with TRISO Spent Fuel – A Historical Perspective: Fort St. Vrain Independent Spent Fuel Storage Installation (ISFSI)	10	5
		Open Discussion	25	

Additional Discussion Participants: Taek K. Kim (Argonne National Laboratory), Pavlo Ivanusa (Pacific Northwest National Laboratory), Blaise Collin (Ultra Safe Nuclear), Jonathan Wright (Ultra Safe Nuclear), Justin Clarity (Pacific Northwest National Laboratory), Jeffery Powers (BWX Technologies, Inc.), Sven Bader (Orano Federal Services LLC), Steven Nesbit (LMNT Consulting), Eddie Lopez Honorato (Oak Ridge National Laboratory), and Public Attendees.

Q & A – Tuesday, Dec 3, 4:45 PM to 5:00 PM Eastern Standard Time
Moderator: Hossein Esmaili
Scribes: Trey Hathaway, Ellie Cohn

Public attendees may participate in session discussions, as well as dedicated Q&A periods.

Session 6: TRISO SNF Materials Performance Part 3 – Wednesday, Dec 4, 10:00 AM to 11:00 AM Eastern Standard Time
Subtopics: *Gas Pressurization (Including from Alpha Decay), Fission Products Leaching, Fission Products Diffusion, SiC Corrosion*
Public Meeting Statement: Andrea Johnson
Moderator: Tekia Govan
Scribes: Ashley Smith, Patrick LaPlante

Presenter	Affiliation	Title	Present (minutes)	Discuss (minutes)
James Corson	*U.S. Nuclear Regulatory Commission*	US NRC Modeling for TRISO Material Performance	10	10
Umapathy R Ganjigatte	*Indian Institute of Technology Delhi, New Delhi, India & Inter University Accelerator Center, New Delhi, India*	Effects of Rare Earth Doping and High-Energy Irradiation in Silicon Carbide for Advanced Nuclear Applications	5	5
		Open Discussion	30	

Additional Discussion Participants: Rebecca E. Smith (Idaho National Laboratory), Lu Cai (Idaho National Laboratory), J. David Arregui-Mena (Oak Ridge National Laboratory), Blaise Collin (Ultra Safe Nuclear), Jonathan Wright (Ultra Safe Nuclear), Benjamin Spencer (Idaho National Laboratory), Joseph Bass (U.S. Nuclear Regulatory Commission), Jeffery Powers (BWX Technologies, Inc.), Wen Jiang (North Carolina State University), Eddie Lopez Honorato (Oak Ridge National Laboratory), and Public Attendees.

Session 7: Metal SNF Nuclear Physics / Neutronics – Wednesday, Dec 4, 11:00 PM to 12:30 AM Eastern Standard Time
Subtopics: *Decay Heat, Neutron Multiplication and Criticality, Shielding and Radiation Protection*
Moderator: Jason Piotter
Scribes: Trey Hathaway, Ellie Cohn

Presenter	Affiliation	Title	Present (minutes)	Discuss (minutes)
Andrew Barto	*U.S. Nuclear Regulatory Commission*	10 CFR Part 71 - Certification of Transportation Packages for Metal Fuel	10	5
Andrew Bielen	*U.S. Nuclear Regulatory Commission*	NRC's simulation capabilities supporting criticality, reactor physics, decay heat, and shielding for metallic fueled non-LWRs	20	5
Gordon Petersen	*Idaho National Laboratory*	Modeling Capabilities for TRISO and Metallic SNF	5	5
Laura Price	*Sandia National Laboratories*	TRISO and Metal Spent Nuclear Fuels Decay Heat	5	5
		Open Discussion	30	

Additional Discussion Participants: Taek K. Kim (Argonne National Laboratory), Justin Clarity (Pacific Northwest National Laboratory), Sven Bader (Orano Federal Services LLC), Steven Nesbit (LMNT Consulting), and Public Attendees.

Lunch Break – Wednesday, Dec 4, 12:30 PM to 1:30 PM Eastern Standard Time

Session 8: Metal SNF Materials Performance and Structural Integrity – Wednesday, Dec 4, 1:30 PM to 4:30 PM Eastern Standard Time
Subtopics: *Corrosion, Reactions with Water and Chemical Treatments, Fission Products Leaching, Fission Products Diffusion, Fission Gas Generation and Release, Cladding Rupture Due to Pressurization, Fuel Swelling, Deformation*
Moderator: Tekia Govan
Scribes: Ashley Smith, Hector Mendoza, Aditya Savara

Presenter	Affiliation	Title	Present (minutes)	Discuss (minutes)
James Corson	*U.S. Nuclear Regulatory Commission*	U.S NRC Modeling Capabilities of Metal Fuel in FAST	10	10
Tiankai Yao	*Idaho National Laboratory*	Fission Product Diffusion	5	5
Tiankai Yao	*Idaho National Laboratory*	Corrosion of Cladding Materials	5	5
Tiankai Yao	*Idaho National Laboratory*	Interactions Between Metallic Fuel and Water	5	5
Walter Williams	*U.S. Nuclear Regulatory Commission*	Fission Product Induced Metal Fuel Swelling	5	5
Walter Williams	*U.S. Nuclear Regulatory Commission*	Assessment on Metal Spent Nuclear Fuel Swelling Effects on Structural Integrity	5	5
		Open Discussion	**20**	
		Break	**20**	**N/A**
Stuart Arm	*Pacific Northwest National Laboratory*	Potential Treatment Options for Sodium-Bonded Metal Fuel	10	15
Steven D. Herrmann	*Idaho National Laboratory*	Removal and Deactivation of Bond Sodium from Fast Reactor Materials	5	5
Jamie Noel	*University of Western Ontario*	Materials interactions leading to enhanced dissolution or protection of fuel in a waste storage	5	5
		Open Discussion	25	

Additional Discussion Participants: Benjamin Spencer (Idaho National Laboratory), Sven Bader Orano (Federal Services LLC), and Public Attendees.

Q & A – Wednesday, Dec 4, 4:30 PM to 5:00 PM Eastern Standard Time
Moderator: Jesse Carlson
Scribes: Ashley Smith, Patrick LaPlante, Aditya Savara
Public attendees may participate in session discussions, as well as dedicated Q&A periods.

Session 9: Additional Topics, Part 1 – Thursday, Dec 5, 10:00 AM to 12:30 PM Eastern Standard Time
Subtopics: Experience and Projection
Public Meeting Statement: Andrea Johnson
Moderator: Laurel Bauer
Scribes: Wendy Reed, Andrea Johnson

Presenter	Affiliation	Title	Present (minutes)	Discuss (minutes)
Ralf Schneider-Eickhoff , Maik Stuke	*BGZ Gesellschaft für Zwischenlagerung mbH*	Dry Storage of THTR Spent Fuel in Germany	15	10
Bret Leslie	*U.S. Nuclear Waste Technical Review Board*	Management and Disposal of U.S. Department of Energy's TRISO- and Metallic-based Spent Nuclear Fuel and Preliminary Considerations for Waste Resulting from Advanced Nuclear Reactors	15	10
Taek K. Kim	*Argonne National Laboratory*	Projection of TRISO spent nuclear fuels and related issues	10	5
Jesse Sloane[1], Steve Sisley[2]	*[1] Deep Isolation, [2] NAC International*	Management of TRISO spent fuel using a Universal Canister System	15	10
		Open Discussion	60	

Additional Discussion Participants: Blaise Collin (Ultra Safe Nuclear), Jonathan Wright (Ultra Safe Nuclear), Steven Nesbit (LMNT Consulting), Andrew Barto (U.S. Nuclear Regulatory Commission), Jason Piotter (U.S. Nuclear Regulatory Commission), Paul Cantonwine (Oak Ridge National Laboratory), Sven Bader (Orano Federal Services LLC), Matt Featherston (X-Energy, LLC), Stephen Vaughn (X-Energy, LLC), Prakash Narayanan (ORANO TN), Eddie Lopez Honorato (Oak Ridge National Laboratory), Rod McCullum (Nuclear Energy Institute), Steven Maheras (Pacific Northwest National Laboratory), and Public Attendees.

Session 10: Additional Topics, Part 2 – Thursday, Dec 5, 1:30 PM to 3:00 PM Eastern Standard Time
Subtopics: *Regulations, Guidance, Crosscutting Topics*
Moderator: Jose Cuadrado
Scribes: Ashley Smith, Patrick LaPlante, Andrea Johnson

Presenter	Affiliation	Title	Present (minutes)	Discuss (minutes)
Steven Maheras	*Pacific Northwest National Laboratory*	Microreactor Transportation Emergency Planning Challenges	10	10
Travis Chapman	*BWX Technologies, Inc.*	Cross-domain Development of Principal Design Criteria for Transportable Reactors	10	10
Prakash Narayanan	*ORANO TN*	System design and safety analysis associated with storage and transportation	10	10
Rod McCullum	*Nuclear Energy Institute*	Building on Established Knowledge to Inform the Regulatory Framework for TRISO and Metal Spent Nuclear Fuels	10	10
		Open Discussion	10	

Additional Discussion Participants: Blaise Collin (Ultra Safe Nuclear), Jonathan Wright (Ultra Safe Nuclear), Steven Nesbit (LMNT Consulting), Andrew Barto (U.S. Nuclear Regulatory Commission), Jason Piotter (U.S. Nuclear Regulatory Commission), Paul Cantonwine (Oak Ridge National Laboratory), Sven Bader (Orano Federal Services LLC), Matt Featherston (X-Energy, LLC), Stephen Vaughn (X-Energy, LLC), Eddie Lopez Honorato (Oak Ridge National Laboratory), and Public Attendees.

Break – Thursday, Dec 5, 3:00 PM to 3:30 PM Eastern Standard Time

Q & A – Thursday, Dec 5, 3:30 PM to 4:00 PM Eastern Standard Time
Moderator: Jesse Carlson
Scribes: Ashley Smith, Patrick LaPlante, Andrea Johnson
Public attendees may participate in session discussions, as well as dedicated Q&A periods.

Session 11: Closing Session – Thursday, Dec 5, 4:00 PM to 4:30 PM Eastern Standard Time
Subtopics: Closing Remarks

Presenter	Affiliation	Title	Present (minutes)	Discuss (minutes)
Jason Piotter [1], Laura McManniman [2], Jorge Narvaez[3]	*[1] U.S. Nuclear Regulatory Commission* *[2] Electric Power Research Institute* *[3] U.S. Department of Energy*	Closing Remarks	15	15

APPENDIX B

WORKSHOP ABSTRACTS

2024 Workshop on

STORAGE AND TRANSPORTATION OF TRISO AND METAL SPENT NUCLEAR FUELS

Abstracts Booklet

December 3-5, 2024

Workshop Website

The Nuclear Regulatory Commission (NRC) is holding the 2024 Workshop on Storage and Transportation of TRISO and Metal Spent Nuclear Fuels as a virtual event on December 3rd to 5th, 2024. The workshop is being held in coordination with the DOE Office of Nuclear Energy and EPRI, with assistance from the Center for Nuclear Waste Regulatory Analyses.

The workshop will be focused on research on technical and regulatory considerations for new fuels spent fuel management.

The program schedule can be downloaded at the workshop website.

Session Name: Session 2

Speaker Name: John Stempien

Title: Matrix Structural Integrity – desirable and undesirable features of matrix materials for TRISO-based fuels

Abstract: Among other functions, the matrix in TRISO-based fuel forms (e.g., cylindrical compacts and spherical pebbles) serves to protect the TRISO particles from mechanical damage from external events and it will retain fission products accumulated in it during irradiation. Some as-fabricated morphologies have been identified as undesirable in part based on post-irradiation examinations where accidental handling damage is believed to have occurred in some cases. Certain engineered features can provide additional protection, though this may not be necessary. The discrete micro-containment each TRISO particle represents is a benefit of this fuel form in the event a fuel element was ever fractured.

Speaker Name: Eddie Lopez Honorato

Title: Implications of new coated particle fuels with new architectures for an expanded service envelope

Abstract: The most mature coated particle fuel design is the Tristructural-isotropic (TRISO) coated particle nuclear fuel developed for high temperature gas-cooled reactors, which is composed of a uranium oxide or multiphase uranium oxide/carbide (UO2/UC/UC2) kernel coated with three layers of pyrolytic carbon (PyC) and one layer of SiC. Coated particle fuels have been proposed for microreactor designs for terrestrial use and space exploration with expanded service envelopes. This expanded service envelope in many cases will require the fabrication of coated particles with new combinations of kernels (composition, shape, and size) and coatings (number, composition, and thicknesses) to meet operational goals. A discussion of the implications of the new architectures on fabrication and resultant microstructure and physical properties of the spent fuel will be discussed.

Speaker Name: Tanner Mauseth

Title: Fracture Behavior Considerations for the TRISO Particle Matrix

Abstract: To assess whether matrix fracture would result in an unacceptable loss of containment or confinement in TRISO fuel particles, it is crucial to evaluate the micro-tensile strength, fracture toughness, and irradiation effects on matrix materials. Current data must be comprehensive and validated for modeling fractures under various conditions. Relevant material properties surrounding matrix fracture will be discussed during the presentation.

Speaker Name: Wen Jiang

Title: Modeling of TRISO and Matrix Fracture

Abstract: Potential TRISO failure mechanisms under normal and off-normal conditions include overpressure failure, irradiation-induced IPyC cracking, debonding between coating layers and buffer tearing. A large majority of failure mode analysis are focused on the TRISO particle itself. However, there is a lack of studies investigating the mechanical interaction between the matrix and the embedded TRISO particles. The deformation of TRISO particles will cause stress concentration on the matrix especially at locations between particles. On the other hand, the cracks initiated in the matrix may have a deleterious impact on the particle coating layer integrity. In addition, the behavior of particle-matrix interactions during long-term needs to be studied.

Speaker Name: John Stempien

Title: TRISO Particle Fracture – importance of strong matrix and careful handling

Abstract: Experiments have shown that in-pile and post-inert-accident-testing TRISO failures have low rates of occurrence. Irradiated TRISO fuel elements routinely withstand handling in hot cell environments via remote equipment, harsh acceleration/deceleration via pneumatic rabbit transfers, and air and land-based transportation. Experience has shown that while accidental damage during post-irradiation handling is possible, it is generally confined to small numbers of TRISO particles in elements composed of friable matrix. Other than generally limited damage from post-irradiation handling (that may be avoidable), additional TRISO SNF fracture or failure from other means (e.g., chemical or mechanical interactions within the fuel or other external mechanical phenomena) are not anticipated.

Session Name: Session 3

Speaker Name Tanner Mauseth

Title: Micro-Tensile Properties of Irradiated AGR-2 TRISO Fuel Pyrolytic Carbon (PyC) and Silicon Carbide (SiC) Coatings

Abstract: Tristructural isotropic (TRISO) coated nuclear fuel particles are emerging as a versatile option for new reactor designs, with the silicon carbide (SiC) layer crucial for retaining fission products. However, the mechanical properties of TRISO coating layers, particularly after irradiation, are not fully understood due to their small size and high radioactivity. Recent in situ micro-tensile testing of various TRISO layers aims to better understand the SiC layer's failure mechanisms, advancing TRISO fuel qualification. These micro-tensile results will be presented.

Speaker Name: Haiming Wen

Title: Oxidation Behavior of the SiC Coating of TRISO Fuel Particles in Air

Abstract: While high-temperature gas reactors use helium as a coolant, in some accident scenarios significant amounts of air can be introduced into the coolant and reactor core. It is important to understand the oxidation behavior and mechanisms of TRISO particles (especially the SiC coating layer) under these conditions. The oxidation mechanisms in relation to the oxidation conditions and microstructures of the SiC will be presented. Passive oxidation occurred at high oxygen partial pressure. At low partial pressure of oxygen, the oxidation mechanism was determined to be a mixture of passive and active oxidation; nanocrystalline grain size promotes activation oxidation, followed by redeposition of SiO_2.

Speaker Name: John Stempien

Title: PyC Creep and SiC Fracture – out-of-pile PyC creep should be zero as should SiC fracture

Abstract: Irradiation-induced PyC creep may occur, but this will terminate once the fuel has been removed from the reactor. SiC mechanical fracture from PyC creep has not been observed in irradiated US UCO TRISO fuels, rather a multi-step process culminating in chemical attack of the SiC at high temperatures has occasionally been observed. Thus, it is expected that no additional PyC or SiC degradation will occur in TRISO SNF under normal circumstances in storage and transportation.

Speaker Name: Wen Jiang

Title: Time-Dependent Weibull Failure Analysis of TRISO Fuel

Abstract: The ability of tri-structural isotropic (TRISO) fuel to contain fission products is dictated by the structural integrity of its coating layers under various conditions. Currently, a Weibull failure criterion is used in fuel performance codes to determine failure for the IPyC and SiC layers. However, this model only considers the instantaneous stress state and does not account for time-dependent effect, stress history and environment conditions. This becomes problematic for long-term failure evaluations, such as during fuel storage, where the stress levels may be below the strength threshold, and subcritical crack growth could dominate. We will discuss strategies to enable more robust and accurate failure analysis for TRISO fuel coating layers.

Session Name: Session 4

Speaker Name: Rebecca E. Smith

Title: Safety Considerations for Irradiated Graphite

Abstract: While this may sound contradictory, graphite oxidizes but it does not burn. This attribute allows the industrial use of (unirradiated) graphite as a fire extinguishing agent. Oxidation of

graphite can degrade the material properties of the remaining material. And irradiated graphite may oxidize at double or triple the rate of the same grade of unirradiated graphite. Observations on graphite oxidation will be presented to help define relevant safety considerations for the transportation and storage of TRISO spent nuclear fuel.

Speaker Name: Lu Cai

Title: Determining the Oxidation Behavior of Matrix Graphite

Abstract: This work presents the oxidation behavior of matrix graphite (or fuel matrix) in air. Matrix graphite, graphite powder/flakes bonded by a small amount of non-graphitic carbon, surrounds coated fuel particles in order to form cylindrical fuel compacts (in prismatic core designs) or spheres (in pebble-bed reactor designs). The oxidation of matrix graphite needs to be addressed either as chronic oxidation or as acute oxidation for the fuel integrity evaluation and safety analysis in the extremely unlikely case of an air ingress accident. This work shows matrix graphite materials may experience preferential oxidation of the non-graphitic carbon. We will also discuss about the irradiation effects on the oxidation behavior.

Speaker Name: J. David Arregui-Mena

Title: Oxidation of graphitic components under accident conditions

Abstract: Various graphitic materials form part of the new generation of nuclear reactors in the US. Under the accidental ingress of air or moisture matrix graphite and nuclear graphite components can undergo acute oxidation. Oxidation under accident conditions would affect the outer layer of graphitic components changing their microstructure. This research covers novel experimental procedures to simulate the conditions during the accidental ingress of air and characterizes the evolution of graphitic components under acute oxidation. Results of in situ experiments will be presented to understand the evolution of various graphitic materials under accidental ingress of air into the core.

Session Name: Session 5

Speaker Name: Andrew Bielen

Title: NRC's simulation capabilities supporting criticality, reactor physics, decay heat, and shielding for TRISO-particle fueled non-LWRs

Abstract: Recent efforts have been made to develop and assess new simulation capabilities in NRC's neutronics code SCALE & NRC's accident progression code MELCOR for modeling TRISO-particle fuel designs in non-LWRs (i.e., high temperature gas-cooled and molten salt-cooled reactors), as outlined in NRC's Volume 3 & 5 strategies. An overview of these newly developed workflows and newly added phenomenological models in SCALE & MELCOR will be highlighted, focusing on criticality, reactor physics, decay heat and shielding type analyses. Addressed modeling gaps, new phenomenological model development, and demonstration of these new capabilities will be given. Perspectives on data availability and where additional validation data would be beneficial will be highlighted.

Speaker Name: Laura Price

Title: TRISO and Metal Spent Nuclear Fuels Decay Heat

Abstract: The decay heat generated by TRISO SNF and metallic SNF per volume of SNF is a function of its burnup and the mass of uranium per volume of fuel (pebble, prismatic block, assembly). TRISO SNF is much cooler than both typical LWR SNF and metal SNF on a basis of volume.

Speaker Name: Gordon Petersen

Title: Modeling Capabilities for TRISO and Metallic SNF

Abstract: Spent Nuclear Fuel (SNF) generated by advanced reactors is expected to have higher burnups and different characteristics than traditional light water reactor SNF. Metallic and TRISO SNF can be modeled using existing nuclear codes to assess radiation protection and maintaining subcriticality

Speaker Name: Andrew Barto

Title: Licensing Experience with TRISO Spent Fuel – A Historical Perspective: Fort St. Vrain Independent Spent Fuel Storage Installation (ISFSI)

Abstract: Several new Non-Light Water Reactor concepts involve the use of TRISO fuel particles, including the HTGR. The Fort St. Vrain (FSV) prismatic fuel HTGR operated from 1979 to 1989, using high enriched uranium TRISO compacts in a hexagonal graphite fuel block. NRC licensed an ISFSI for the FSV reactor site. This presentation will discuss licensing experience related to the FSV ISFSI, and key differences expected between this facility and future TRISO spent fuel facilities.

Session Name: Session 6

Speaker Name: James Corson

Title: US NRC Modeling for TRISO Material Performance

Abstract: Recent efforts have been made to develop and assess new simulation capabilities in NRC's fuel performance code, FAST (Fuel Analysis under Steady-state and Transients) for modeling TRISO-particle fuels, as outlined in NRC's Volume 2 strategy. An overview of FAST's modeling capabilities, for modeling will be highlighted, including newly added material property models, new fuel performance models, and future assessments. Upcoming efforts in validation activities will be discussed as well. Perspectives on data availability and where additional validation data would be beneficial will be highlighted.

Speaker Name: Umapathy R Ganjigatte

Title: Effects of Rare Earth Doping and High-Energy Irradiation in Silicon Carbide for Advanced Nuclear Applications

Abstract: Silicon carbide (SiC) is the outermost protective layer in TRISO (Tri-structural Isotropic) fuel particles, acting as a critical barrier to fission product release in advanced nuclear reactors. Optimizing SiC for Small Modular Reactor (SMR) applications and improving accident-tolerant fuel designs is vital. This study explores the effects of rare earth doping and high-energy irradiation on SiC's performance under nuclear conditions. By doping SiC with rare earth elements and subjecting it to high-energy MeV Ar^+ irradiation, the study aims to improve Kr/Xe gas retention, reduce thermal swelling, and enhance long-term stability. Advanced characterization techniques such as RBS, FESEM, XRD, and TEM are used for pre- and post-irradiation analysis. Additionally, a protective interface layer of Al(x)ReO(x-1) is proposed to further enhance SiC's durability. These insights are crucial for optimizing SiC in nuclear fuel designs and extending material lifespan in advanced reactors.

Session Name: Session 7

Speaker Name: Andrew Barto

Title: 10 CFR Part 71 - Certification of Transportation Packages for Metal Fuel

Abstract: NRC has issued many Certificates of Compliance for packages to transport unirradiated uranium and plutonium metal. This presentation will discuss licensing experience with certification of such packages, and anticipated issues related to eventual certification of irradiated SFR fuel transportation packages.

Speaker Name: Andrew Bielen

Title: NRC's simulation capabilities supporting criticality, reactor physics, decay heat, and shielding for metallic fueled non-LWRs

Abstract: Recent efforts have been made to develop and assess new simulation capabilities in NRC's neutronics code SCALE & NRC's accident progression code MELCOR for modeling metallic fuels in non-LWRs (i.e., sodium fast reactors), as outlined in NRC's Volume 3 & 5 strategies. An overview of these newly developed workflows and newly added phenomenological models in SCALE & MELCOR will be highlighted, focusing on criticality, reactor physics, decay heat and shielding type analyses. Addressed modeling gaps, new phenomenological model development, and demonstration of these new capabilities will be given. Perspectives on data availability and where additional validation data would be beneficial will be highlighted.

Speaker Name: Gordon Petersen

Title: Modeling Capabilities for TRISO and Metallic SNF

Abstract: Spent Nuclear Fuel (SNF) generated by advanced reactors is expected to have higher burnups and different characteristics than traditional light water reactor SNF. Metallic and TRISO SNF can be modeled using existing nuclear codes to assess radiation protection and maintaining subcriticality

Speaker Name: Laura Price

Title: TRISO and Metal Spent Nuclear Fuels Decay Heat

Abstract: The decay heat generated by TRISO SNF and metallic SNF per volume of SNF is a function of its burnup and the mass of uranium per volume of fuel (pebble, prismatic block, assembly). TRISO SNF is much cooler than both typical LWR SNF and metal SNF on a basis of volume.

Session Name: Session 8

Speaker Name: James Corson

Title: U.S NRC Modeling Capabilities of Metal Fuel in FAST

Abstract: Recent efforts have been made to develop and assess new simulation capabilities in NRC's fuel performance code, FAST (Fuel Analysis under Steady-state and Transients) for modeling metallic fuels, as outlined in NRC's Volume 2 strategy. An overview of FAST's modeling capabilities, for modeling metallic fuels will be highlighted. Discussion will be centered around FAST's methodology for modeling fission gas production and diffusion within metallic fuels. Perspectives on data availability and where additional validation data would be beneficial will be highlighted.

Speaker Name: Tiankai Yao

Title: Fission Product Diffusion

Abstract: The diffusion of fission products (FPs) in metallic fuel primarily involves the movement of lanthanide FPs along the temperature gradient during irradiation. These lanthanide FPs tend to accumulate on the inner surface of the cladding, where they react with HT9 cladding to form low melting point eutectic compounds. During the storage of metal spent nuclear fuel (SNF), localized fuel temperatures can increase due to the accumulation of decay heat and potential accidental exposure of sodium to air, reaching up to approximately 570°C. This presentation will focus on recent post-irradiation examination (PIE) characterization of the liquid-like movement of lanthanide FPs through connected pores and discuss its implications for the safety of metal SNF under both designed and accident conditions.

Speaker Name: Tiankai Yao

Title: Corrosion of Cladding Materials

Abstract: Metal spent nuclear fuel (SNF) for advanced reactors is likely to involve large amounts of HT9 steel, used in cladding and ducting. During the storage of metal SNF, decay heat can elevate the fuel temperature to as high as 480°C. In accident scenarios, exposure of residual sodium can further increase the fuel temperature to approximately 570°C. The understanding of corrosion mechanism and corrosion rate estimation of HT9 when exposed to sodium and steam at such high temperatures is crucial for the safe handling of metal SNF. Long-term thermal effects on the microstructure of HT9 will also directly impact the mechanical properties of the cladding and ducting. This presentation will focus on both historical and recent studies on the long-term corrosion of HT9 at elevated temperatures and discuss the resultant changes of microstructure and mechanical properties of HT9 for metal SNF.

Speaker Name: Tiankai Yao

Title: Interactions Between Metallic Fuel and Water

Abstract: In the accidental exposure conditions, the interaction between U-10Zr metallic fuel and water is of great safety concern. The reaction is highly exothermic with significant amount of heat being released. The reaction can lead to fuel damage with rapid temperature increase, fuel rapture due to volatile volumetric expansion. The accumulation of hydrogen can also be a concern. This presentation will focus on the basic understanding of metal fuel water interaction mechanism and how it can impact the safety of spent metal fuel.

Speaker Name: Walter Williams

Title: Fission Product Induced Metal Fuel Swelling

Abstract:

The reaction between uranium metal fuel and water is exothermic, generating heat during the storage and transport of metal spent nuclear fuel (SNF). When exposed to air, the fine and loose reaction products can potentially lead to pyrophoric incidents. A better understanding of this reaction can be achieved through detailed characterization of the corrosion morphology and products. This presentation will focus on previous experimental knowledge regarding the corrosion mechanism and discuss its implications for the safe handling of metal SNF.

Speaker Name: Walter Williams

Title: Assessment on Metal Spent Nuclear Fuel Swelling Effects on Structural Integrity

Abstract: Fuel swelling in metallic spent nuclear fuel (SNF) is being assessed for potential dimensional changes due to swelling, both radioactively and thermally induced, that may occur during long-term storage and transportation. This phenomenon could affect the structural integrity

of the fuel cladding, leading to a loss of containment or confinement of fission products. This presentation will explore the key factors that must be assessed to evaluate whether fuel swelling poses a significant hazard. Criteria for structural integrity, including stress limits, thermal expansion, and cladding deformation, will be discussed. Additionally, potential scenarios where swelling could result in unacceptable degradation of the fuel will be considered with open discussion and posed scenarios welcomed. The presentation will also address whether additional simulation tools or empirical data are necessary to more accurately predict fuel swelling behavior and its implications for safe storage and transport of metal SNF.

Speaker Name: Stuart Arm

Title: Potential Treatment Options for Sodium-Bonded Metal Fuel

Abstract: Spent sodium-bonded metal fuel may require treatment to mitigate the reactivity hazard from the sodium metal. Various treatment concepts are available although none have been demonstrated at the scale likely needed to support the deployment of commercial reactors.

Speaker Name: Steven D. Herrmann

Title: Removal and Deactivation of Bond Sodium from Fast Reactor Materials.

Abstract: A melt-drain-evaporate process demonstrated the removal of more than 99.9998% of bond sodium from full-length Fermi-1 blanket elements and an assembly within an inert atmosphere enclosure. The complete deactivation of the recovered bond sodium into a non-hazardous form was subsequently demonstrated in the same enclosure using a dry technique.

Speaker Name: Jamie Noel

Title: Materials interactions leading to enhanced dissolution or protection of spent fuel in long-term storage

Abstract: The oxidative dissolution (corrosion) of spent nuclear fuel in a storage container with water present could be either enhanced or slowed by contact with other materials in the container and by interactions with container corrosion products and the products of water radiolysis. Container corrosion may release hydrogen, which may function as an antioxidant for the fuel surface, yet water radiolysis will be a source of oxidants (e.g., hydroxyl radical, hydrogen peroxide). Galvanic coupling to the internal structures of the container and other potential contained components (cladding, graphite, etc.) may amplify the effects of oxidants and reducing agents within the container. The degree of fuel oxidation will depend on the competition between oxidants and reducing species determined by their relative reactivities on both the fuel surface and the surfaces of electrically coupled materials (e.g., metals, graphite). This presentation will introduce some of the possibilities.

Session Name: Session 9

Speaker Name: Ralf Schneider-Eickhoff: Maik Stuke

Title: Dry Storage of THTR Spent Fuel in Germany

Abstract: Between June 1992 and April 1995, approximately 320,000 THTR spent fuel elements from the German THTR-300 high-temperature thorium nuclear reactor in Hamm-Uentrop were transported to the interim storage facility in Ahaus. These elements were securely transported and stored in 305 CASTOR THTR/AVR dual-purpose casks. This presentation provides an overview of the spent fuel management, along with experiences gained over 30 years of storage.

Speaker Name: Bret Leslie

Title: Management and Disposal of U.S. Department of Energy's TRISO- and Metallic-based Spent Nuclear Fuel and Preliminary Considerations for Waste Resulting from Advanced Nuclear Reactors

Abstract: As a part of its ongoing review of the U.S. Department of Energy's (DOE) activities related to management and disposal of high-level radioactive waste and spent nuclear fuel (SNF) under the Nuclear Waste Policy Act, the U.S. Nuclear Waste Technical Review Board (NWTRB) has made several findings, conclusions, and recommendations that apply to storage, transportation, and disposal of SNF from advanced reactors. In 2017, the NWTRB released a report that identified the characteristics of SNF, including TRISO and metallic fuel types, that affect disposal which include heat generation, criticality, and degradation. In 2021, the NWTRB held a public meeting and reviewed the DOE's research and development activities related to advanced light water reactor spent fuels, and identified some preliminary considerations for storage, transportation, and disposal that also apply to waste from advanced Generation IV nuclear reactors. In 2023, the NWTRB held a public meeting and gained updates on DOE's advanced reactor SNF and waste stream disposition strategies, and activities related to DOE's technical assessment of the feasibility of storage, transportation, and disposal of advanced reactor SNF.

Speaker Name: Taek K. Kim

Title: Projection of TRISO spent nuclear fuels and related issues

Abstract: Various advanced reactors adopt TRISO particulate fuels in the form of pebbles or prismatic blocks because of their excellent capability to contain nearly all fission products within the particles. The demand for TRISO fuels is expected to increase to meet the demand for nuclear energy, recent emerging demands to support data centers, and special purposes such as microreactors. The projection of the TRISO fuel demand and the related issues, such as flooded criticality and cask during transportation of spent fuel, will be presented.

Speaker Name: Jesse Sloane, Steve Sisley

Title: Management of TRISO spent fuel using a Universal Canister System

Abstract: Several advanced reactor designs supported by the US Department of Energy's (DOE) Advanced Rector Demonstration Program (ARDP) utilize TRI-structural ISOtropic (TRISO) fuel. As these reactor concepts mature and move into development, consideration must be given to the back-end management of the resultant spent fuel. With support from DOE's Advanced Research Projects Agency – Energy (ARPA-E), Deep Isolation, in collaboration with NAC International, University of California, Berkeley, and Lawrence Berkeley National Laboratory, is developing a Universal Canister System (UCS). This system aims to enable the safe storage, transport, and disposal of advanced reactor waste streams, including TRISO spherical pebbles, cylindrical compacts, and full prismatic assemblies, in either conventional mined repositories or deep boreholes.

The preliminary design of the UCS is informed by structural, thermal, shielding, and criticality analyses of the most limiting storage, transport, and disposal configurations. These analyses specifically addressed the shielding and criticality aspects of a limiting cargo of TRISO spent fuel.

Fabrication of a prototype UCS canister is nearing completion, with plans underway for prototypic testing. Additionally, Deep Isolation is working with Kairos Power to validate the UCS design based on the expected characteristics of spent TRISO fuel from Kairos Power's KP-FHR reactor.

<u>Session Name: Session 10</u>

Speaker Name: Steven Maheras

Title: Microreactor Transportation Emergency Planning Challenges

Abstract: Transporting microreactors containing irradiated TRISO or metal fuel poses unique transportation emergency response planning challenges. Many challenges are because of the unique aspects of microreactor designs and because State and Tribal emergency responders along potential truck and rail routes are likely to be unfamiliar with microreactor transport. This presentation examines these potential transportation emergency response planning challenges. These challenges are organized into cross-cutting emergency response challenges and specific transportation emergency response challenges. The results of the evaluation discussed in this presentation include:

- Unique aspects of TRISO and metal fuels
- Use of hazardous materials in microreactor designs
- Revisions to the DOT Emergency Response Guidebook
- Potential compensatory measures
- External Engagement, Emergency Response Training, and Accident Recovery Plans
- State and Tribal perspectives on emergency planning challenges

Speaker Name: Travis Chapman

Title: Cross-domain Development of Principal Design Criteria for Transportable Reactors

Abstract: Both Parts 50 and 52 require applicants to develop principal design criteria (PDC) for a technology. These PDC form both functional design requirements that guide overall technology development and demonstration of safety principles by technology developers, as well as criteria and a basis by which a regulator may make a safety finding. Such design criteria and their development are well established to evaluate reactor safety while in the operational domain and have analogs in the review guidance for transportation and storage domains under Parts 71 and 72. An approach to establish design criteria that address all domains of a transportable reactor technology lifecycle and guide development of fuel system performance figures of merit will be described with examples.

Speaker Name: Prakash Narayanan

Title: System design and safety analysis associated with storage and transportation

Abstract: Several Dry Storage Systems (DSS) have been certified by the NRC for the Storage and Transportation of high burnup, high heat load LWR fuel assemblies. The design features of the dry storage systems have evolved over the past several years to extend service life, enhance performance, and accommodate newer contents. With the development of new types of fuel designs which include TRISO and Metallic fuel, it is important to incorporate the valuable design and operating experience associated with the current generation of dry storage systems. This presentation will discuss the applicability of these DSS designs for the storage and transportation of spent fuel that will be discharged from the next generation reactors designs, particularly with TRISO and Metallic fuels.

Speaker Name: Rod McCullum

Title: Building on Established Knowledge to Inform the Regulatory Framework for TRISO and Metal Spent Nuclear Fuels

Abstract: Building on Established Knowledge to Inform the Regulatory Framework for TRISO and Metal Spent Nuclear Fuels

Considerable scientific and technical work has been completed in the last several years to prepare for the management of advanced reactor used fuels. This work has been conducted by advanced reactor developers seeking to minimize business risks going forward as well as under the auspices of a number of coordinated national and international programs. The result of these efforts is a high degree of confidence that these fuels can be managed under NRC's existing regulations. This presentation will examine what has been learned through the efforts of the developers as well as DOE's BEMAR project, ARPA-E's ONWARDS and UPWARDS project, IAEA's COGS Cooperative Research Project, and NEA's project WISARD. The presenter will recommend that this knowledge be applied to guide efficient process going forward in the spirit of the recently enacted ADVANCE Act.

APPENDIX C

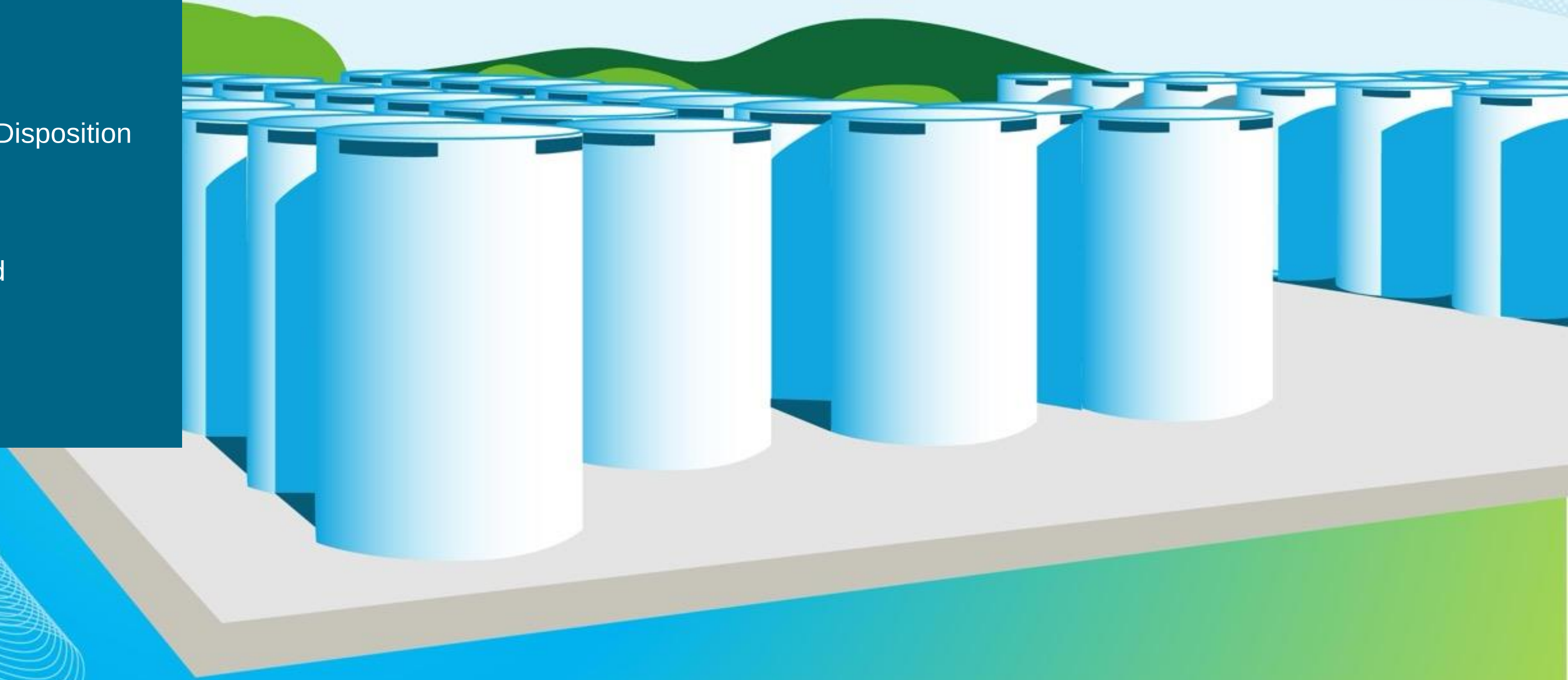
WORKSHOP PRESENTATIONS

Advanced Reactor Spent Nuclear Fuel (SNF) Management

Mr. Paul Murray
Deputy Assistant Secretary, Spent Fuel & High-Level Waste Disposition
U.S. Department of Energy (DOE)

2024 Workshop on Storage and Transportation of TRISO and Metal Spent Nuclear Fuels

December 03, 2024



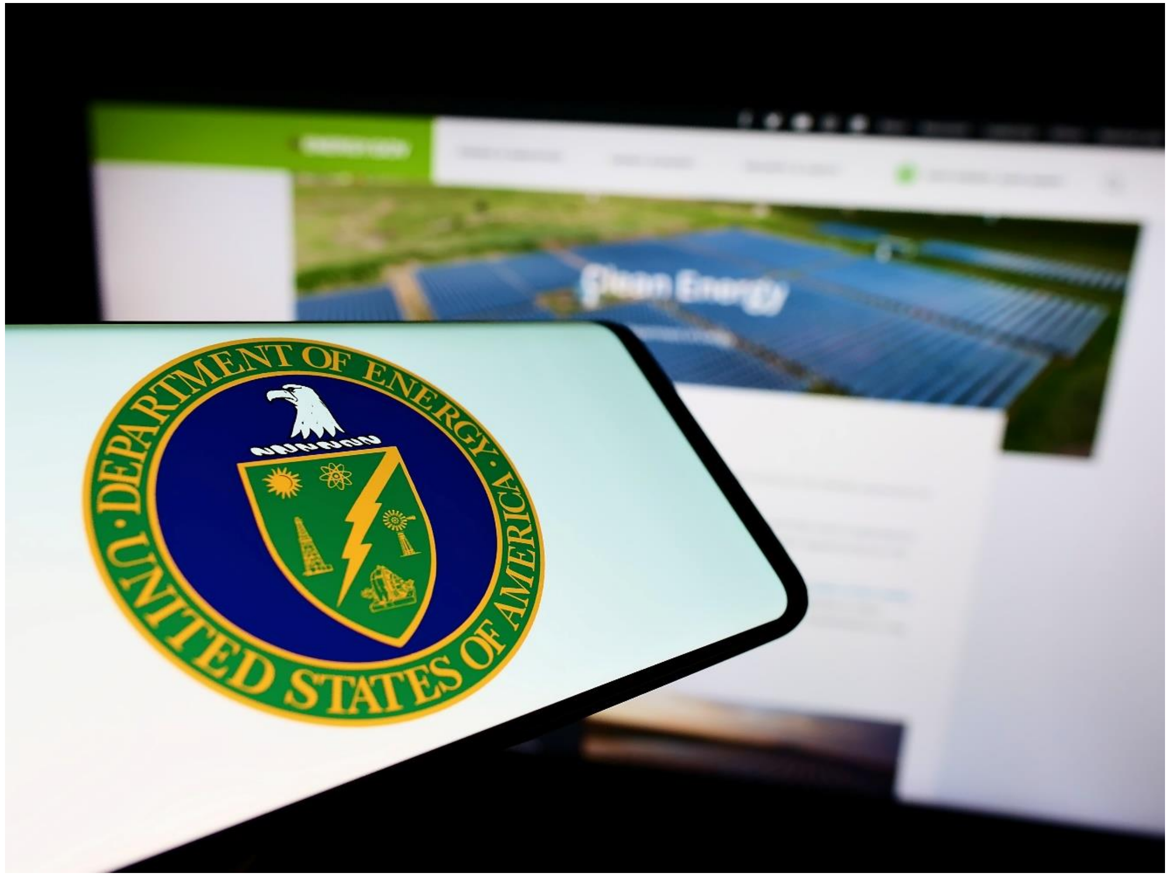
U.S. DEPARTMENT OF ENERGY | Office of NUCLEAR ENERGY | SPENT FUEL & HIGH-LEVEL WASTE DISPOSITION

DISCLAIMER

This is a technical presentation that does not take into account contractual limitations or obligations under the Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste (Standard Contract) (10 CFR Part 961).

To the extent discussions or recommendations in this presentation conflict with the provisions of the Standard Contract, the Standard Contract governs the obligations of the parties, and this presentation in no manner supersedes, overrides, or amends the Standard Contract.

This presentation reflects technical work which could support future decision making by the U.S. Department of Energy (DOE or Department). No inferences should be drawn from this presentation regarding future actions by DOE, which are limited both by the terms of the Standard Contract and Congressional appropriations for the Department to fulfill its obligations under the Nuclear Waste Policy Act including licensing and construction of a spent nuclear fuel repository.



DOE FUNDS ADVANCED REACTOR PROJECTS

ADVANCED REACTOR DEMONSTRATION PROGRAM (ARDP)

Demonstrations

X-energy
Xe-100

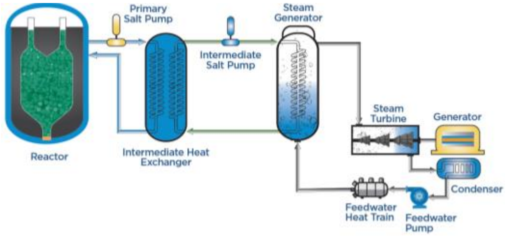
https://x-energy.com/reactors/xe-100

TerraPower
Natrium

https://www.terrapower.com/

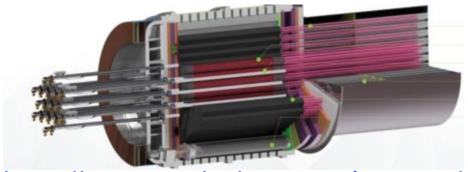
Risk Reduction

Kairos Power
KP-FHR



https://kairospower.com/technology/

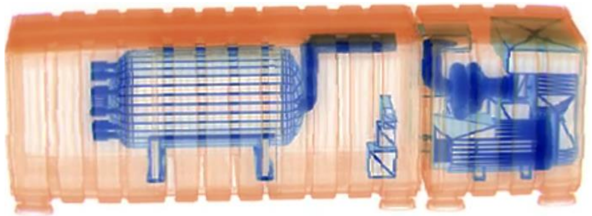
Westinghouse
eVinci



https://www.westinghousenuclear.com/

FRONT-END ENGINEERING AND EXPERIMENT DESIGN (FEEED)

Ultra Safe Nuclear
Pylon



https://www.usnc.com/pylon/

Radiant Industries
Kaleidos

https://www.radiantnuclear.com/

TIMELINE HISTORY OF DEEP GEOLOGIC REPOSITORY PROGRAM

1950s
Active investigation began on the concept of mined geological repositories for disposal of radioactive waste

1970s
U.S. studied feasibility of reprocessing commercial spent nuclear fuel to extract plutonium; approach later canceled, leaving disposal as primary option for managing radioactive waste.

1980
DOE issued the final environmental impact statement: *Management of Commercially Generated Radioactive Waste.*

1981
DOE issued a record of decision announcing its determination to pursue mined geologic disposal for spent nuclear fuel and high-level radioactive waste.

1982
Congress passed the Nuclear Waste Policy Act, establishing a site-selection process for a geologic repository.

1986
DOE issued environmental assessments for five potential sites; recommended three of the five sites for characterization. President Reagan approved the recommendations.

1987
Congress amended the Nuclear Waste Policy Act to direct DOE to characterize only Yucca Mountain as a potential repository.

1995
DOE issued a notice of intent to prepare the Yucca Mountain Repository environmental impact statement.

1999
DOE issued draft Yucca Mountain Repository environmental impact statement.

2001
DOE issued supplement to the draft environmental impact statement. DOE issued documents relevant to site suitability and a potential site recommendation.

2002
DOE issued final Yucca Mountain Repository environmental impact statement. DOE made site suitability determination and recommendation.

2008
DOE submitted a license application to the Nuclear Regulatory Commission.

2010
Yucca Mountain Repository funding ceased.

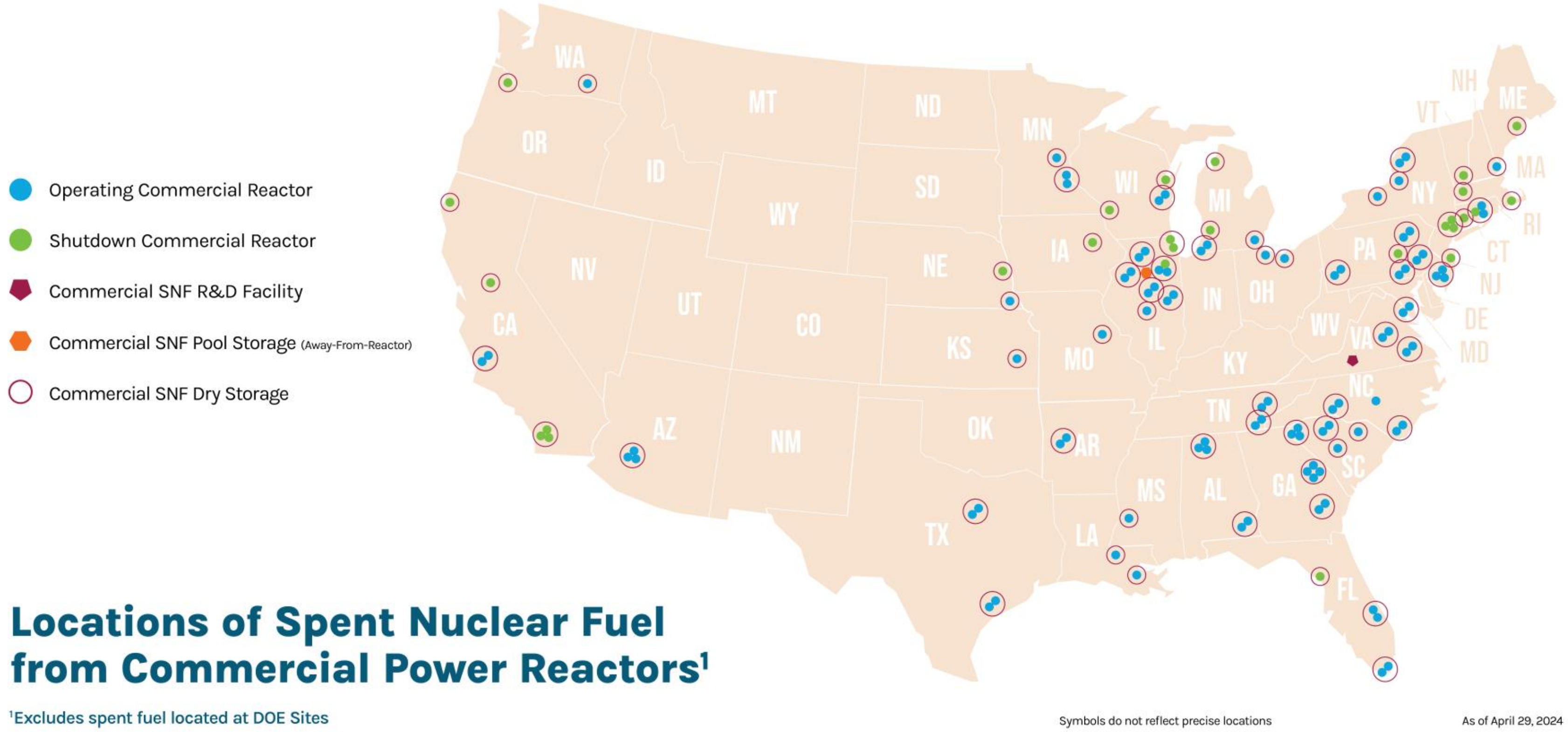
Yucca Mountain repository program ceased.

Program ceased before DOE received construction authorization.

Program ceased before repository emplacement operations began.

2024
Federal consolidated interim storage facility (CISF) critical design (CD-0) granted.

U.S. Spent Nuclear Fuel (SNF) in Context



Locations of Spent Nuclear Fuel from Commercial Power Reactors[1]

[1]Excludes spent fuel located at DOE Sites

1958

United States began using commercial nuclear power

2024

94 operating commercial reactors at 54 nuclear power plants in 28 states

- 20 nuclear power plants have shut down
- ~95,000 metric tons of spent nuclear fuel (SNF)

End of Current Fleet

United States estimated to have more than 140,000 metric tons of spent nuclear fuel

Reprocessing Waste/High-Level Waste (HLW)

HANFORD
7,000 - 9,100
Canisters
(Projected)

IDAHO
1,272 - 11,335 Canisters
(Projected)

West Valley
278 Canisters

TOTAL
4,624 Canisters (2022)
~16,500 - 29,000 Canisters
(Total Projected)

SAVANNAH RIVER
4,346 Canisters (2022)
8,000-8,300 Canisters
(Total Projected)

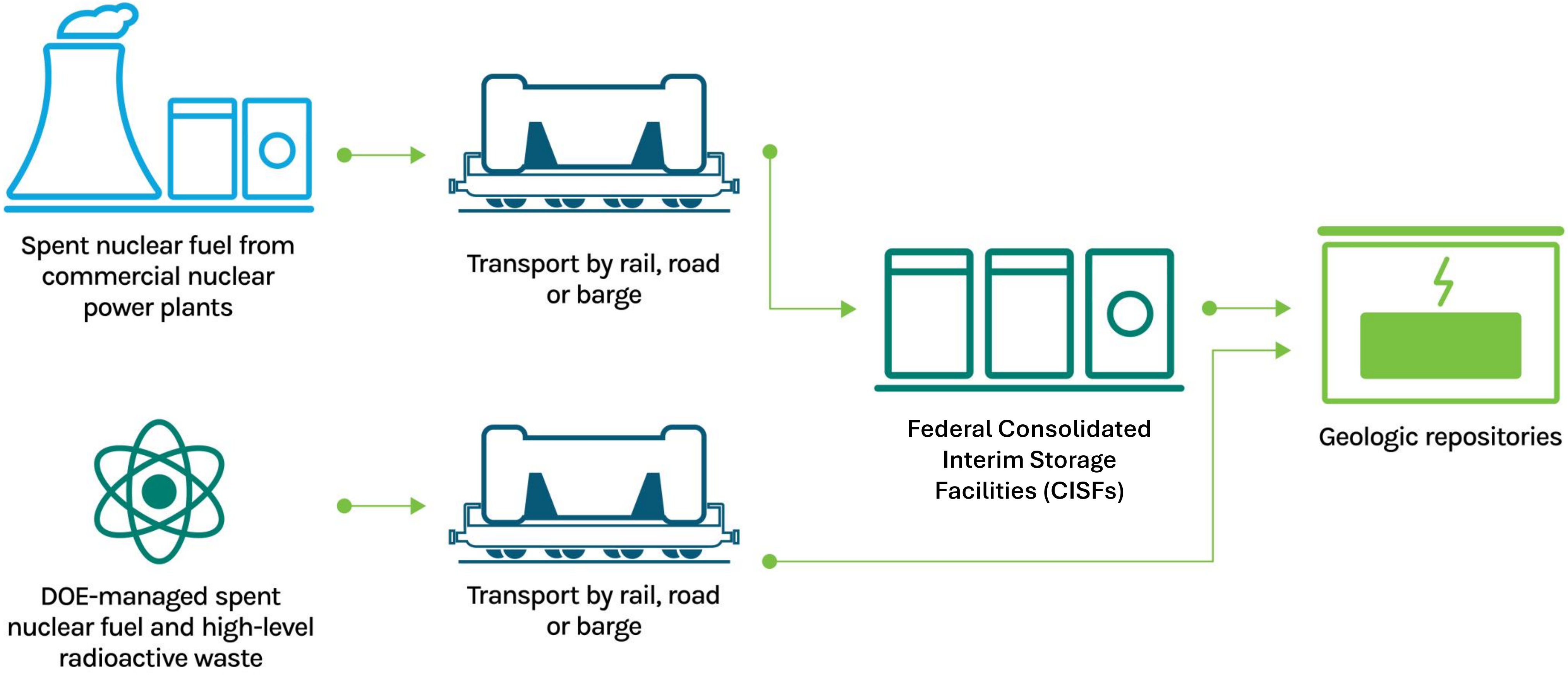
Source: BRC staff graphic updated using information from DOE

Cumulative DOE Liability vs. Cumulative DOE's Spent Fuel Program Appropriations FY 2014-2024

U.S. DEPARTMENT OF ENERGY | Office of NUCLEAR ENERGY | SPENT FUEL & HIGH-LEVEL WASTE DISPOSITION

*Source: DOE Nuclear Waste Fund Annual Financial Statement Audit Reports.
**DOE's cumulative Spent Fuel Program Appropriations includes funding from the Integrated Waste Management System, Used Nuclear Fuel Disposition, and the Nuclear Waste Fund Oversight Programs.

Integrated Waste Management System



High Burnup Research Cask (HBURC)

- NRC licenses the storage of SNF in dry storage.
- For high burnup (HBU) LWR fuel the initial license is for 40 years.
- Most new SNF is now high burn-up, >45 GWd/t.
- The HBU project supports the safe long term dry storage of high burn-up fuel at nuclear power plant sites past 40 years.
- Important to license renewal applications for over 60 of the current commercial fleet
- New reactor/fuel designers may have to demonstrate safe long term storage.

DOE’s Back-End Management of Advanced Reactors (BEMAR)

Need to inform DOE in anticipation of any potential Standard Contract negotiations

BEMAR team formed at the end of 2022, actual work started in early 2023

Led by DOE’s Office of Nuclear Energy
- Office of Clean Energy Demonstrations
- Office of General Counsel
- National Laboratories

BEMAR Goals
1. Technical assessment of the feasibility of storage, transportation, and disposal
2. Rough-order-of-magnitude cost estimate/comparison to LWR SNF inventory

Prioritizing projects near demonstration:
Kairos Power (Hermes/Hermes 2)
X-energy (Xe-100)
TerraPower (Natrium)

National Repository Program

- The repository is designed to a waste acceptance criteria.
  - No RCRA material.
  - No pyrophoric material.
  - No liquids.
- The design of the repository is in part based on criteria such as heat load.
- Everyone pays their fair share of cost for final waste management.
- DOE not responsible for treating SNF or HLW to meet the criteria.
  - **Part of the O&M cost for the reactor operator.**
- **For public confidence retrievability of SNF for 50 to 150 years is expected to be required.**

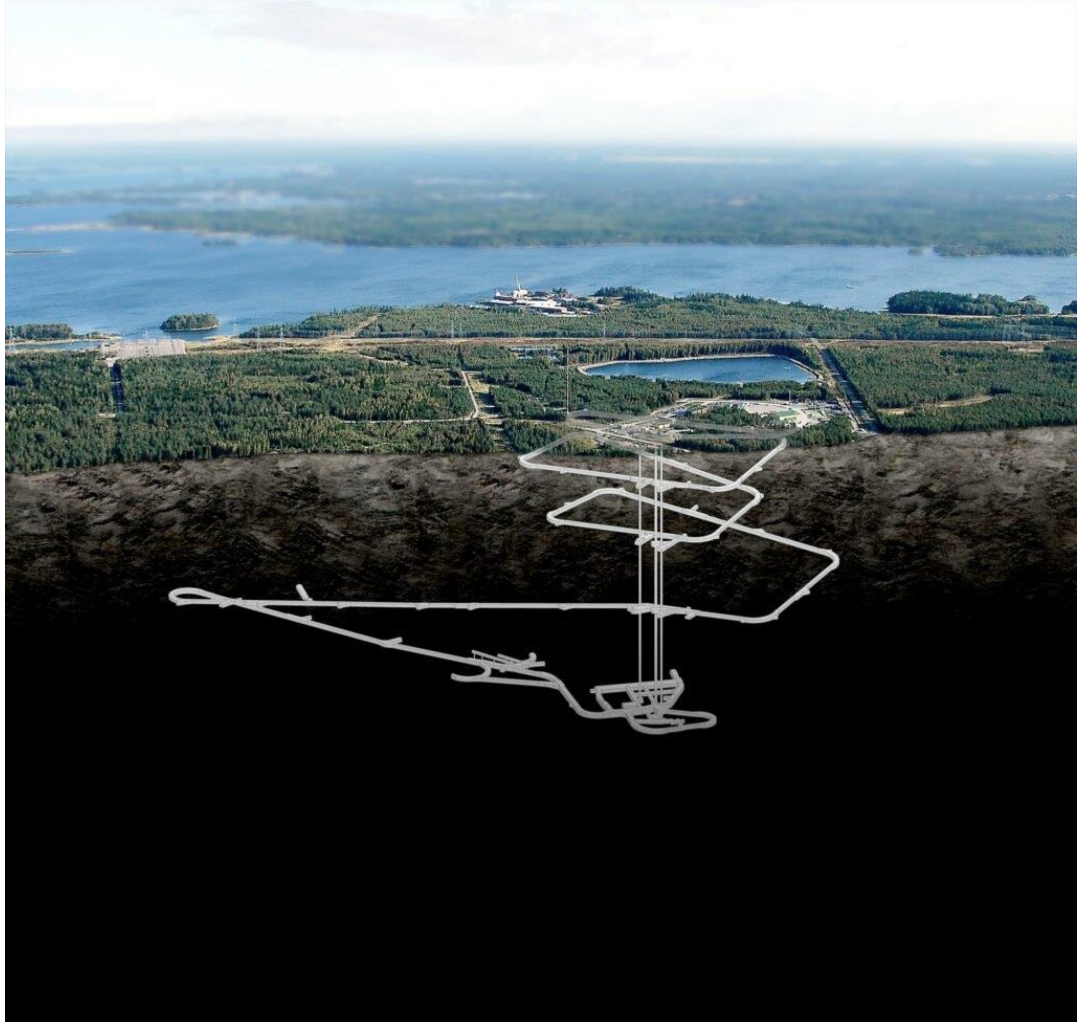


Image credit: Posiva

U.S. DEPARTMENT OF ENERGY | Office of NUCLEAR ENERGY | SPENT FUEL & HIGH-LEVEL WASTE DISPOSITION

Conclusion

- We are over 50 years into a nuclear waste management program.
- Spent nuclear fuel management cannot be ignored and is an important part of the NRC licensing process for new reactors.
- Reactor operator is responsible for costs of conditioning their SNF to meet the waste acceptance criteria.
- DOE is not responsible for the coolant etc.
- Final fee for disposal of the SNF is a function of storage costs, transportation costs, volume in a repository and other technical considerations.
- For final disposal everyone should pay their fair share.
- Reprocessing still results in HLW. EM has reprocessed 140,000 tHM of SNF and will produce ~21,000 HLW canisters of vitrified waste.
- To obtain operating license from NRC. New reactor operators has to sign an amended standard contract with DOE to accept the SNF for long term disposal.

LEARN MORE

Office of Spent Fuel & High-Level Waste Disposition

energy.gov/ne/office-spent-fuel-and-high-level-waste-disposition

THANK YOU

For more information, visit us at:
energy.gov/ne/office-spent-fuel-and-high-level-waste-disposition

EPRI

Advanced Reactor Fuels

The Value of Innovation

NUCLEAR

Craig Stover

Senior Program Manager

Advanced Nuclear Technology Program

Joint Workshop on TRISO and Metallic Fuels

2024 ANT Membership

ANT Participation Extended to Over 90 Companies

NUCLEAR SECTOR BASE MEMBERS

51 Global Members

>83% of the world's commercial nuclear units

>340 reactors worldwide

FULL ANT SUPPLEMENTAL MEMBERS

Bechtel, Fluor Marine Propulsion, GE, Hitachi, SRP, Kairos Power, Nippon Steel & Sumitomo Metal Corporation, Nuclear AMRC, Puget Sound Energy (PSE), NuScale (Power for all humankind), Rolls-Royce, X-energy, Framatome, Doosan, Great British Nuclear, Fluor, TerraPower, epe, Holtec International, Shell, Microsoft, enel Green Power

ADVANCED REACTOR INITIATIVE MEMBERS

The Electric Company (El Paso Electric), NPX, AMS, Frazer-Nash Consultancy (A KBR Company), SGH, Structural Integrity Associates, Inc., IHI, NEIL, Pittsburgh Technical (Enabling Nuclear Power), Worley (energy | chemicals | resources), Paragon, Tractebel (Engie), Orano, New York State NY Power Authority, Jensen Hughes, Dominion Engineering, Inc., Oklo, Studsvik

EPRI

Advanced Nuclear Technology (ANT) Program Focus

MISSION: Accelerating the deployment of nuclear power around the world

- **Informing Resource Planning**
- **Technology Development**
- **Reducing Deployment Costs**
- **Supporting Plant Startup**

Training | Siting & Owner Requirements | Energy Economics | Design & Engineering | Technical Basis | Construction Optimization | Advanced Manufacturing | Commissioning | Initial Operations

- More than 90 companies
- 200+ Past Products
- Dozens of Ongoing Projects

From project initiation through initial operation,
ANT is an extension of your team.

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Existing Products

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DESIGN
70+ Documents

CONSTRUCTION
70+ Documents

40+
Ongoing Projects

Engineering and Construction Innovation (Hasan Charkas)	**Advanced Manufacturing and Materials** (Marc Albert)	**Initial Operations and Maintenance** (Bruce Greer)
Nuclear Design and Fuel Cycle (Dan Moneghan)	**Nuclear Beyond Electricity** (Daniel Klein)	**Project Development and Execution** (Chad Boyer)

3002027650 - Advanced Nuclear Technology: 2024 Product Catalog

Advanced Reactor Roadmap

A shared strategy to ensure success at scale

Serving government, academic, industrial, and public **stakeholders**

Almost 100 GWe of **new nuclear** will be needed by 2050. This means around **300 ARs** in the next **30 years**

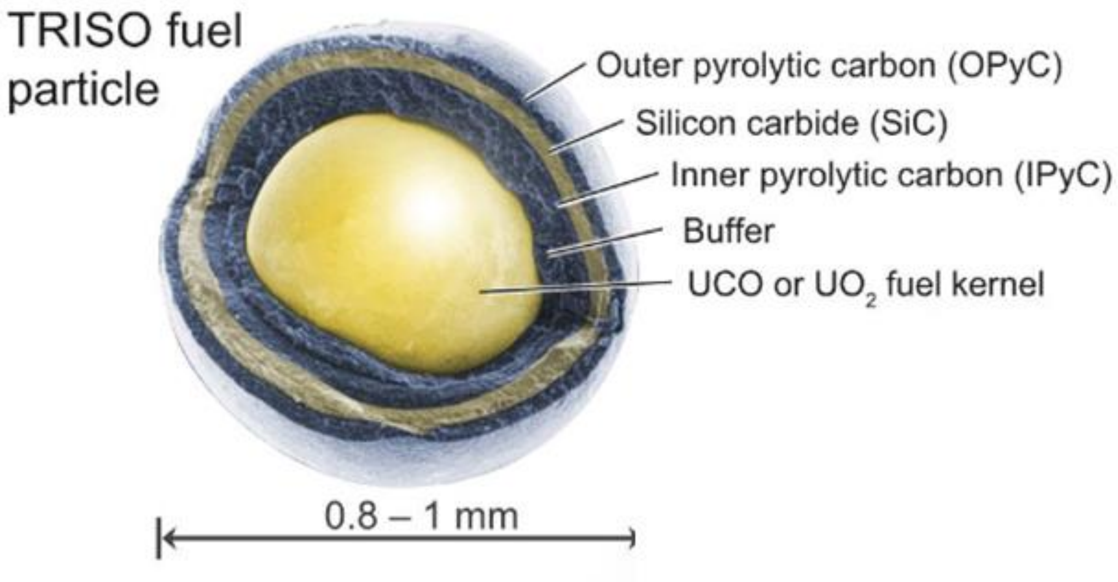
7 Enablers and **46 key actions** chart our path towards a **net-zero future**

Convening the industry for **strategic action**

Industry's roadmap to the future fleet

ARRoadmap.com

Fuel Management Action



Develop spent fuel handling and storage strategy:

Evaluate the applicability of existing storage and transportation technologies to irradiated advanced reactor fuels. Determine any major challenges and identify areas for efficiency and cost improvement. Perform cost assessment for various fuel-cycle back-end strategies. Propose generic waste acceptance criteria for disposal of advanced reactor fuel and wastes.

Need date:

- TRISO, 2025;
  Metallic, 2027;
  Molten Salt, 2028.

TR-FC-01 – Fuel Management

Current Focus – Understanding Where the Gaps Are

Example – TRISO PIRT

Report Coming Soon™
3002029246

Define
Issue, Objectives, Scenarios, Criteria

→ **Identify**
Source Data, Phenomena

→ **Evaluate**
Assess Impact, Rank, Propose Solutions

Purpose: Evaluate if the barriers to radiological release and dose consequence of a TRISO particle can be credited for storage and transportation activities.

Safety Objective: Identify, quantify, and rank the various physical phenomena that affect radiological release and dose consequences of a used TRISO particle.

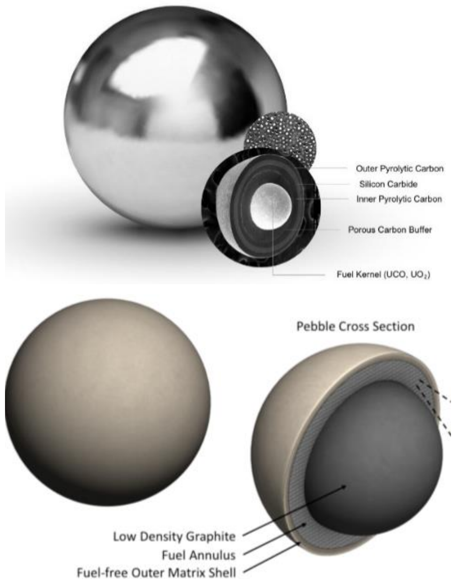
Conclusions: Most phenomena found to have low significance in all scenarios, with a few issues were found to have medium significance in various scenarios. A lack of criticality validation data leads to medium significance in all areas and high in transportation accidents.

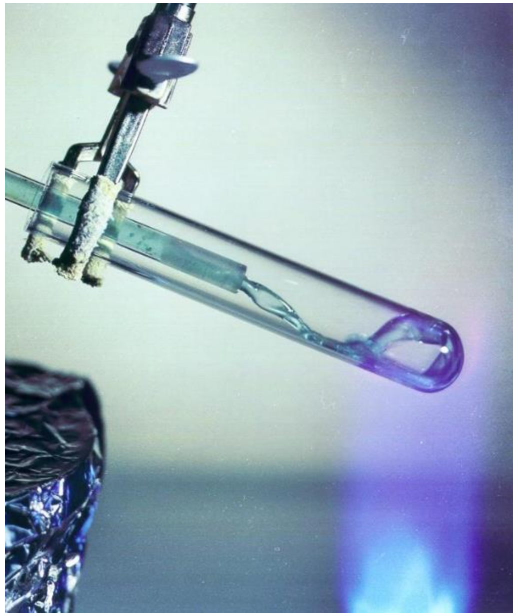
EPRI

Future Focus – Taking a Holistic View

- Taking the learning from the TRISO PIRT and applying process to other fuel types to identify the data gaps and enable prioritization of data needs.
- Bringing the data together with development expertise to look at pathways to commercialization.
- Benefitting from the operating fleet experience with the ESCP AR Fuels Task Group.
- Culminating in a Techno-Economic Assessment of back-end options for various fuels planned for advanced reactors.

Ceramic/Oxide	TRISO	Metallic	Dissolved





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December 3-5, 2024

Gordon Petersen
Spent Fuel Analyst

TRISO Spent Nuclear Fuel PIRT-Storage and Transportation

2024 Workshop on the Storage and Transportation of TRISO and Metal Spent Nuclear Fuels

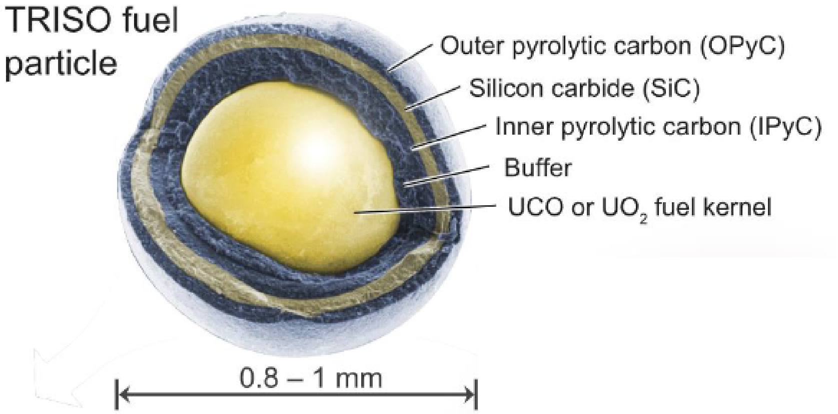
Disclaimer

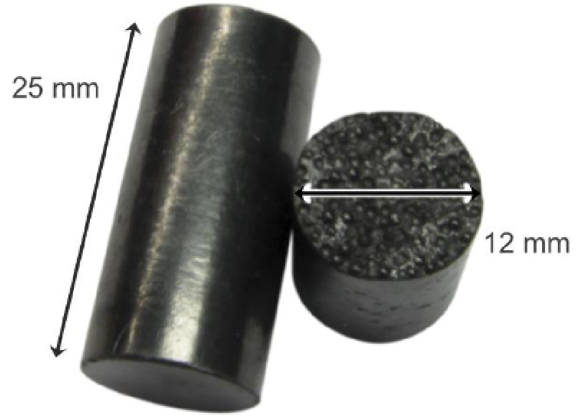
- This presentation reflects technical work which could support future decision making by the U.S. Department of Energy (DOE or Department). No inferences should be drawn from this presentation regarding future actions by DOE, which are limited both by the terms of the Standard Contract and Congressional appropriations for the Department to fulfill its obligations under the Nuclear Waste Policy Act including licensing and construction of a spent nuclear fuel repository.

- The views expressed in this presentation are those of the presenter and may or may not reflect the views of DOE.

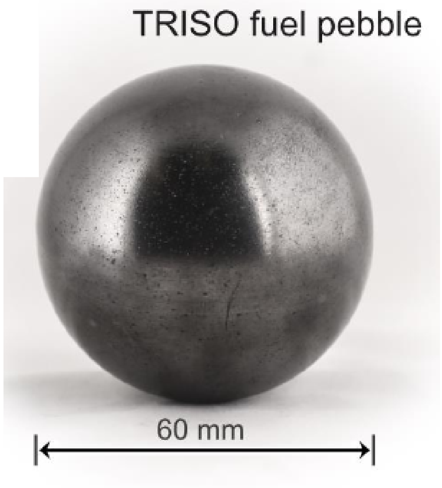
Background

- Tri-structural Isotropic (TRISO) Fuel
  - Layers within the particle work together as a singular containment system
  - TRISO particles combined with carbon matrix into compacts or pebbles
- Phenomena Identification and Ranking Table (PIRT): systematic way of gathering information from experts on a specific concept, and ranking the importance of the information, in order to meet some decision-making objective
  - *Tristructural Isotropic (TRISO) Coated Particle Fuel Performance* [1]
  - *Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance* [2]





TRISO fuel compacts



1. Marciulescu, C., and A. Sowder. Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance. Topical Report EPRI-AR-1(NP). Palo Alto, CA: EPRI, 2019.
2. Nuclear Regulatory Commission. "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.

The PIRT Process

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.

PIRT Purpose and Involvement

- **Purpose**: Evaluate if the barriers to radiological release and dose consequence of a TRISO particle can be credited for storage and transportation activities

Panelists:

- Drew Barto (NRC)
- Jason Piotter (NRC)
- Harold Adkins (PNNL)
- Gordon Petersen (INL)
- Paul Demkowicz (INL)
- Jim Kinsey (INL)
- Steve Nesbit (*industry*)
- Finis Southworth (*industry*)

Observers:

- Tyler Gerczak (ORNL)
- Ryan Latta (Kairos)
- Alex Shrier (BWXT)
- Blaise Collin (USNC)
- Ben Holtzman (NEI)
- Loren Howe (NRC)
- James Tompkins (X-Energy)
- Raymond Wang (X-Energy)

NRC: Nuclear Regulatory Commission
PNNL: Pacific Northwest National Laboratory
INL: Idaho National Laboratory
ORNL: Oak Ridge National Laboratory

BWXT: BWX Technologies
USNC: Ultra Safe Nuclear Corporation
NEI: Nuclear Energy Institute

PIRT Phenomena and Scenarios

- Scenarios
  - Short-term loading activities
  - Storage: Normal up to 60 years
  - Storage: Long-term 60–100 years
  - Storage: Off-normal
  - Storage: Accident
  - Unloading activities
  - Transport: Normal conditions
  - Transport: Hypothetical accident conditions

- Assumptions
  - No volume reduction
  - Cask integrity not compromised via aging management programs
  - Standard fuel handling

- Phenomena
  - Matrix fracture
  - Non-fuel block fracture
  - Abrasive wear
  - TRISO particle layer fracture
  - PyC creep
  - SiC corrosion
  - Particle, block, and matrix oxidation
  - Helium pressurization
  - Fission product leaching
  - Fission product diffusion
  - Neutron multiplication
  - Decay heat

Ranking Rules

- Operability: Does the phenomenon occur?
- Knowledge: Is data available?
- Confidence: What is the quality of existing data and models?
- Significance: What extent does the phenomenon contribute to a release of radionuclides?

Scenario	Operable (Y/N)	Knowledge (L/M/H)	Confidence (L/M/H)	Significance (L/M/H)
Loading Activities	—	—	—	—
Storage: Normal	—	—	—	—
Storage: Long-Term	—	—	—	—
Storage: Off-Normal	—	—	—	—
Storage: Accident	—	—	—	—
Unloading	—	—	—	—
NCT	—	—	—	—
HAC	—	—	—	—

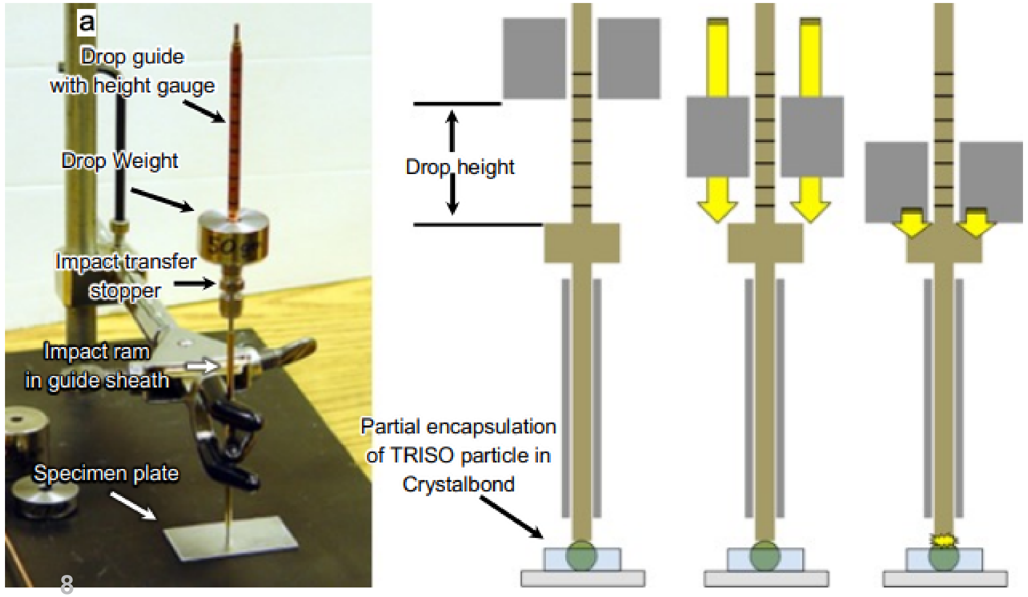
NCT: Normal Conditions of Transport HAC: Hypothetical Accident Conditions

Ranking Conclusions

- Most phenomena had low significance for all scenarios.
- Low to medium quantities of data available.

Medium and High Significance Phenomena

Scenario	Significance (M/H)	Description
Neutron Multiplication	M/H	Criticality validation data is lacking for graphite moderated U-235 and 19.75% enriched uranium.
Oxidation	M	Temperature will be insufficient to drive oxidation in most scenarios.
Non-fuel block fracture	M	Significant mechanical shock could fracture block.
Matrix fracture	M/H	Significant mechanical shock could fracture matrix leading to loose TRISO particles.



Single TRISO particle fracture test set up [3]

3. Gerczak, T., et al. Preparation of Simulated LBL Defects for Round Robin Experiment. ORNL/TM-2015/722-R3, Rev. 3. Oak Ridge, TN: Oak Ridge National Laboratory, September 2022.

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.
- Guidance for spent fuel should be updated for TRISO fuel (NUREG-2215 and -2216). The ongoing analysis by the Nuclear Regulatory Commission (NRC) to review the regulatory framework and determine the impacts of TRISO fuel was deemed necessary by the PIRT panel.

Observations and Recommendations

- The lack of designs for storage and transportation systems makes it challenging to fully evaluate against storage and transportation requirements.
- Most analyses cited in this PIRT were applicable to the Advanced Gas Reactor (AGR) program and the fuel that was examined. The panel believes that uncertainty regarding SNF behavior increases as fuels diverge from the parameters tested in the AGR program.
- Volume reduction activities were not considered and would affect the conclusions of this PIRT.
- Continue the ongoing program to establish criticality benchmarks and nuclear data [4], 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.

4. U.S. Department of Energy, Office of Nuclear Energy. "Criticality Benchmarking." Accessed July 9, 2024. https://www.energy.gov/ne/criticality-benchmarking.

References

1. Marciulescu, C., and A. Sowder. Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance. Topical Report EPRI-AR-1(NP). Palo Alto, CA: EPRI, 2019.
2. Nuclear Regulatory Commission. "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. Gerczak, T., et al. Preparation of Simulated LBL Defects for Round Robin Experiment. ORNL/TM-2015/722-R3, Rev. 3. Oak Ridge, TN: Oak Ridge National Laboratory, September 2022.
4. U.S. Department of Energy, Office of Nuclear Energy. "Criticality Benchmarking." Accessed July 9, 2024. https://www.energy.gov/ne/criticality-benchmarking.

Idaho National Laboratory

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ADVANCING THE VISION OF NEXTGEN FUELS

JASON PIOTTER – NEW FUELS TEAM

DIVISION OF FUEL MANAGEMENT - OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

DIVISION OF FUEL MANAGEMENT - NMSS

Vision Statement

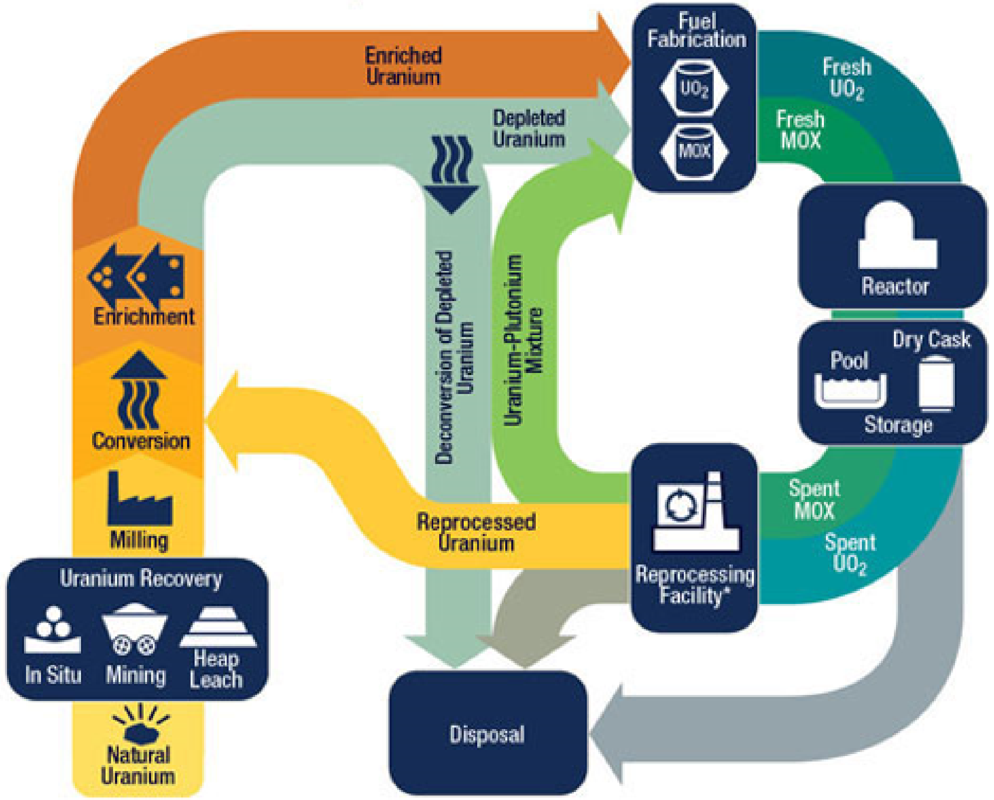
Be an innovative global leader in safe, timely, and efficient regulatory decisions for nuclear fuel management.

Mission Statement

Ensure safe transportation, storage, and possession of nuclear fuels and materials through the application of NRC's Principles of Good Regulation and Values. We accomplish this through timely licensing, appropriate oversight, and effective safeguards.

NEW FUELS TEAM - GENESIS

PURPOSE AND OBJECTIVES



Fuel Cycle Front End

- Critical Experiments
- Code Validation
- Operations
- Security

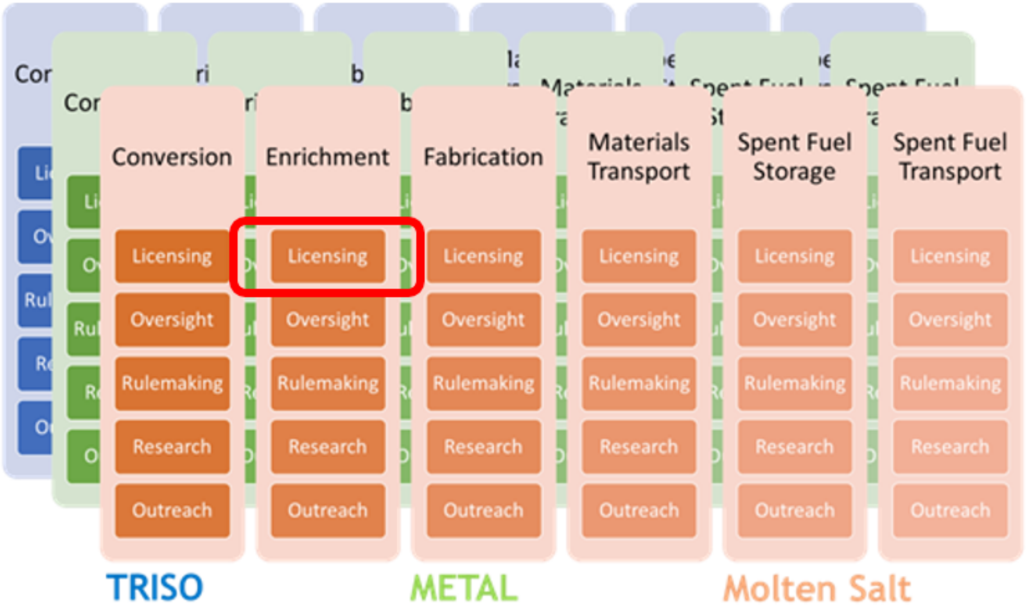
Transportation

- Critical Experiments
- Code Validation

Fuel Cycle Back End

- Source Terms
- Criticality & Shielding
- Thermal & Materials
- Operations

New Fuels Atlas



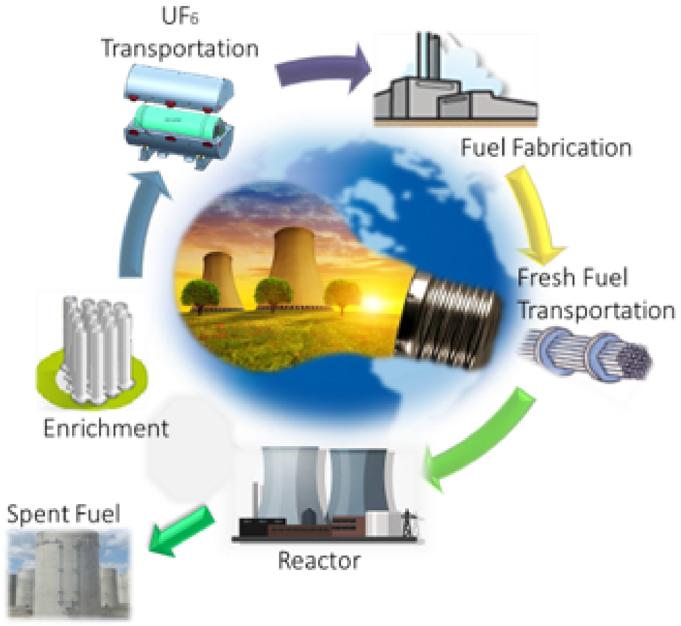
RESEARCH ROADMAP – TRISO & METAL FUELS

Technical Areas	Enrichment*	Fabrication	FF Transportation	Reactor Operation	SF Storage	SF Transportation	SF Disposal
Criticality Safety							
Shielding & Radiation Protection							
Thermal Performance							
Structural Analysis							
Materials Performance				TRISO			
Confinement & Containment							
Fire Safety							
Chemical Process Safety							
Material Control & Accounting							

RESEARCH ROADMAP – TRISO & METAL FUELS

Technical Areas	Enrichment*	Fabrication	FF Transportation	Reactor Operation	SF Storage	SF Transportation	SF Disposal
Criticality Safety							
Shielding & Radiation Protection							
Thermal Performance							
Structural Analysis							
Materials Performance				Metallic			
Confinement & Containment							
Fire Safety							
Chemical Process Safety							
Material Control & Accounting							

TRISO AND METAL FUELS – BACK END OF THE FUEL CYCLE



WHY NOW – PREPAREDNESS MEETS OPPORTUNITY

NEW FUELS ATLAS

Regulatory Planner
Licensing Timelines Dashboard

SRP Updates – Annotated Outline

NUREGs 2215,2216

Regulatory Framework Scan

New Fuels Dashboard Example

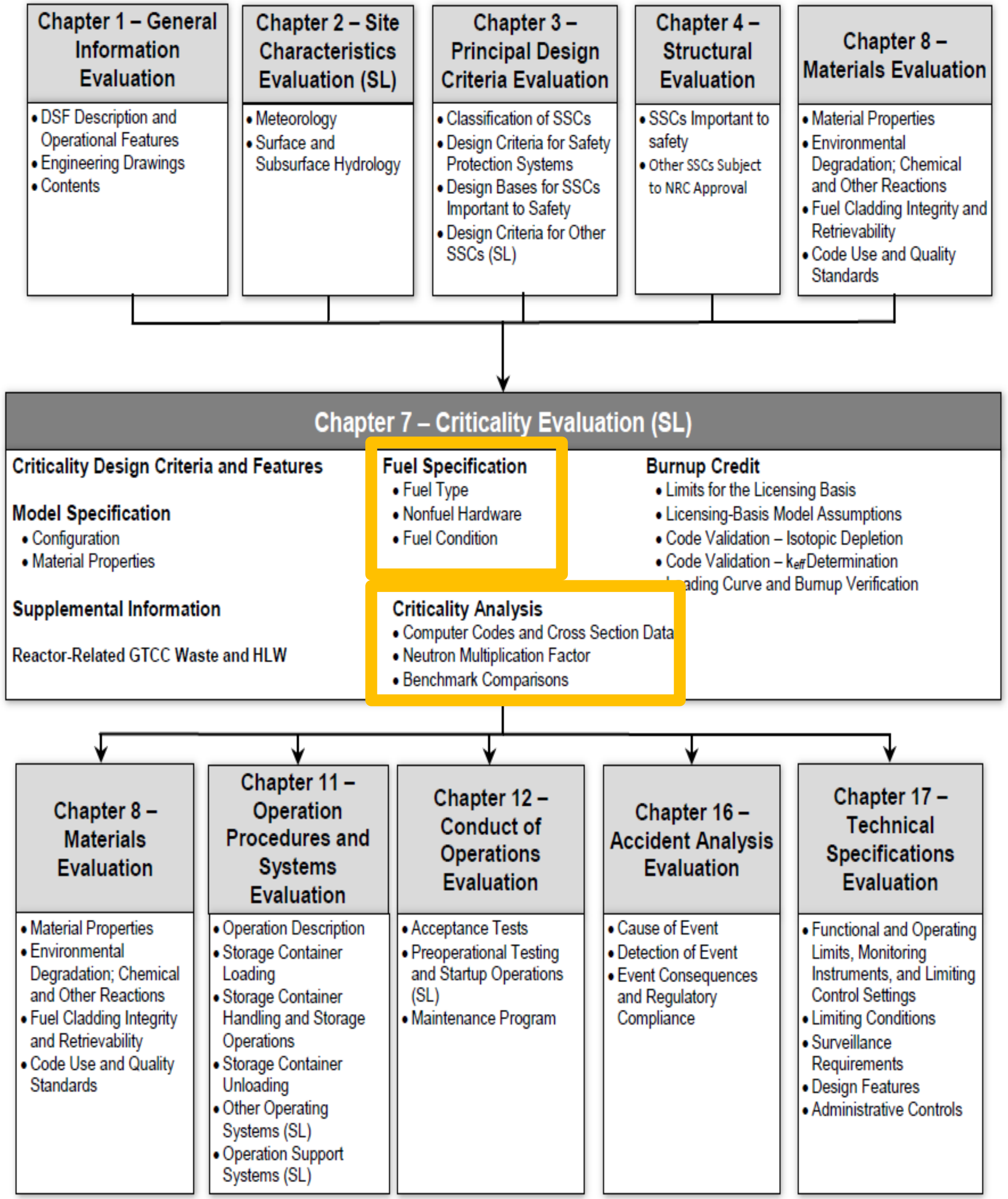
STORAGE

Table 7-1a Relationship of Regulations and Areas of Review for a DSF (SL)

Areas of Review	10 CFR Part 72 Regulations			
	72.24	72.40(a)(13)	72.44(c)	72.124
Criticality Design Criteria and Features	(b)(c)(g)	•	•	(a)(b)(c)
Fuel Specifications	(b)(c)(g)		•	(a)(b)
Model Specification	(d)			(a)(b)
Criticality Analysis	(d)	•		(a)(b)
Burnup Credit	(b)(c)(d)(g)		•	(a)(b)
Reactor-Related GTCC Waste and HLW	(b)(c)(g)		•	(a)

Table 7-1b Relationship of Regulations and Areas of Review for a DSS (CoC)

Areas of Review	10 CFR Part 72 Regulations		
	72.124	72.236(a)	72.236
Criticality Design Criteria and Features	(a)(b)(c)	•	(b)(c)(g)(h)(m)
Fuel Specification	(a)(b)	•	(b)(c)
Model Specification	(a)(b)	•	(b)(c)
Criticality Analysis	(a)(b)	•	(b)(c)
Burnup Credit	(a)(b)	•	(b)(c)(g)



STORAGE

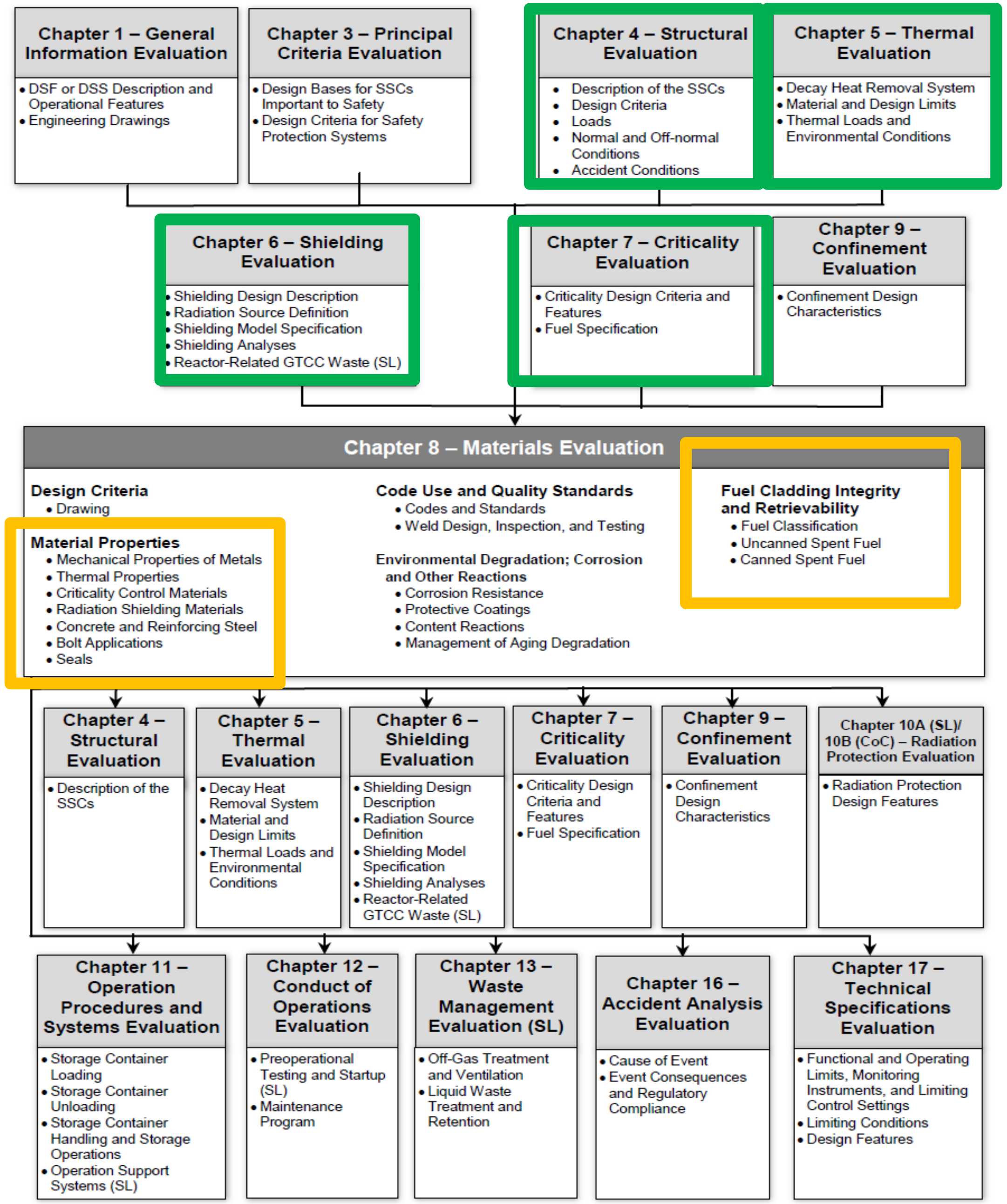
Table 8-1a Relationship of Regulations and Areas of Review for a DSF (SL)

Areas of Review	10 CFR Part 72 Regulations				
	72.24	72.120	72.122	72.124	72.128
Design Criteria	(c)(3)	(a)			(a)
Code Use and Quality Standards	(c)(4)		(a)		
Material Properties	(d)			(b)	
Environmental Degradation; Chemical and Other Reactions		(d)	(b)(1), (c)	(b)	
Fuel Cladding Integrity and Retrievability			(h)(1), (h)(5), (l)		

Table 8-1b Relationship of Regulations and Areas of Review for a DSS (CoC)

Areas of Review	10 CFR Part 72 Regulations			
	72.122[A]	72.124	72.234	72.236
Design Criteria				(b)
Code Use and Quality Standards	(a)		(b)	
Material Properties		(b)		(g)
Environmental Degradation; Chemical and Other Reactions	(b)(1), (c)	(b)		(h)
Fuel Cladding Integrity and Retrievability	(h)(1), (h)(5), (l)			(a), (m)

[A] While not directly applicable to CoCs, DSS design should facilitate general licensee compliance with these requirements.



TRANSPORTATION

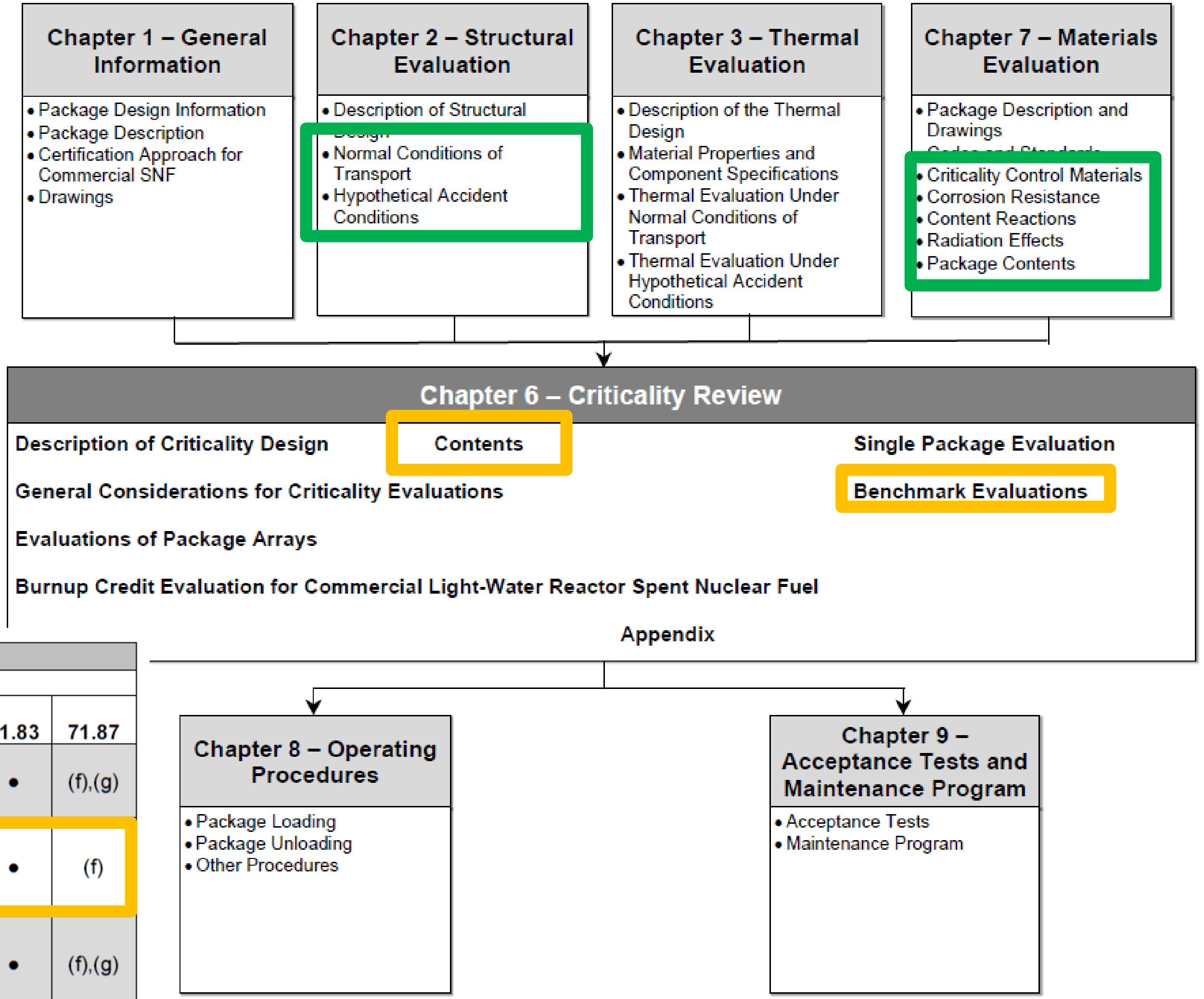


Table 6-1 Relationship of Regulations and Areas of Review for Transportation Packages

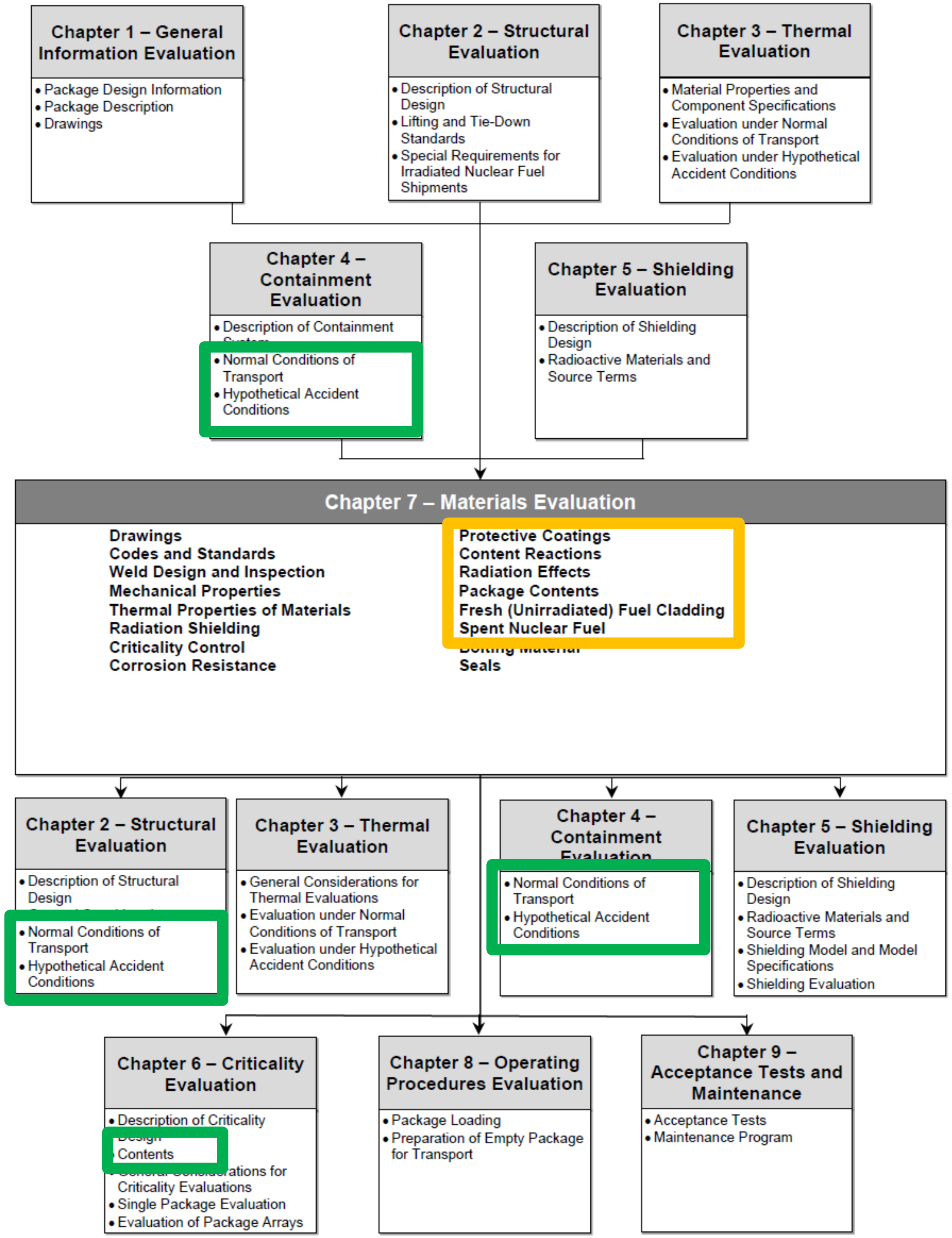
Areas of Review	10 CFR Part 71 Regulations															
	71.31	71.33	71.35	71.41	71.43	71.51	71.55	71.59	71.61	71.63	71.64	71.71	71.73	71.74	71.83	71.87
Description of criticality design	(a)(1), (c)	(a)(1) (5)	(b),(c)		(f)	(a)(1)	(a),(b), (d),(e), (f),(g)	(a),(b)			(a)(1)(iii), (b)(2)				•	(f),(g)
Contents	(a)(1)	(b)(1) (2) (3)(4) (8)					(b),(d), (e),(f),(g)			•					•	(f)
General considerations for criticality evaluations	(a)(2), (b)		(a)	(a),(d)	(d),(f)	(a)(1)	(b),(d), (e),(f),(g)		•	•	(a)(1)(iii), (b)(2)	•	•	•	•	(f),(g)
Single package evaluation	(a)(2), (b)		(a)	(a),(d)	(f)	(a)(1)	(b),(d), (e),(f), (g)		•	•	(a)(1)(iii), (b)(2)	•	•	•		(f),(g)
Evaluations of package arrays	(a)(2), (b)		(a),(b)	(a)	(f)	(a)(1)	(d),(e)	(a)(1) (2),(b)	•	•	(a)(1)(iii), (b)(2)	•	•	•		(f)
Benchmark evaluations	(a)(2), (b),(c)		(a)				(b),(d),(e)	(a)								
Burnup credit evaluation for commercial LWR SNF	(a)(2), (b)	(b)(1) (2)(3) (4)	(a),(c)				(b)(1), (d)(1)(3), (e)(1)(2)	(a)							•	(f)

Note: The bullet (•) indicates the entire regulation as listed in the column heading applies.

TRANSPORTATION

Table 7-1	Relationship of Regulations and Areas of Review for Transportation Packages											
Areas of Review	10 CFR Part 71 Regulations											
	71.31	71.33	71.35	71.43	71.51	71.55	71.64	71.71	71.73	71.74	71.85	71.87
Material description	(a)(1)	•		(c),(d),(f)	(a)(1)	(b),(d),(e),(f)	(a),(b)					
Codes and standards; quality controls	(c)											
Material properties	(a)(1)(2)	•	(a)	(c),(d),(f)	(a)(1)	(b),(d),(e),(f)	(a),(b)	•	•	•	(a)	(a),(b),(c),(f),(g)
Corrosion, chemical reactions, and radiation effects	(a)(2)		(a)	(d),(f)	(a)(1)	(b)(1),(d)(3),(e)(1)(2),(f)		•	•	•	(a)	(a),(b),(c),(f),(g)
Content integrity	(a)(1)(2)	(b)	(a),(c)	(f)	(a)(1)(2)	(b),(d)(2)(4),(e)(1)(2),(f)(1)(2)	(a)	•	•	•		(a),(f)

Note: The bullet (•) indicates the entire regulation as listed in the column heading applies.



THANK YOU

December 3-5, 2024

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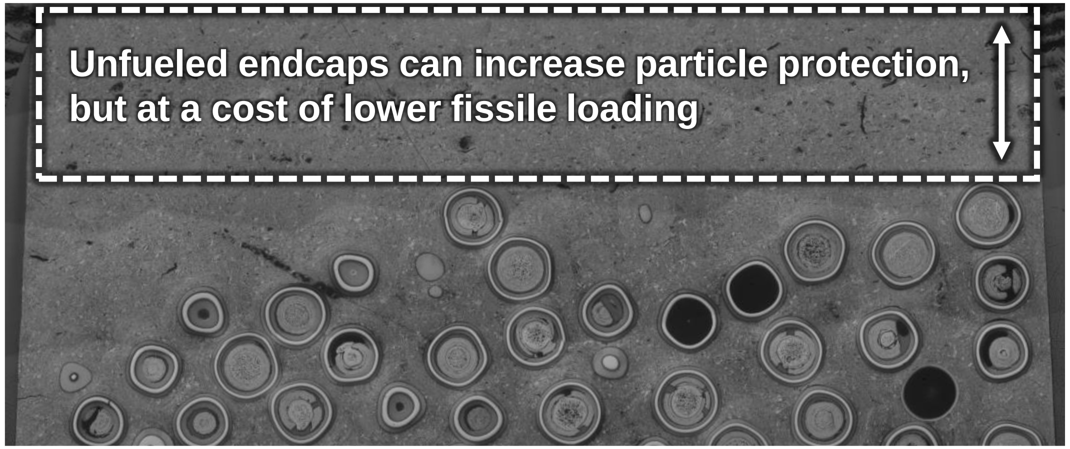
Matrix Structural Integrity

Desirable and undesirable features of matrix materials for TRISO-based fuels

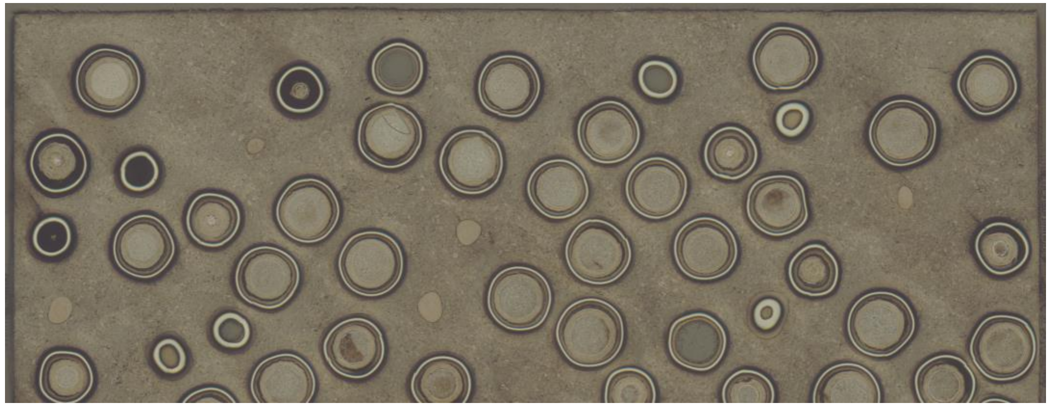
2024 Workshop on Storage and Transportation of TRISO and Metal Spent Nuclear Fuels

One function of matrix is to protect TRISO particles. Some variants may have more protection than others.

Irradiated AGR-1 compact cross section. With endcaps.

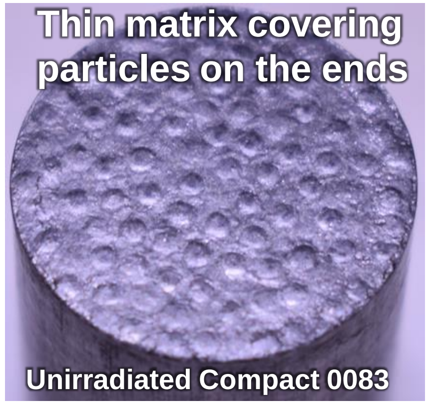


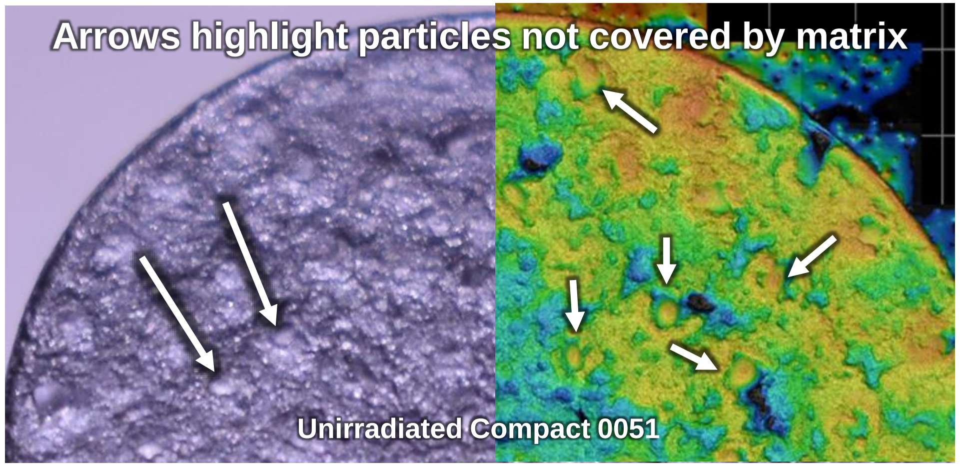
Irradiated AGR-2 compact cross section. No endcaps.



Good irradiation and post-irradiation performance was observed in AGR-1 and AGR-2, with and without endcaps.

AGR-5/6/7 fuel generally has less matrix covering the particles, especially on the top and bottom faces

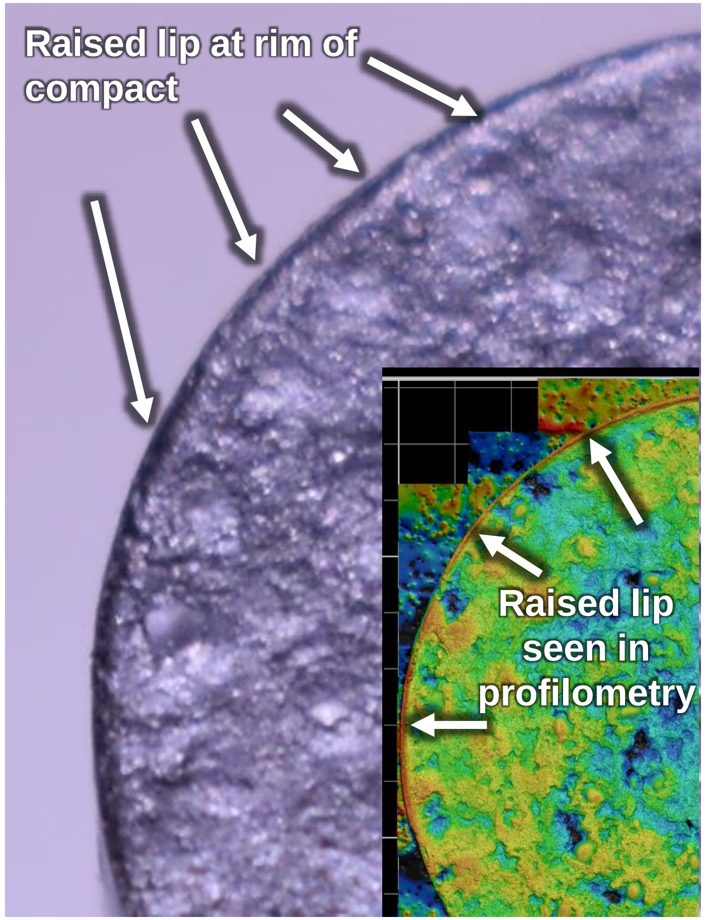


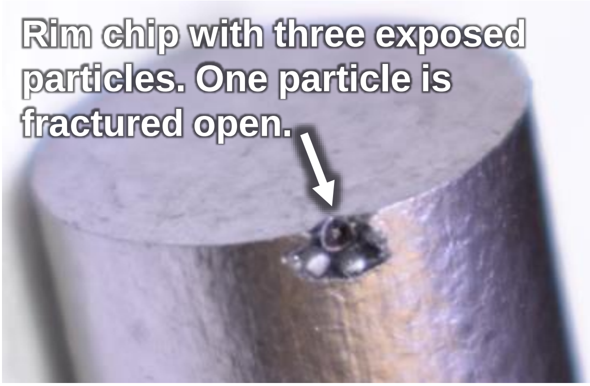


AGR-5/6/7 performance (using A3-27-type matrix) is similar to AGR-1 and AGR-2 (both A3-3-type matrix), but some surface particles have been damaged in post-irradiation handling of the AGR-5/6/7 compacts.

Lip at compact rim may increase susceptibility to handling damage in both as-fabricated and irradiated fuel

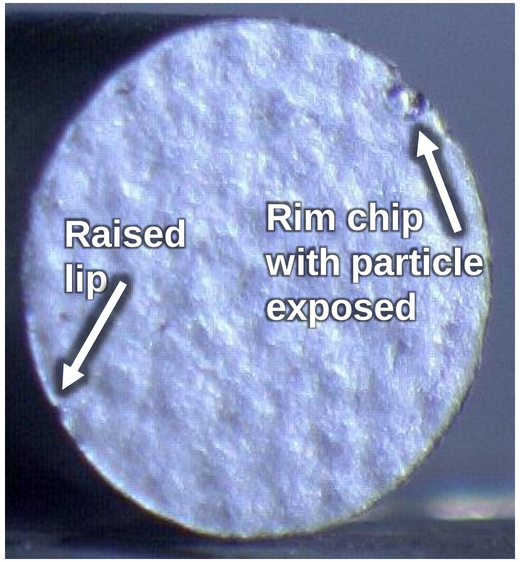
Examples from three as-fabricated compacts





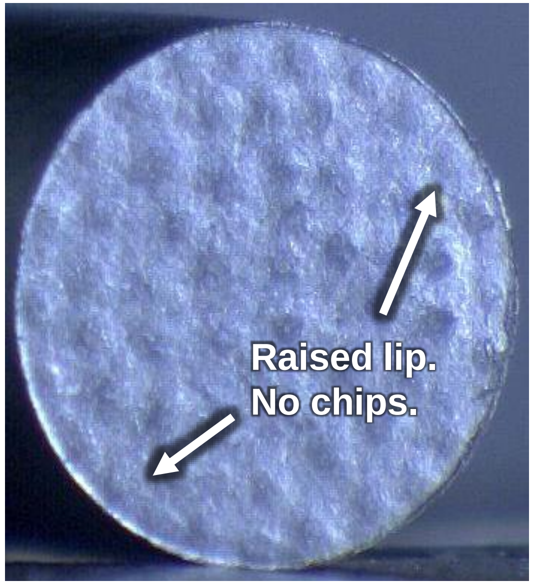
These chipped compacts were separated from compact lot, and not used in irradiations.

Examples from irradiated compacts



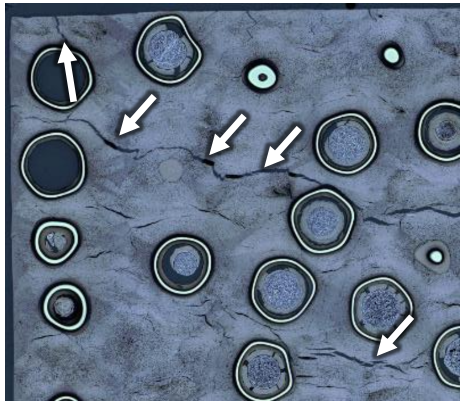
- Most compacts with raised lips do NOT have rim chips, however.

- Not all variants will have raised lips. AGR-1 & 2 compacts do not seem to have these.

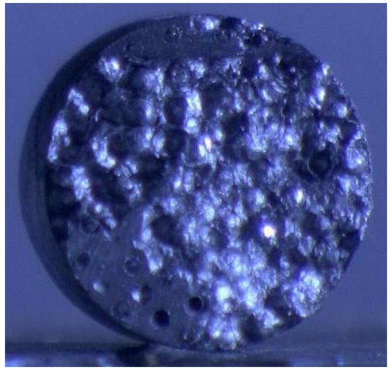


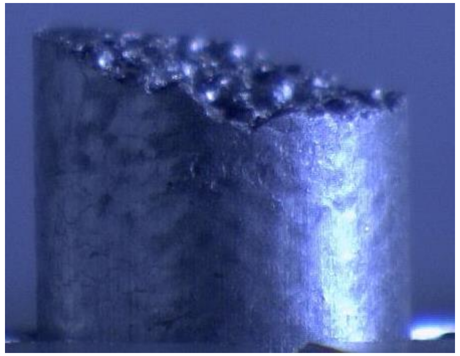
Matrix Fissures or Cracking: some amount is OK

- In AGR carbon-matrix fuel (below), matrix fissures have not been observed to breach the particles. SiC-matrix cracks extending into particles have been observed [1, 2].

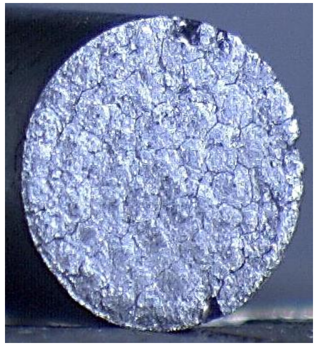


- AGR-5/6/7 Compact 2-8-4 accidentally broken by a power tool during irradiation test train disassembly. Estimated <3.5% of particles broken. The other >96.5% are intact and will retain fission products as usual.

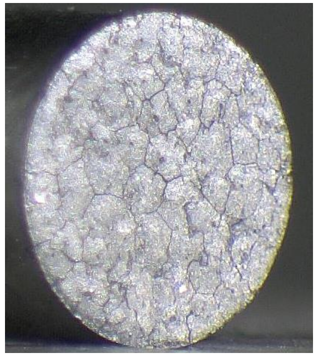




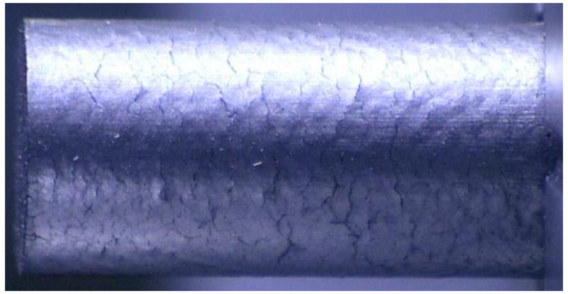
- Fissures extending to the compact surface are typically not a problem, but they may
  - Act as pathway for oxidants (air/moisture) in accidents [3]
  - Contribute to rim chips during handling



Surface fissures with rim chips



Surface fissures NO rim chips



Surface fissures NO chips

- Irradiated AGR fuel compacts routinely
  - Withstand harsh pneumatic transfers between PIE facilities
  - Are shipped by land and air between INL (Idaho) and ORNL (Tennessee)

1. Petrie et al. 2023. J. Nucl. Mater, Vol 580, 155419.
2. Schappel et al. 2023. J Nucl. Mater. Vol 586, 154691.
3. Gerczak et al. 2019. ORNL/TM-2019/1341

Battelle Energy Alliance manages INL for the U.S. Department of Energy's Office of Nuclear Energy. INL is the nation's center for nuclear energy research and development, and also performs research in each of DOE's strategic goal areas: energy, national security, science and the environment.

OAK RIDGE
National Laboratory

Implications of coated particle fuels with new architectures for an expanded service envelope

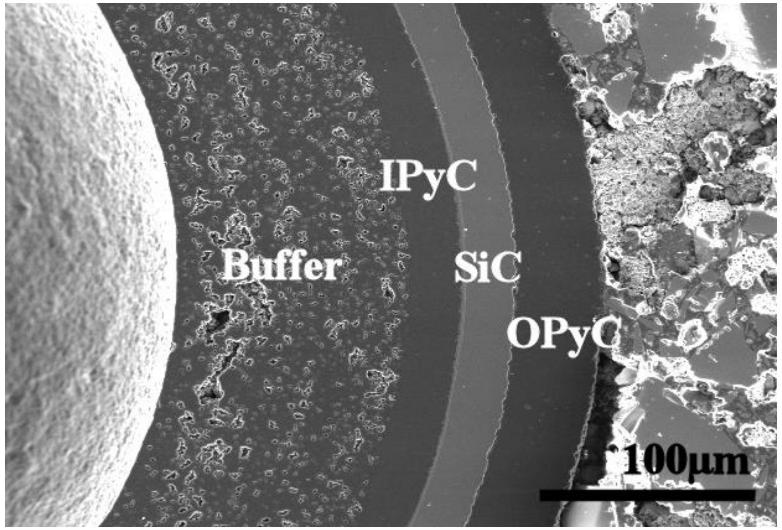
Eddie Lopez Honorato, Tyler Gerczak

Nuclear Energy and Fuel Cycle Division

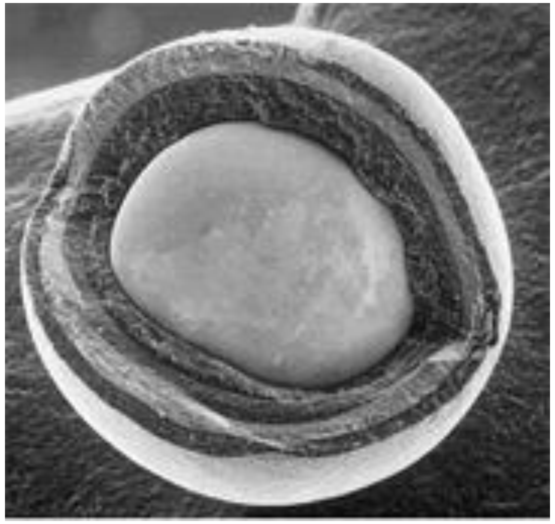
U.S. DEPARTMENT OF ENERGY

ORNL IS MANAGED BY UT-BATTELLE LLC
FOR THE US DEPARTMENT OF ENERGY

TRISO fuel particle



- **Kernel** (UO_2 or UO_2/UC/UC_2, 350-500 µm)
- **Buffer Layer (Low density PyC, 100 µm)**
  - ✓ Void volume for fission gases
- **IPyC (40 µm)**
  - ✓ Stops some fission products and protects SiC
  - ✓ Seals off the buffer
  - ✓ Protects the kernel from chlorine attack
- **SiC (35 µm)**
  - ✓ Retains gas and metal fission products
  - ✓ Provides mechanical stability
- **OPyC (40 µm)**
  - ✓ Protects SiC mechanically
  - ✓ Fission product barrier in particles with defective SiC



This fuel was originally designed for gas-cooled high-temperature reactors

The Advanced Gas Reactor (AGR) Fuel Development and Qualification Program has demonstrated the robustness of this fuel

Beyond TRISO

DOE- Advanced Reactor Demonstration Program

TRISO and other coated particle fuels are candidates for several modular, gas-cooled or salt-cooled rectors and micro reactors concepts

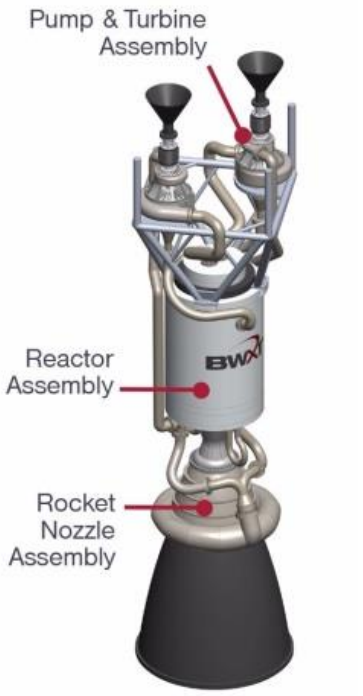
NASA- Space exploration

Nuclear thermal propulsion engine and fission surface power systems

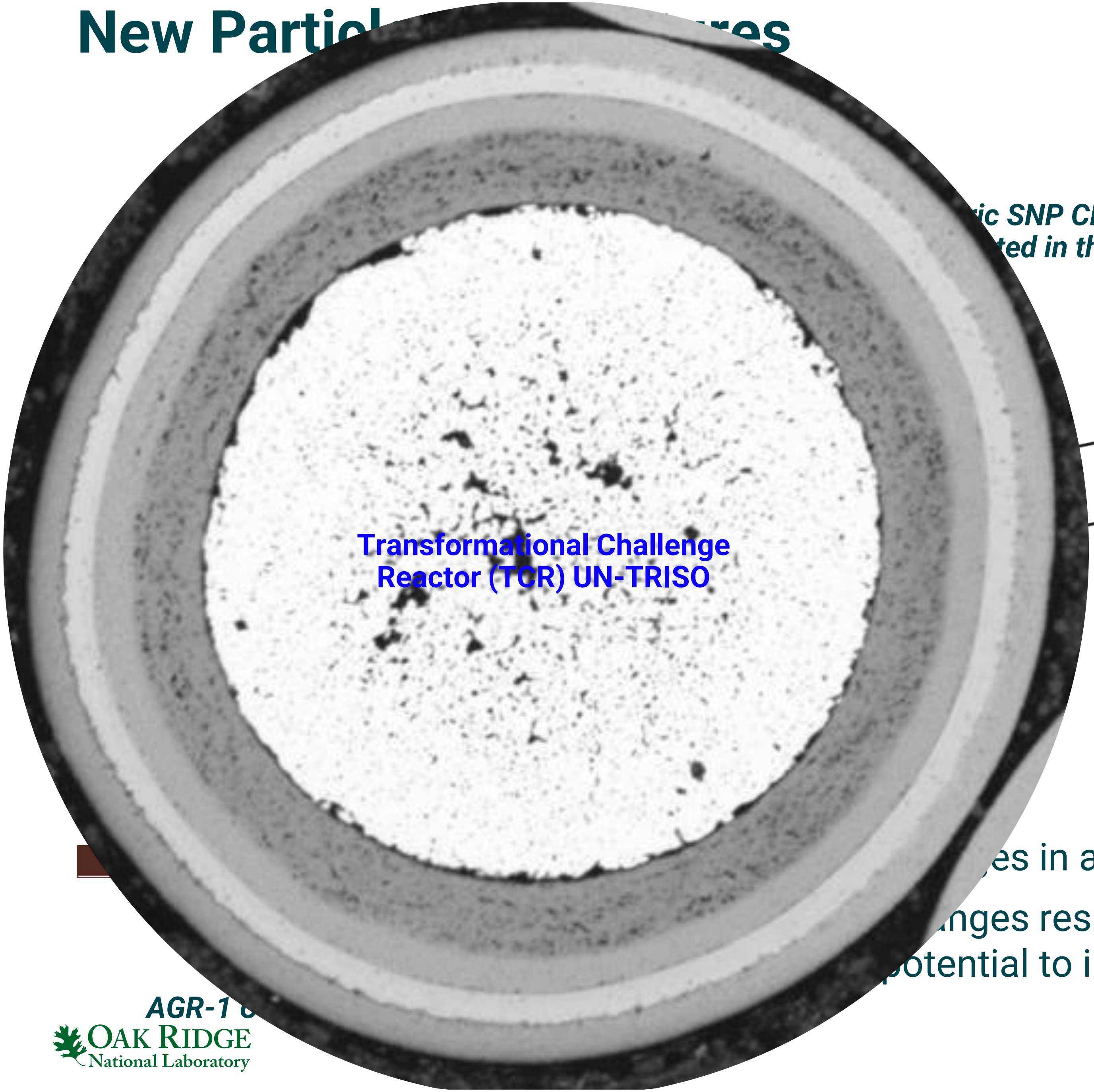
Many of these new reactor configurations require COATED PARTICLES with different characteristics from traditional TRISO

"Lack of timely and affordable commercial availability of TRISO fuel is the most critical market limitation"

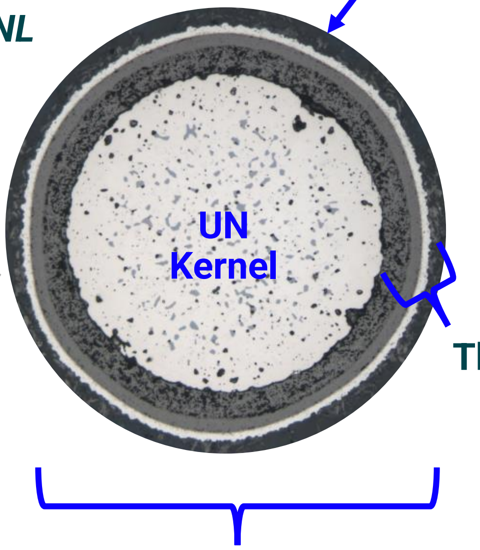
Industrial feedback- Advanced Reactor Demonstration Program



New Partic res



ic SNP CERCER Particle
ted in the CPFDL at ORNL

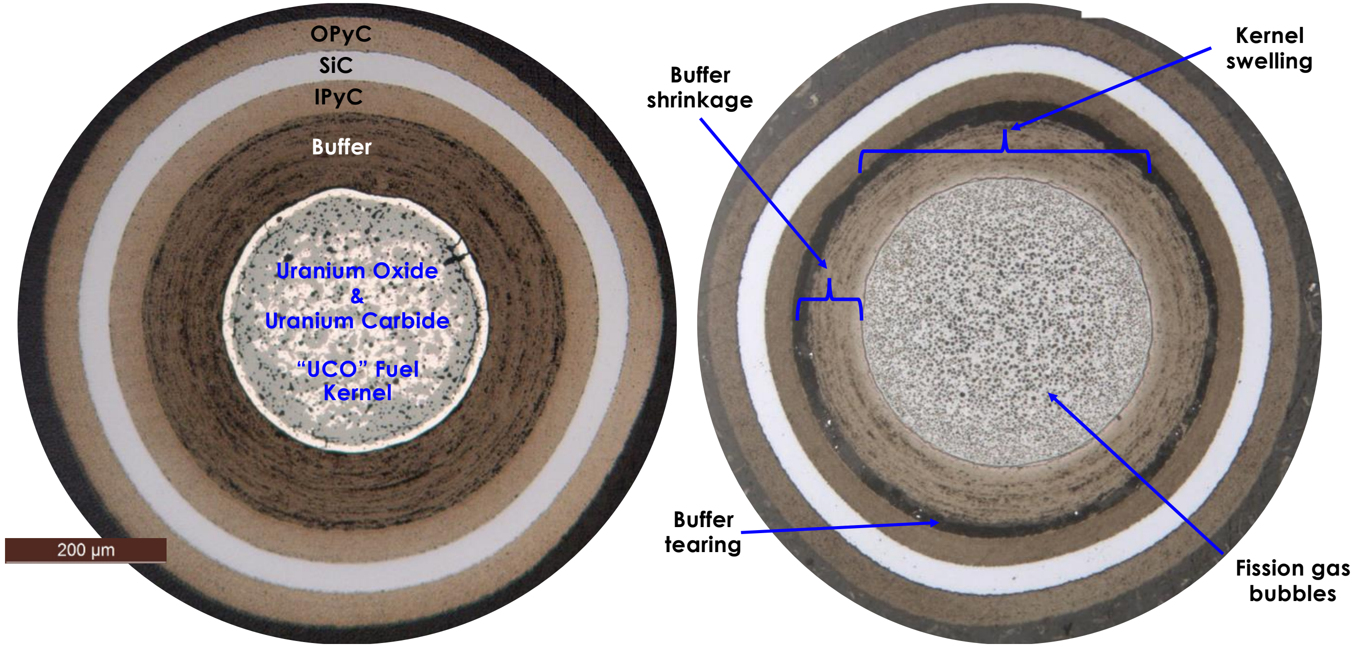


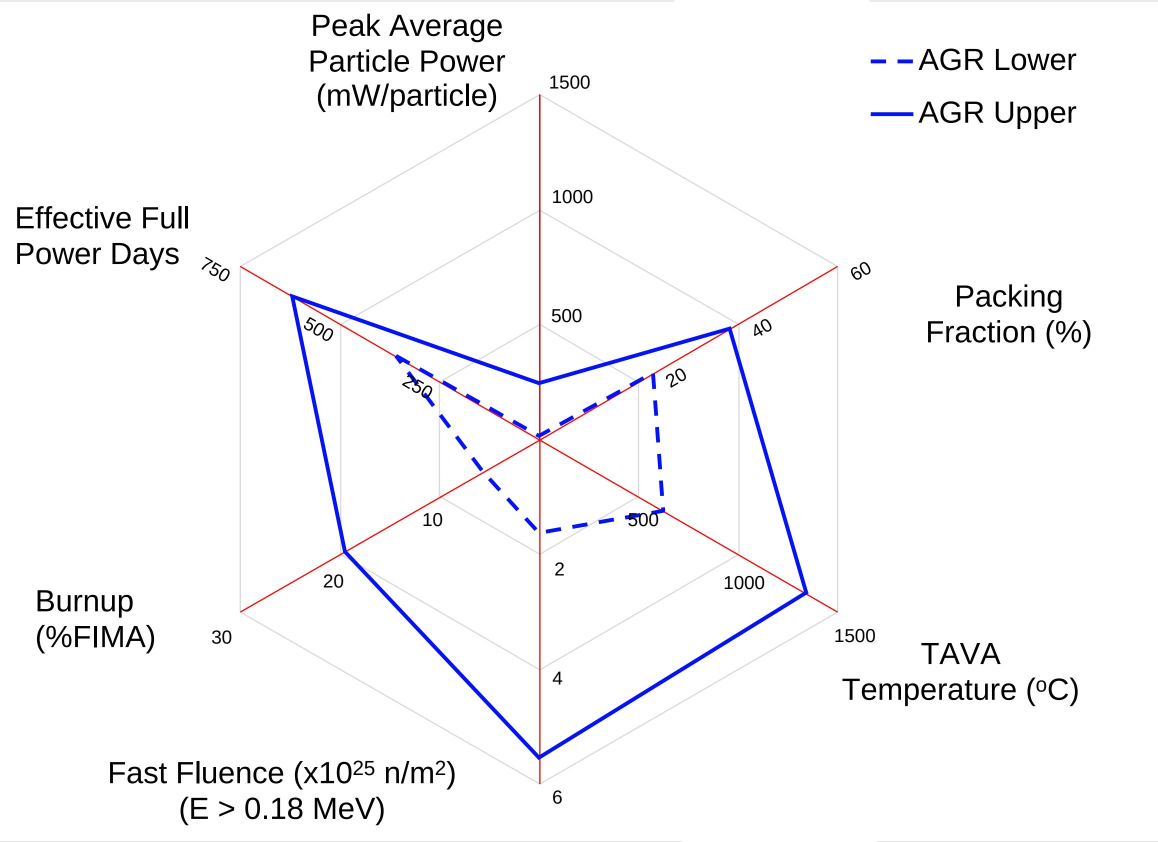
es in architecture to account for different functionality

nges result in new challenges for synthesis and coating and
potential to introduce "unaccounted for phenomena"

AGR-1

Fission and high operating temperatures leads to a complex, evolving materials system





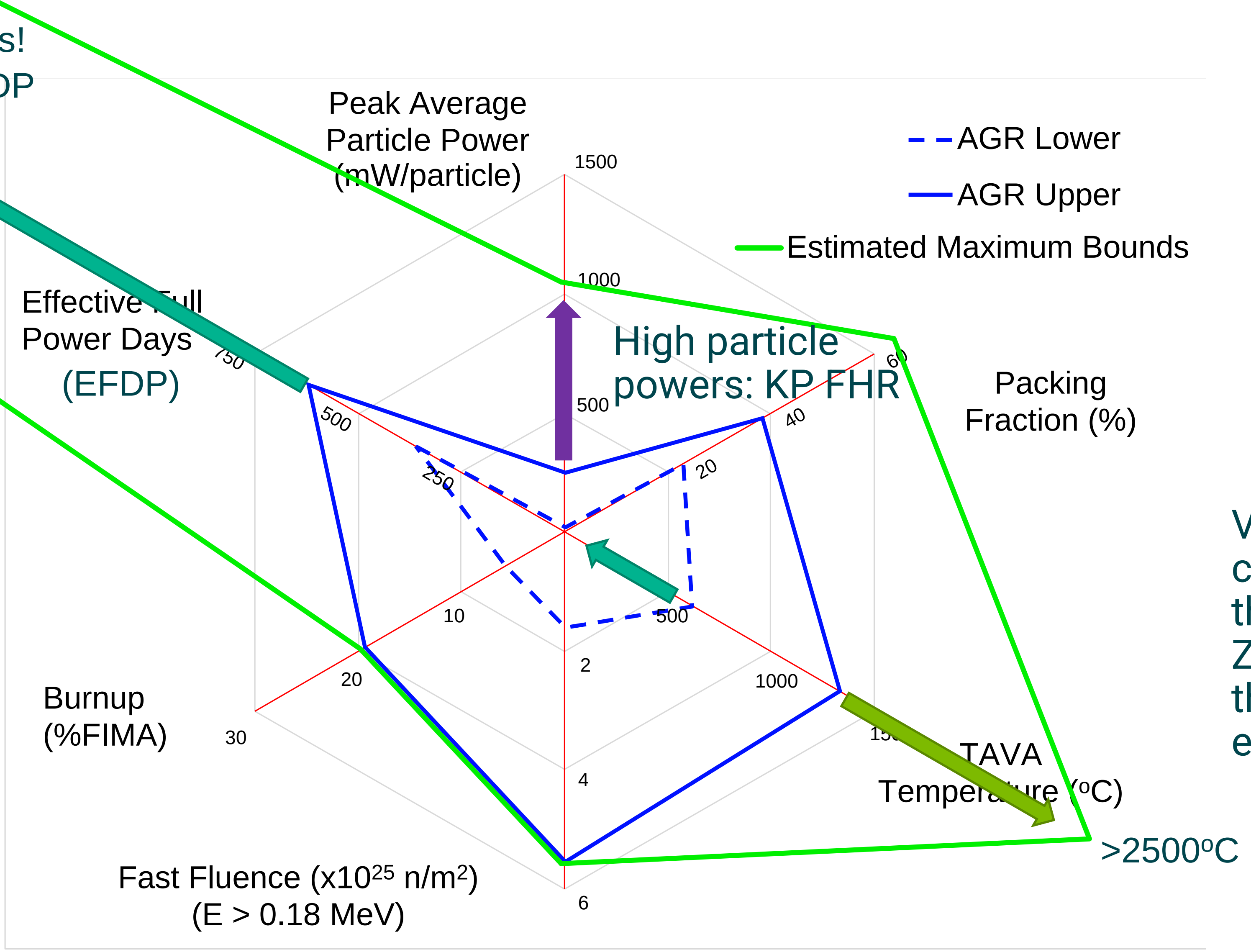
Fuel temperatures up to ~1400°C
Burnups up to ~20% FIMA

FIMA = Fissions per initial metal atom

Extending performance beyond established operational envelope

Microreactors:

UCO in graphite matrix compacts but also SiC matrix fuel forms, UN - TRISO



VHTRs or other concepts (e.g. Nuclear thermal Propulsion): ZrC coatings pushing the temperature envelope

Change in anisotropy and density will have an impact on the mechanical properties of PyC

Table 3 – Changes of mechanical properties of PyC coatings after thermal treatment.

Sample	As deposited		Thermal treated at 1800 °C		Thermal treated at 2000 °C	
	H (GPa)	E (GPa)	H (GPa)	E (GPa)	H (GPa)	E (GPa)
High density						
C1	4.68 ± 0.25	26.70 ± 1.19	1.03 ± 0.18	14.82 ± 1.31	0.90 ± 0.13	13.37 ± 0.93
C2	4.35 ± 0.48	25.13 ± 1.17	1.32 ± 0.19	10.91 ± 0.69	0.76 ± 0.21	12.04 ± 1.26
C3	4.90 ± 0.36	28.78 ± 1.17	–	–	0.91 ± 0.26	12.71 ± 1.25
C4	3.97 ± 0.19	22.91 ± 0.76	1.71 ± 0.10	13.13 ± 0.34	1.10 ± 0.10	13.70 ± 0.51
C5	4.56 ± 0.10	26.10 ± 0.36	1.32 ± 0.15	11.77 ± 0.51	1.77 ± 0.25	13.61 ± 1.01
Low density						
C6	3.88 ± 0.35	21.65 ± 1.91	2.96 ± 0.22	19.12 ± 1.13	2.44 ± 0.23	16.47 ± 0.88
C7	3.95 ± 0.53	21.49 ± 2.00	2.92 ± 0.36	19.34 ± 1.14	2.32 ± 0.33	15.68 ± 1.82
C8	3.54 ± 0.27	19.45 ± 0.70	2.92 ± 0.36	19.04 ± 1.13	2.32 ± 0.63	16.78 ± 2.40
C9	2.84 ± 0.40	19.38 ± 0.94	2.26 ± 0.57	16.77 ± 1.78	2.63 ± 0.42	17.33 ± 1.51
C10	1.89 ± 0.09	12.66 ± 0.35	2.13 ± 0.19	13.63 ± 0.76	1.88 ± 0.23	13.81 ± 0.87
C11	1.68 ± 0.17	11.66 ± 0.82	1.78 ± 0.34	12.84 ± 1.06	0.86 ± 0.14	11.67 ± 1.51

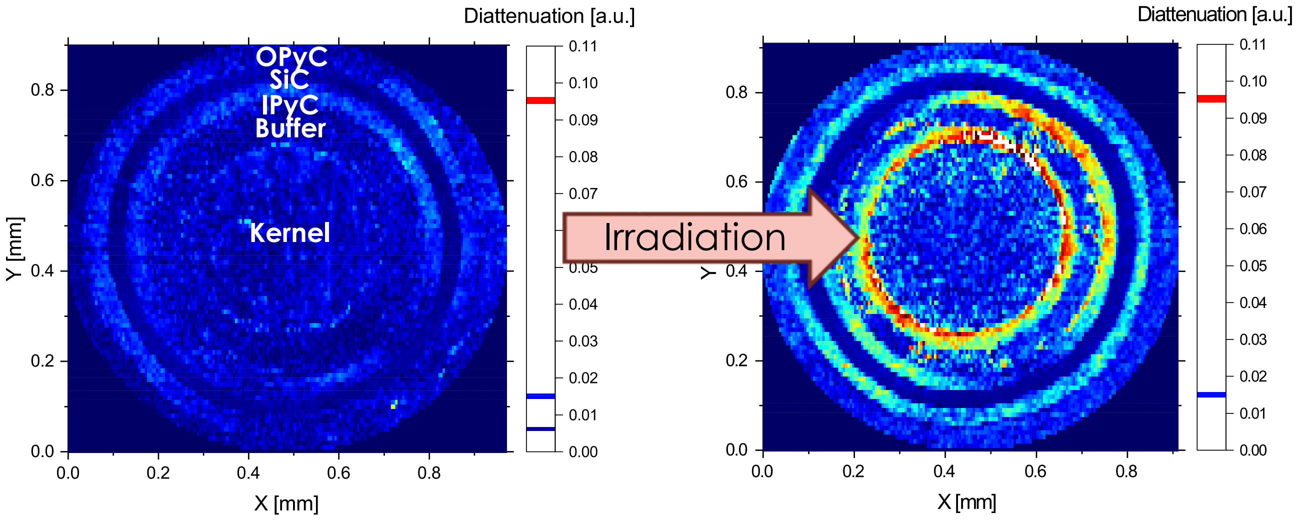
Zhang, Lopez Honorato, Xiao. Carbon 91 (2015) 346

*Preliminary data	Diattenuation (N) (St. Dev.)	
Sample	IPyC	OPyC
AGR-2-LEU09 Unirradiated/compacted	**0.0166** (0.009)	**0.0136** (0.007)
AGR-2-221 12.5% FIMA/1287°C TAVA	**0.0378** (0.019)	**0.0382** (0.011)
Relative Change in N	**0.0212**	**0.0246**
Predicted Change in N [1]	**0.0133 (~60–85% smaller than measured)**	

Diattenuation values representative of average from measurement of 5 particles

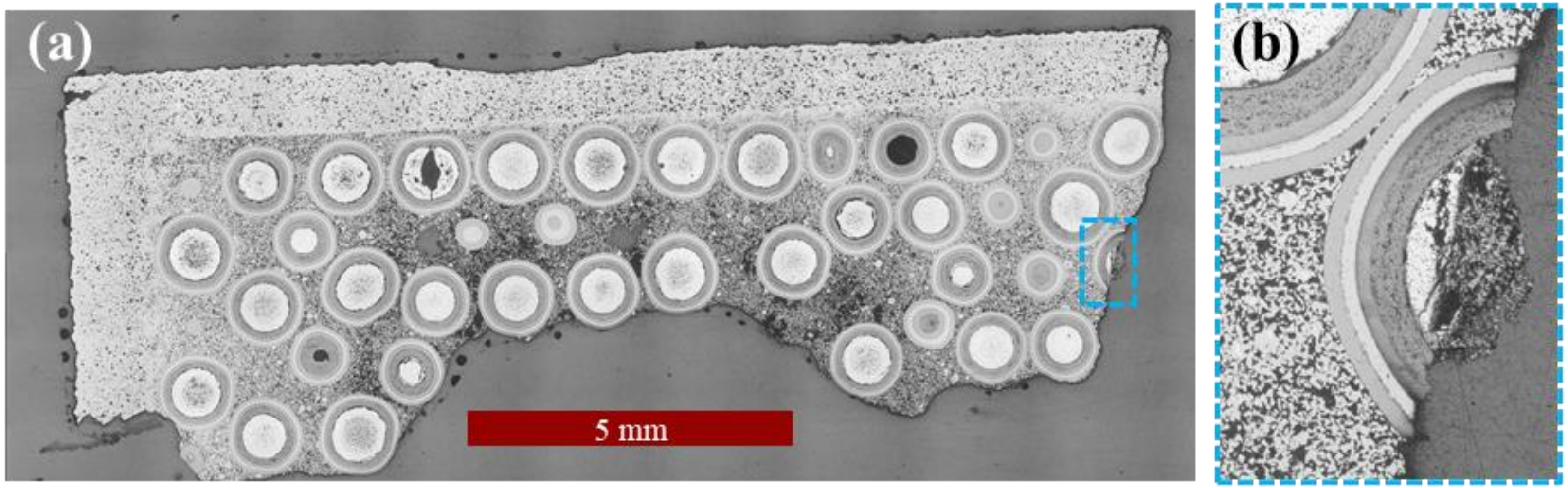
We still have knowledge gaps

- Anisotropy of PyC layers influences the thermomechanical response of the particles (e.g., probability of cracking the layers!)
- What would it be the impact of longer times, higher temperatures, etc.?

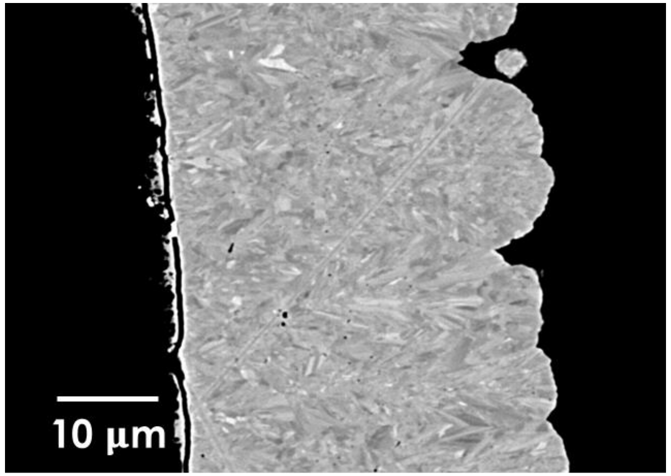


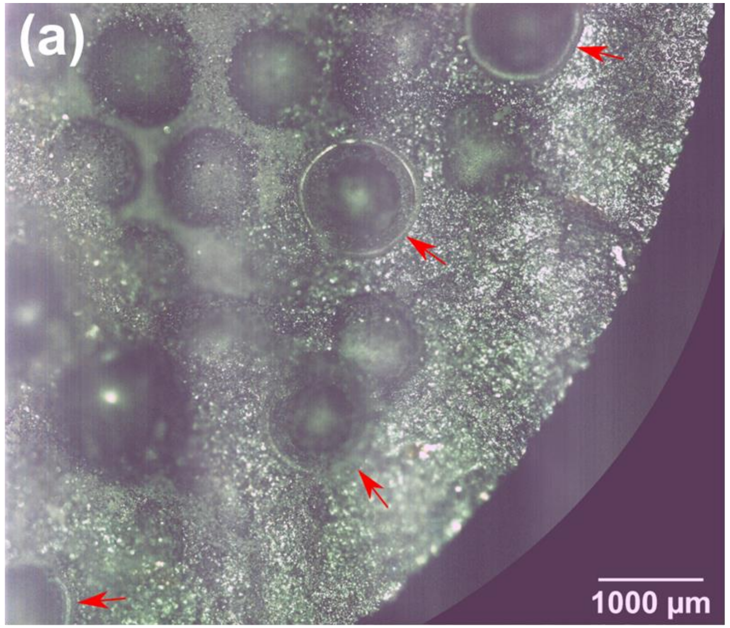
Provided by Dr. Will Cureton

Matrix and other interfaces have an important impact on fracture behavior



Cracks propagated through TRISO particles due to strong SiC/OPyC interface resulting from rough SiC TRISO layers

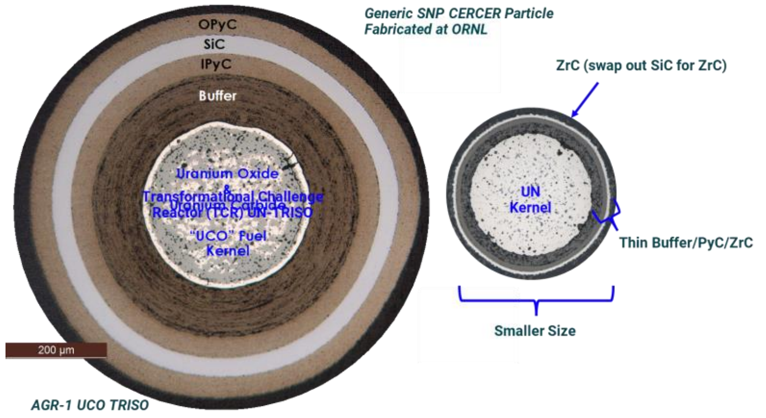




Fracture surface of compact C2 showing TRISO failures in the outer dense matrix region

Provided by Chris Petrie

SUMMARY

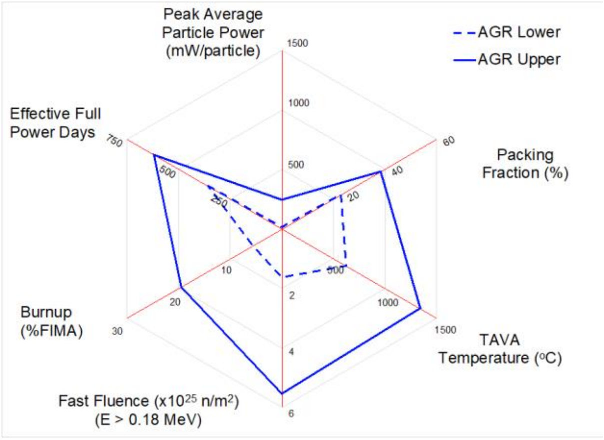


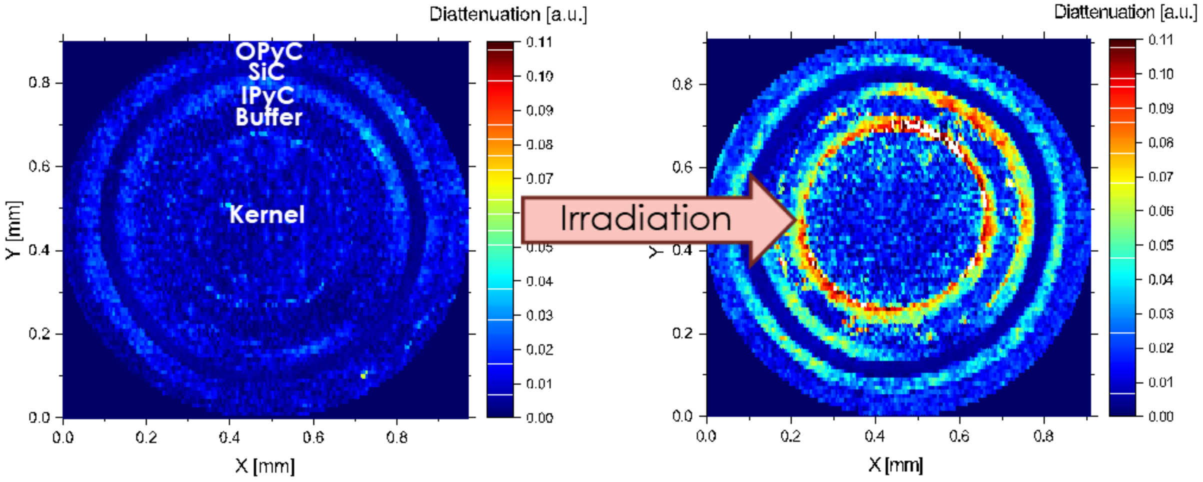
- TRISO particles and other coated particle fuels are being fabricated/proposed for an expanded operational envelope

- Despite the wealth of knowledge on TRISO fuel, we still have knowledge gaps that will impact its properties as SNF

- ORNL has the knowledge and infrastructure to support TRISO/coated particle fuel SNF studies

- **E-mail: honoratole@ornl.gov**





2024 Workshop on Storage and Transportation of TRISO and Metal Spent Nuclear Fuels, December 3-5

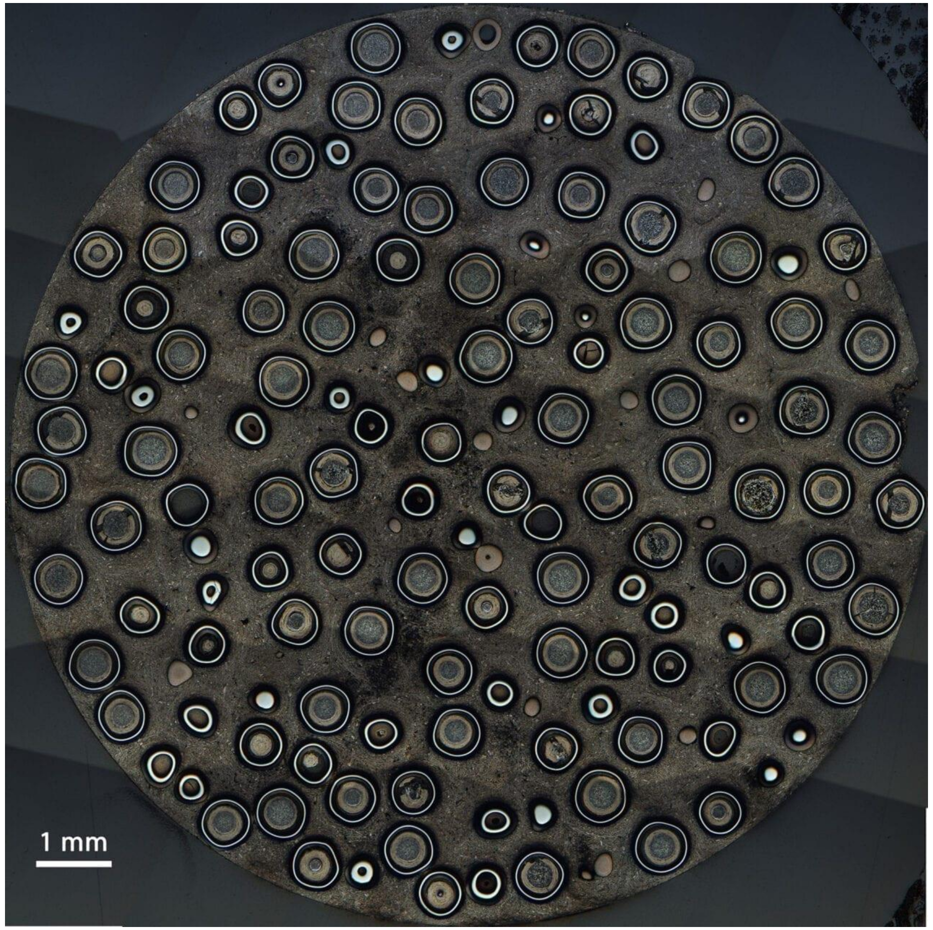
Tanner Mauseth

Post-Irradiation Examination (PIE) Research Scientist

Fracture Behavior Considerations for the TRISO Particle Matrix

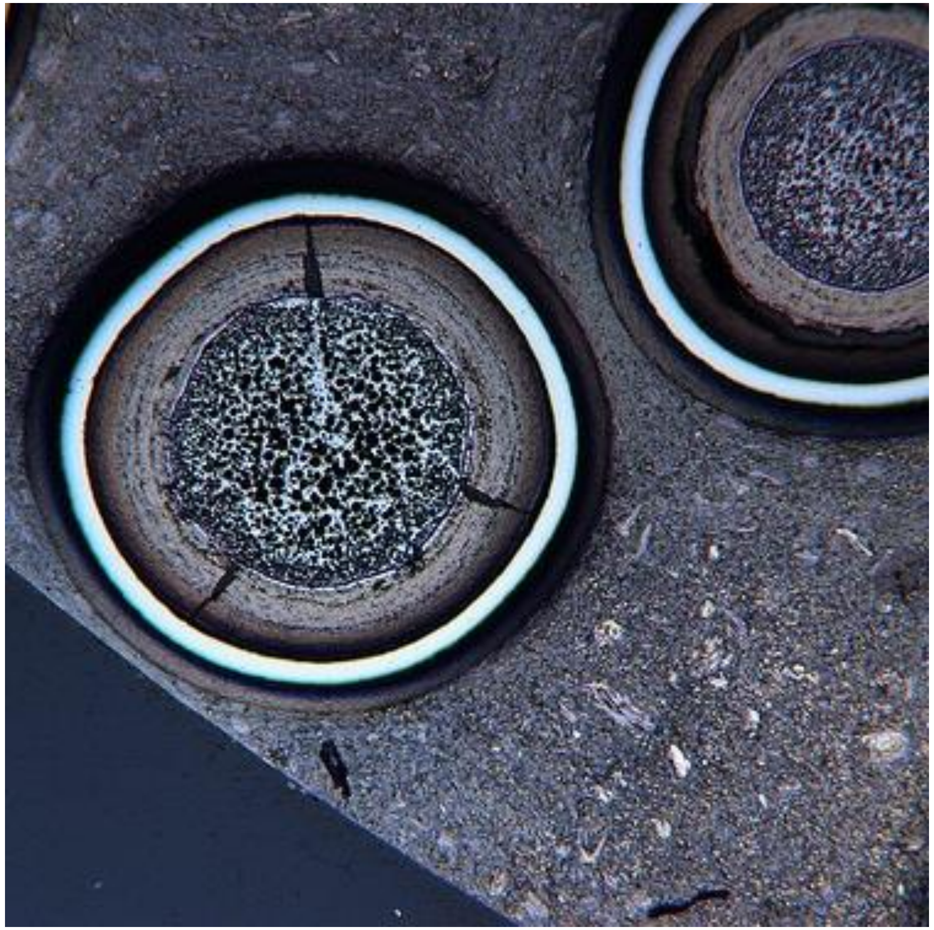
Introduction to Matrix Fracture Concerns

- **Matrix Role in TRISO Particles:**
  - The graphite or carbon composite matrix provides mechanical support and stability to TRISO particles, preventing interaction and damage, and maintaining the integrity of the fuel assembly under high temperatures and mechanical stresses.
  - The matrix acts as an additional barrier, capturing escaped fission products and reducing the risk of their release, ensuring the safe operation of the reactor and minimizing radioactive contamination.
- **Fracture Behavior:**
  - Matrix fracture refers to the cracking or breaking of the graphite or carbon composite matrix embedding TRISO particles, compromising mechanical integrity due to thermal stresses, irradiation-induced damage, or mechanical impacts.
  - Matrix fractures can lead to radiological releases, loss of containment, and compromised fuel integrity due to the release and migration of fission products and the physical displacement of TRISO particles.
- **Gaps in Understanding:**
  - While the matrix provides structural support and withstands mechanical stresses and irradiation, further research is needed to understand the long-term effects of these conditions and the detailed mechanisms leading to matrix fracture.
  - The matrix captures escaped fission products and aids in heat distribution, but additional studies are required to explore its interactions with fission products and the potential for alternative materials to enhance its performance and durability.



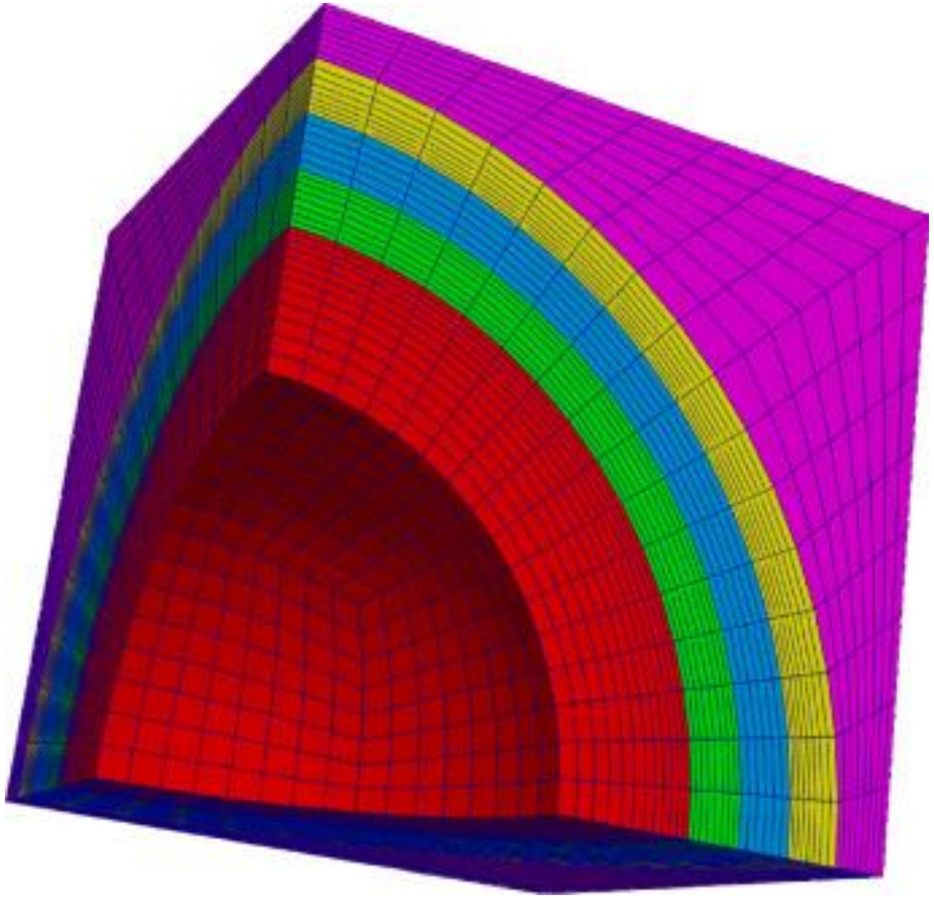
Key Considerations for Matrix Fracture Assessment

- **Micro-Tensile Strength:**
  - Evaluating the micro-tensile strength of the matrix is crucial for predicting fracture behavior and mechanical integrity, ensuring continued support and containment for TRISO particles under reactor conditions.
  - The micrometer-scale tensile strength characterization techniques developed by Mauseth (2023) for TRISO particle layers and interfaces can be applied to the TRISO particle matrix.
- **Fracture Toughness:**
  - Fracture toughness is essential for the TRISO particle matrix's ability to resist cracking under operational stresses, maintaining structural integrity and ensuring the stability and containment of TRISO particles.
  - Material composition, irradiation effects, thermal stresses, and mechanical stresses significantly impact the fracture toughness of the TRISO particle matrix, affecting its ability to resist crack propagation and remain durable under reactor conditions.
- **Irradiation Effects:**
  - Neutron irradiation significantly affects the mechanical properties of the TRISO particle matrix, causing defects, swelling, embrittlement, and changes in thermal conductivity and strength, which can compromise structural integrity.
  - Further research is needed to understand and mitigate irradiation impacts on matrix materials, focusing on long-term effects, microstructural changes, radiation-induced creep, material optimization, and interactions with fission products.



Current Data and Modeling Capabilities

- **Existing Data:**
  - The properties of graphite and carbonaceous matrix materials embedding TRISO particles, including density, porosity, thermal conductivity, and tensile strength (20-40 MPa), are well-documented. These materials exhibit increased brittleness and reduced tensile strength under stress due to irradiation effects.
  - Further research is needed to understand long-term irradiation effects, microstructural changes, radiation-induced creep, material optimization, and interactions with fission products in TRISO matrices, including mechanisms of degradation and defect evolution.
- **Modeling Fractures:**
  - Modeling fractures under both normal and off-normal conditions is crucial for ensuring the safety, reliability, and containment integrity of TRISO fuel, aiding in predictive maintenance and accident assessment.
  - Challenges in modeling fractures in TRISO matrices include incomplete long-term data on cumulative damage, limited understanding of irradiation-induced microstructural changes, scarce studies on radiation-induced creep, variability in advanced materials, and insufficient data on interactions with fission products, all adding uncertainties to predictive models.
- **Validation Needs:**
  - Comprehensive and validated data are essential for accurate predictive models to ensure TRISO fuel's safety, reliability, and performance, meet regulatory standards, mitigate risks, and support material optimization and innovation.
  - Data validation for TRISO matrices requires long-term irradiation experiments, microstructural analysis, benchmarking, inter-laboratory studies, advanced simulations, standardized protocols, data sharing, and mechanical and thermal testing.



Discussion Points and Questions

- Research Needs:
  - What additional research is needed to better understand matrix fracture behavior?
- Data Sufficiency:
  - Is current data sufficient to model matrix fracture under various conditions?
- Design Optimization:
  - How can design features be optimized to mitigate matrix fracture?
- Collaboration:
  - How can industry and regulators collaborate to ensure safety and compliance?

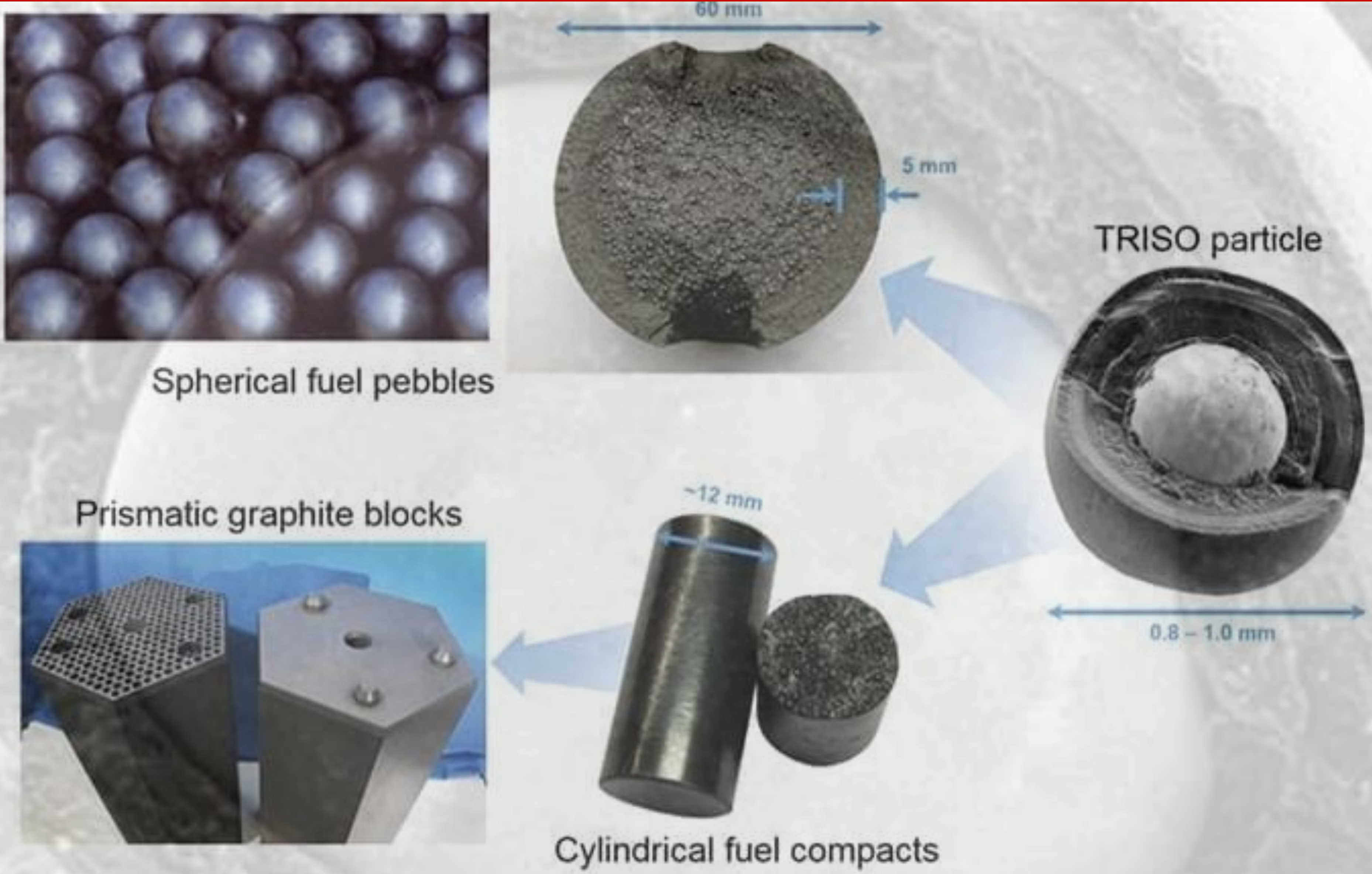
Modeling of TRISO and Matrix Fracture

Wen Jiang [a,b]

[a] Department of Nuclear Engineering, NC State University

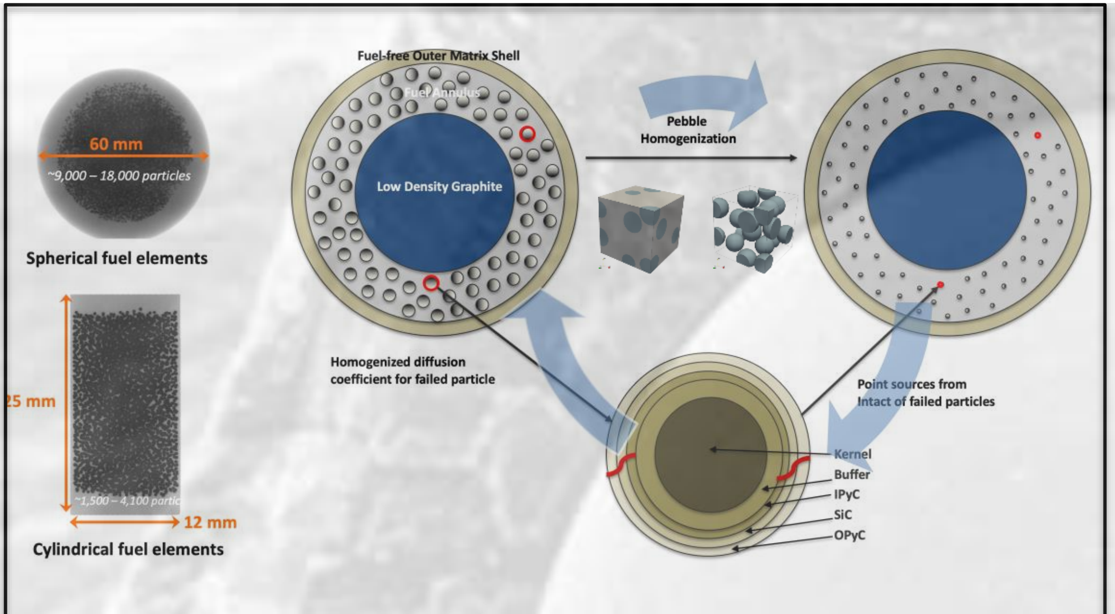
[b] Idaho National Laboratory

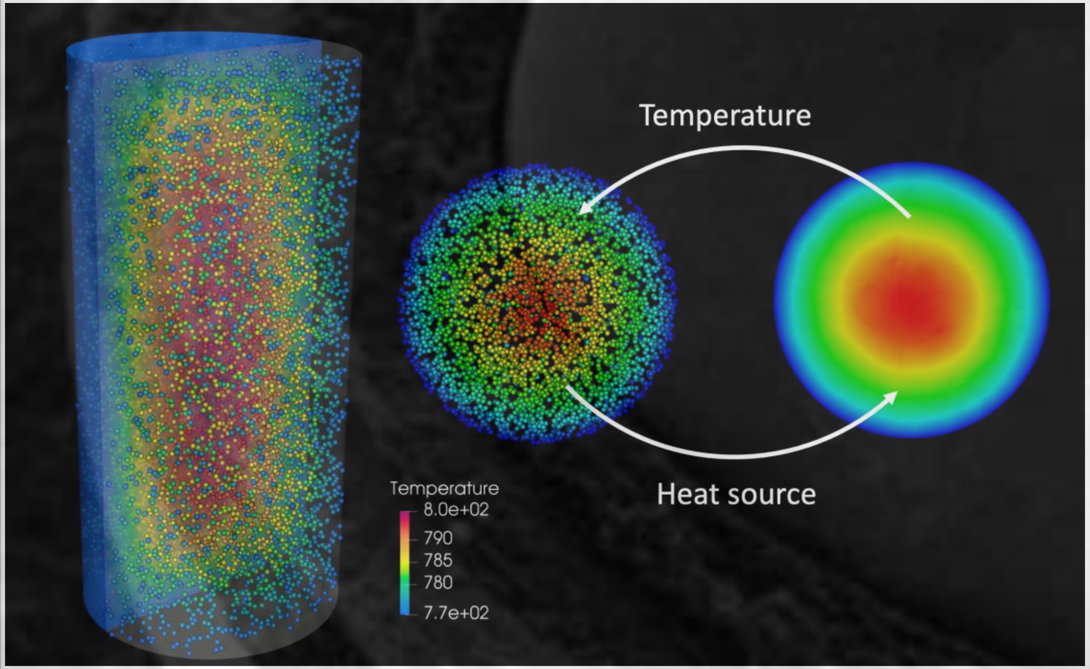
TRISO Fuel Element

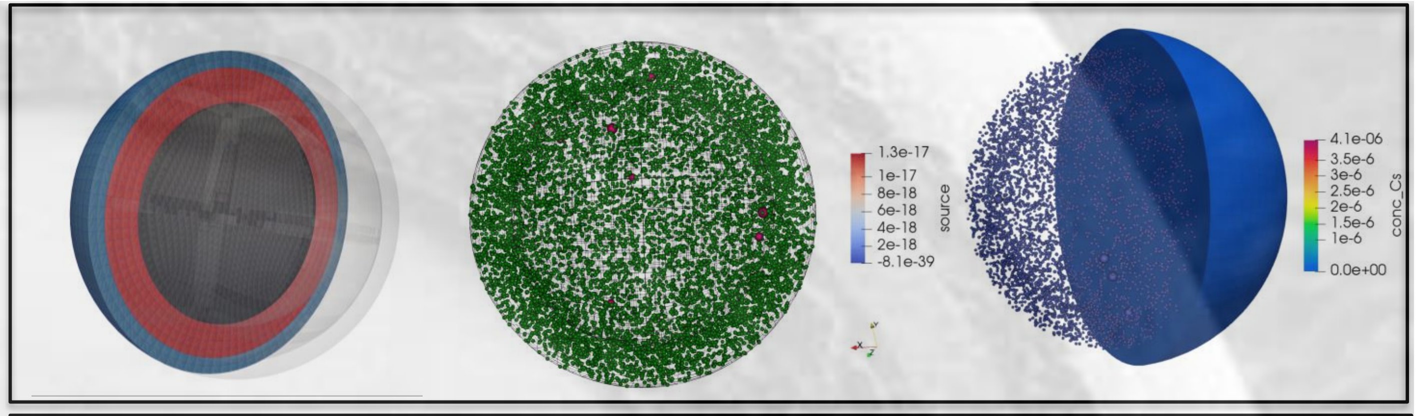


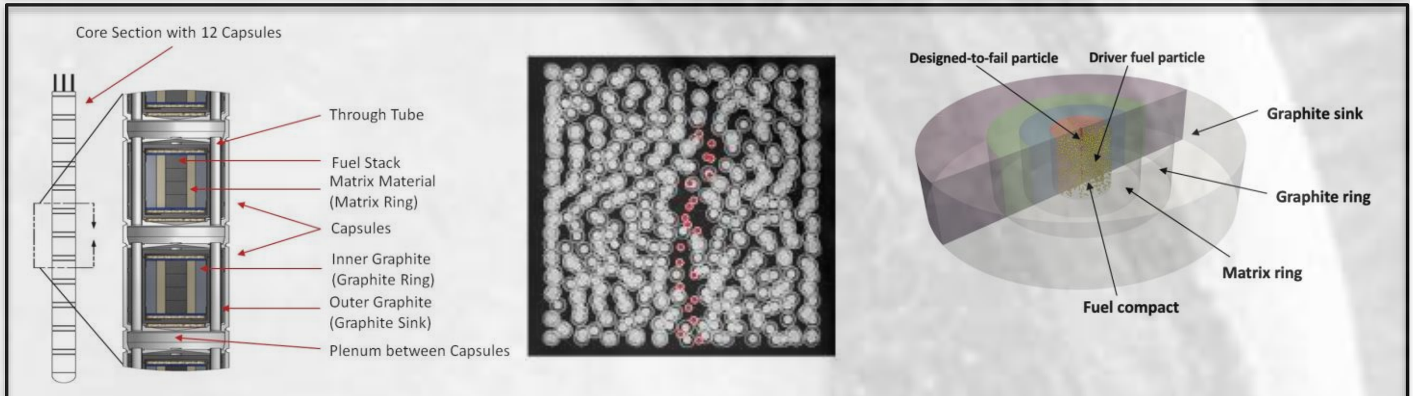
A large majority of failure mode analysis are focused on the TRISO particle itself. However, there is a lack of studies investigating the interaction between the matrix and the embedded TRISO particles.

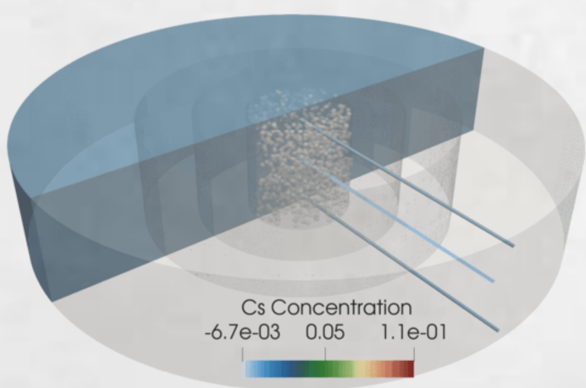
FP diffusion through fuel element



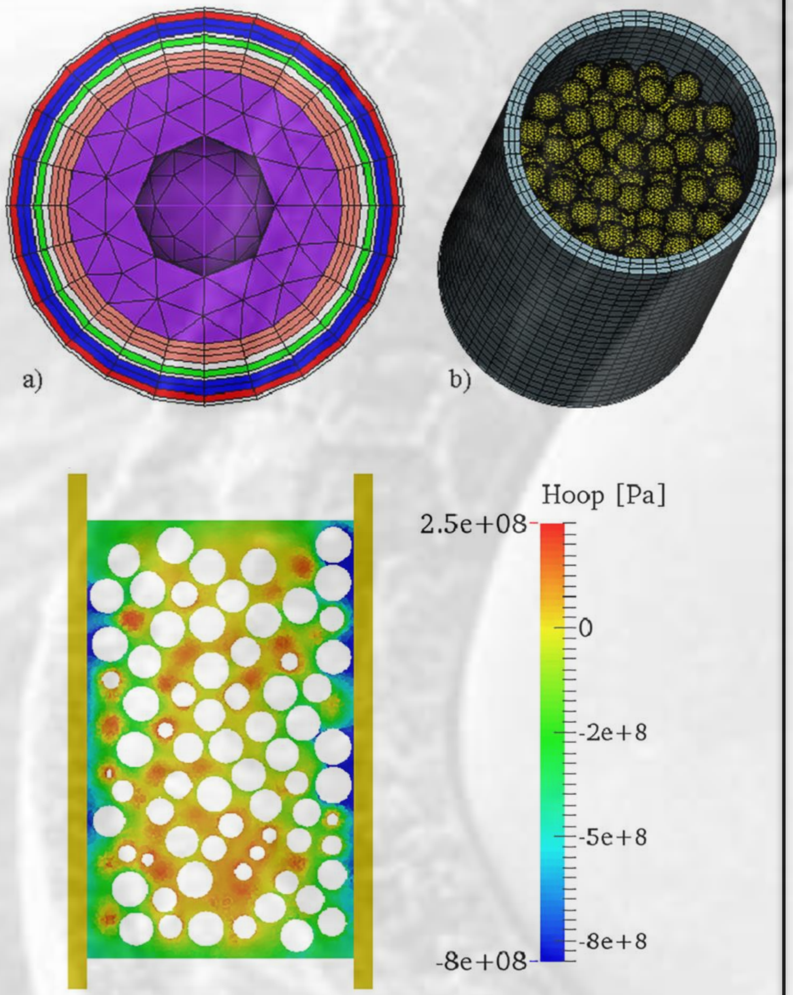




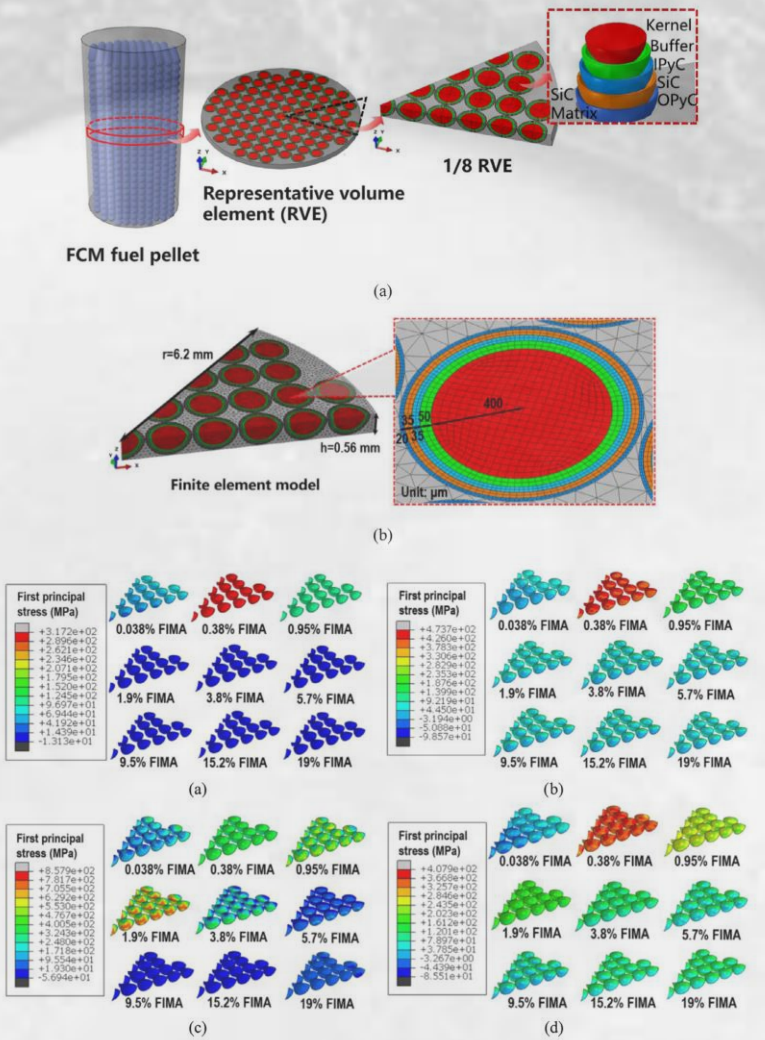




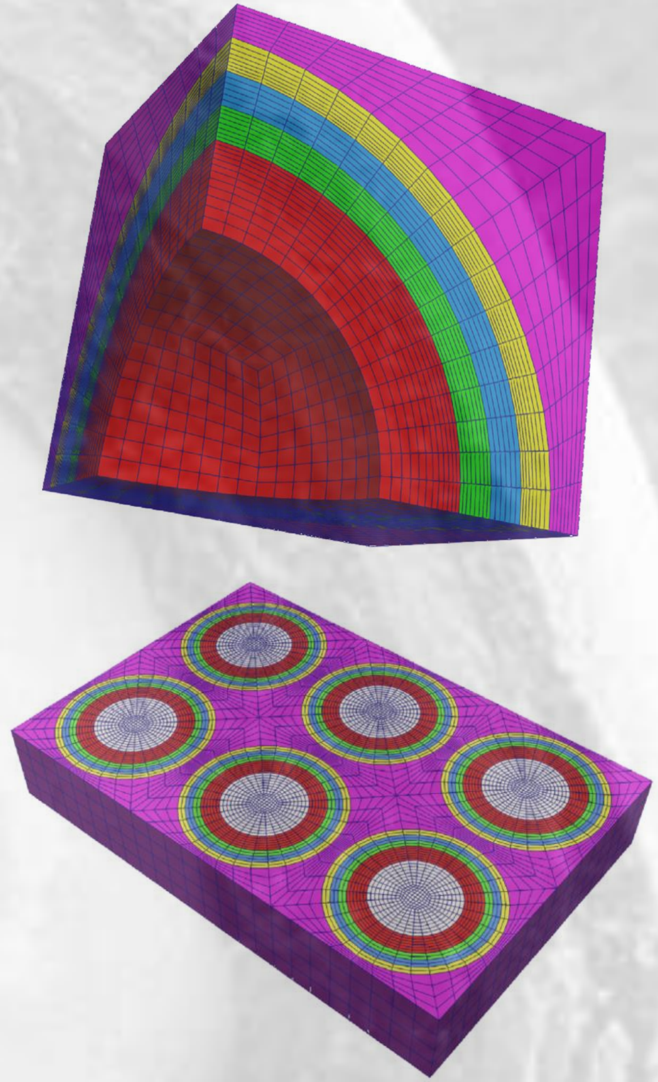
Thermo-mechanical modeling



Schappel, Danny, Nicholas R. Brown, and Kurt A. Terrani. "Modeling reactivity insertion experiments of TRISO particles in NSRR using BISON." Journal of Nuclear Materials 530 (2020): 151965.



Wei, Hongyang, et al. "Modeling of irradiation-induced thermo-mechanical coupling and multi-scale behavior in a fully ceramic-microencapsulated fuel pellet." *Journal of Nuclear Materials* 544 (2021): 152673.



Hales, Jason D., and **Wen Jiang**. "Versatile TRISO fuel particle modeling in Bison." *Nuclear Engineering and Design* 428 (2024): 113515.

Matrix Structural Integrity and particle interaction

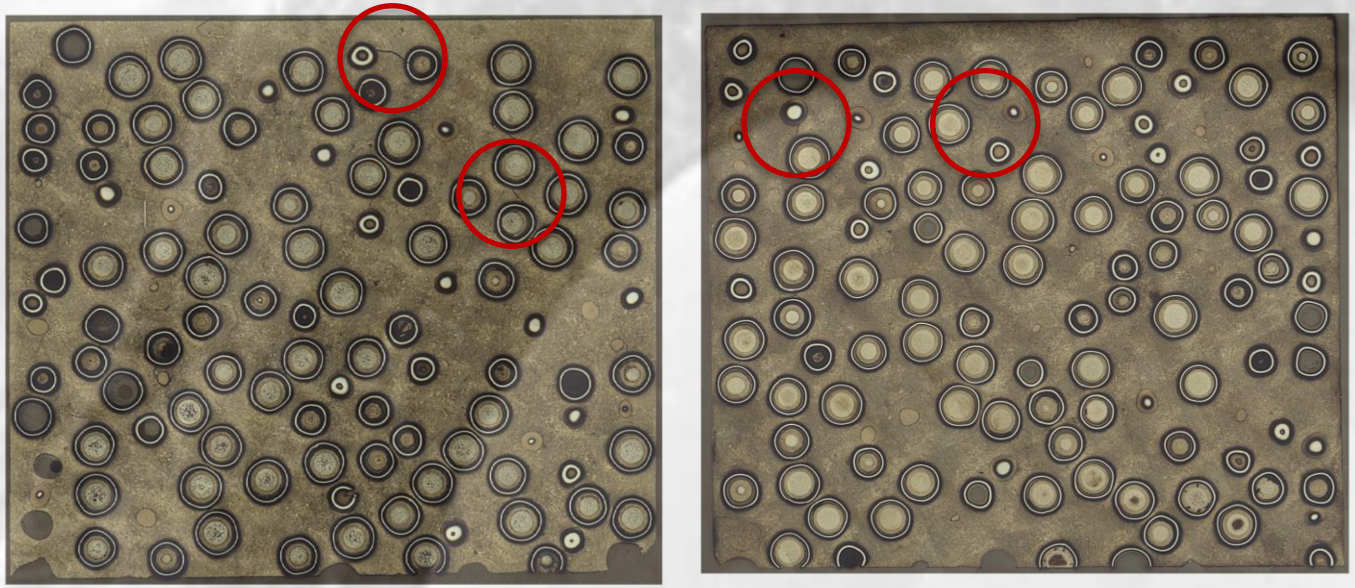
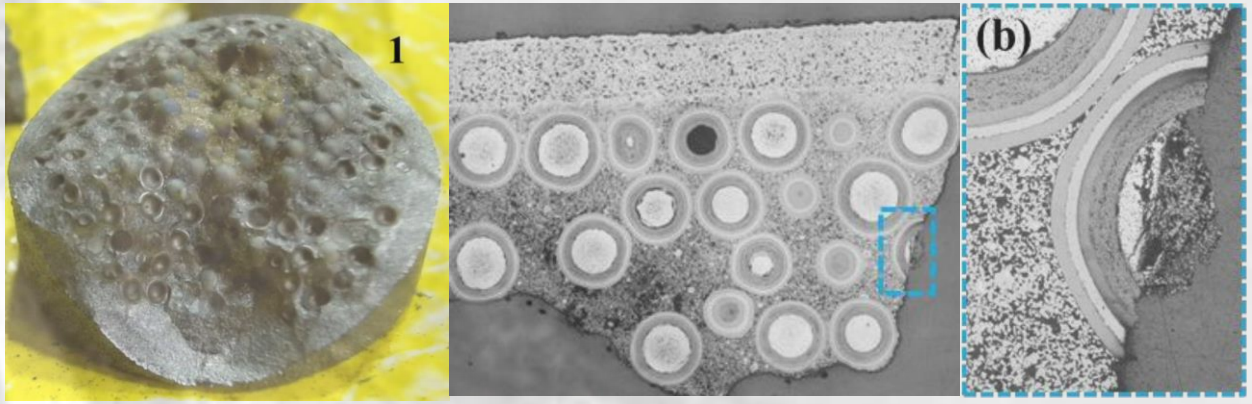


Figure A-5. UCO Compact 2-4-3 top, MNT 60X micrograph montage.

AGR Compact Graphite



TCR/FCM SiC Matrix

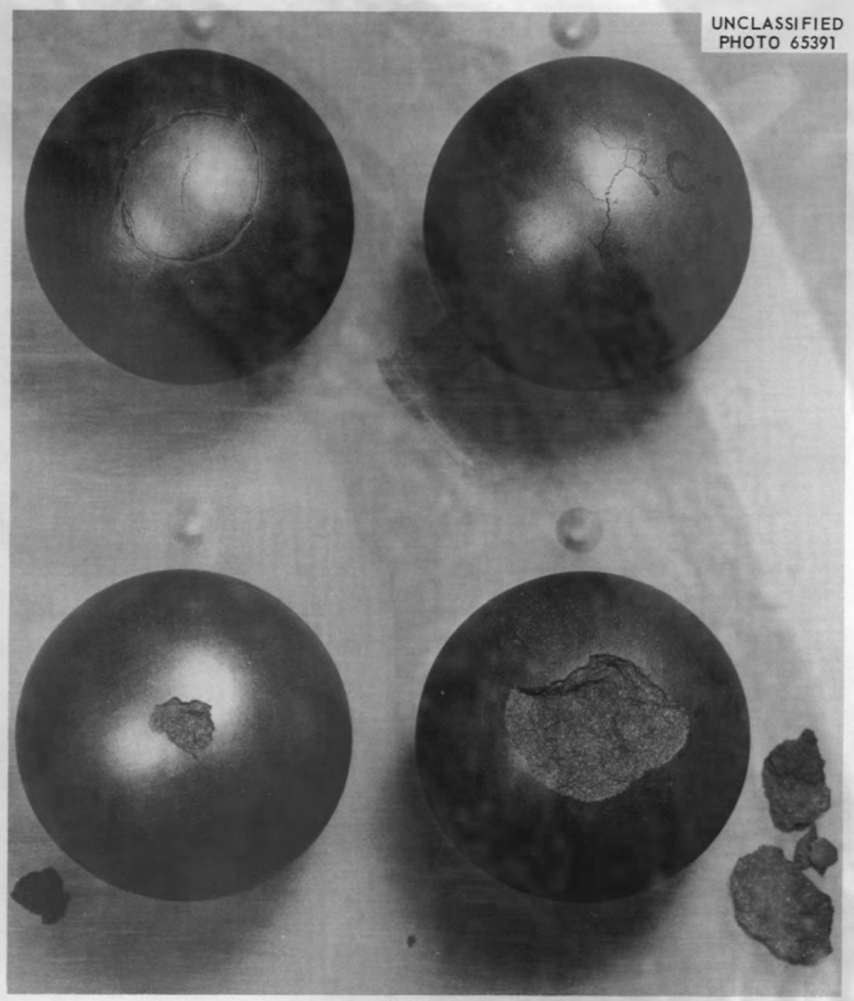


Fig. 37. Typical Failures in 6-cm-diam ATJ Graphite Spheres in Drop-Weight Tests with 18.07 ft-lb of Kinetic Energy in Weight.

Drop testing

Knowledge gaps and challenges

- Fission product diffusion through matrix
  - AGR 3/4: the model calibration is complicated.
- Matrix structural integrity
  - Static – thermal-mechanical: computational cost is high
  - Dynamic – drop and impact testing: rely on empirical data, and mechanisms are not well understood.
- Matrix and particles interaction
  - Stress concentration
  - Competing mechanisms:
    - Matrix and OPyC debonding
    - Crack propagation from matrix into particles
- What is long term behavior of the fuel matrix under storage conditions?
  - Oxidation? Degradation/damage?

December 3-5, 2024

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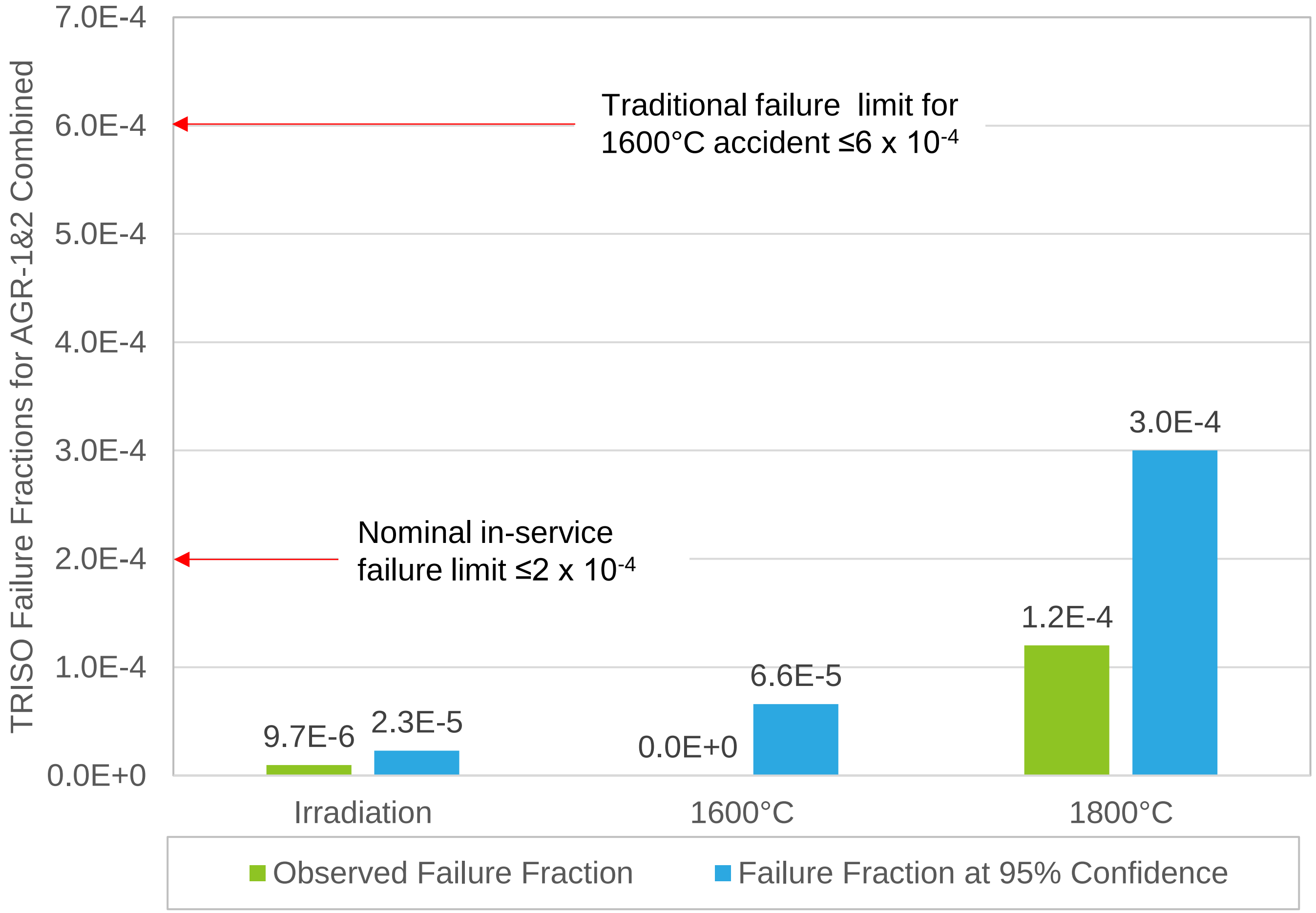
TRISO Particle Fracture

Importance of strong matrix and careful handling

2024 Workshop on Storage and Transportation of TRISO and Metal Spent Nuclear Fuels

What TRISO failure rates do we currently know?

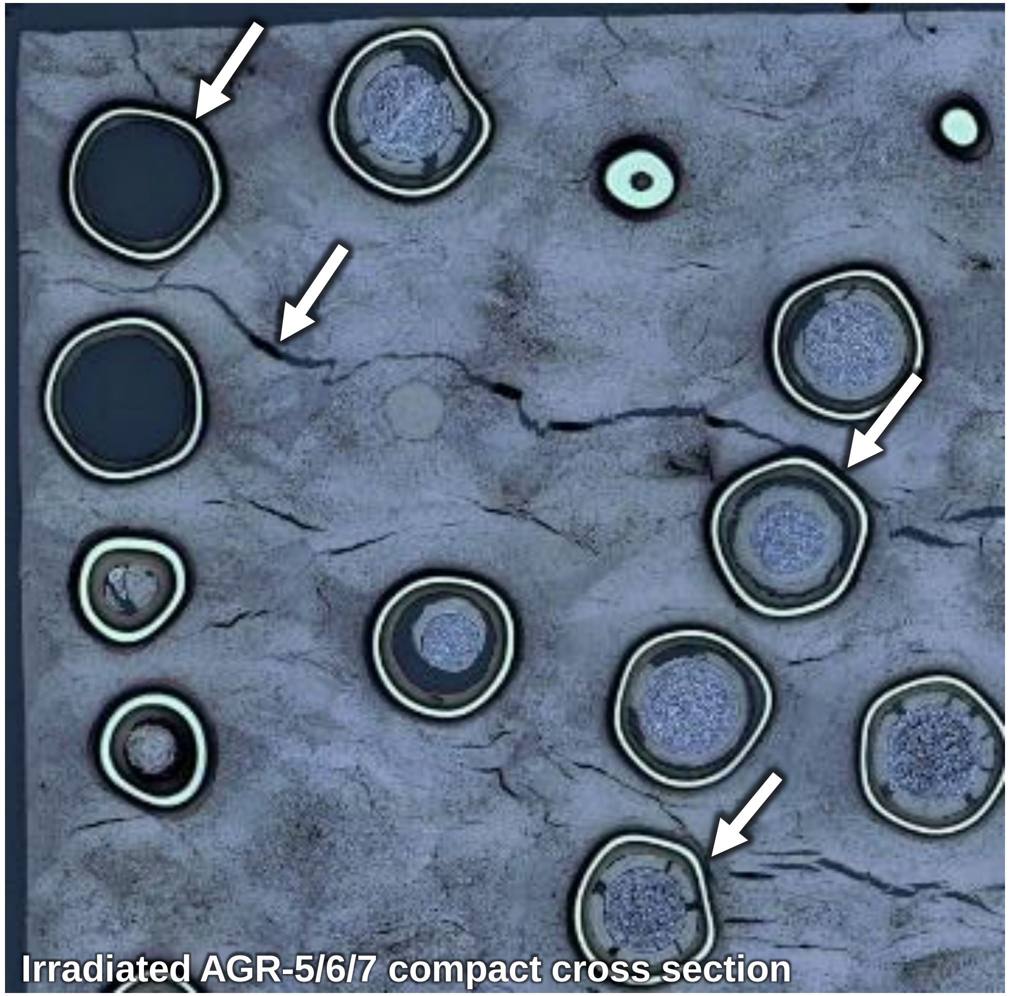
- Combined, AGR-1 and AGR-2 had an observed in-pile TRISO failure fraction of ≤9.67E-6 [1]
- Failure rate increases modestly during 300-hour long inert safety tests (more severe than reality)
- To date, there have been no signs of irradiated AGR particles being broken during harsh pneumatic sample transfers or shipments. AGR-1+2 involved 108 compacts and >400,000 particles.



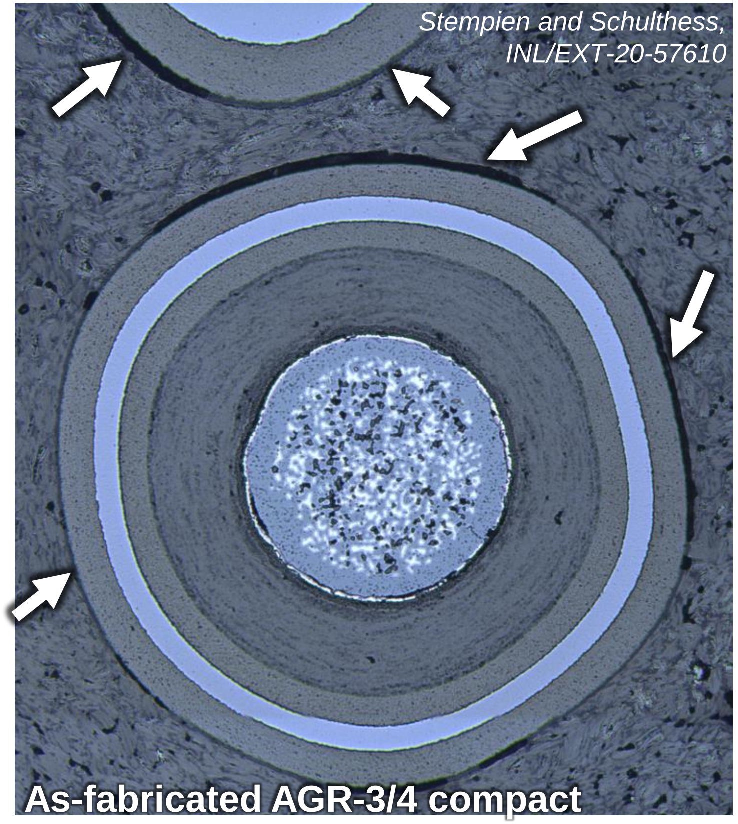
1. *Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance: Topical Report EPRI-AR-1(NP).* EPRI, Palo Alto, CA: 2019. 3002015750.

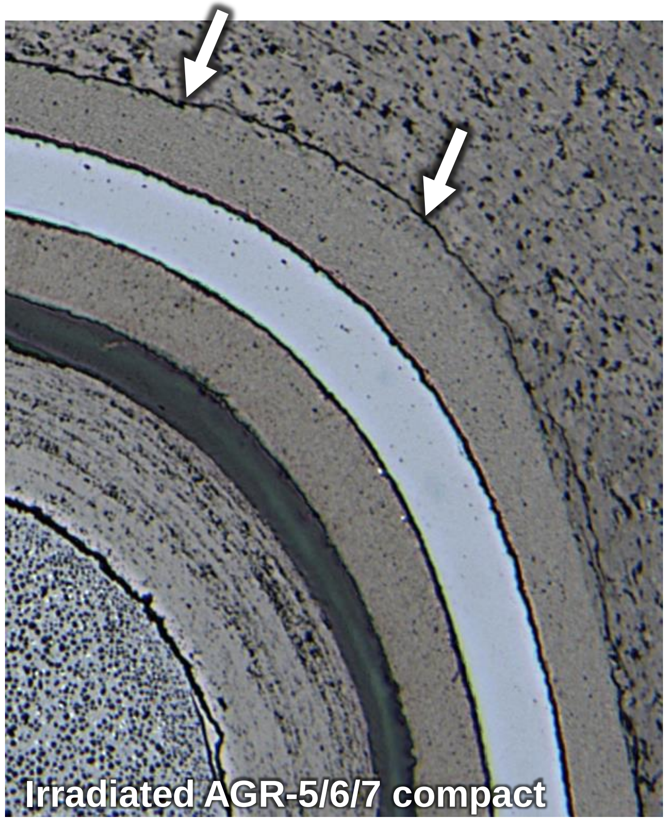
Graphitic matrix and particle OPyC may not be strongly bonded. This allows matrix stresses to be directed around the particles.

- AGR experience: as-fabricated matrix fissures go around particles



- Thin matrix-OPyC gaps may form at time of fabrication around portions of particles. This prevents transmission of stress from matrix to the particles. A different matrix, such as SiC, may behave differently [1, 2]!

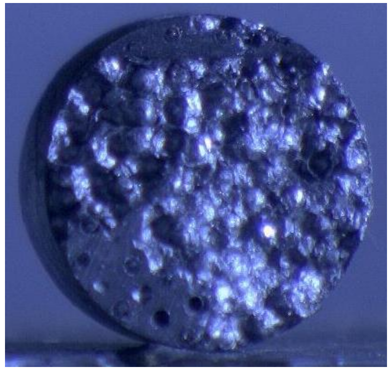
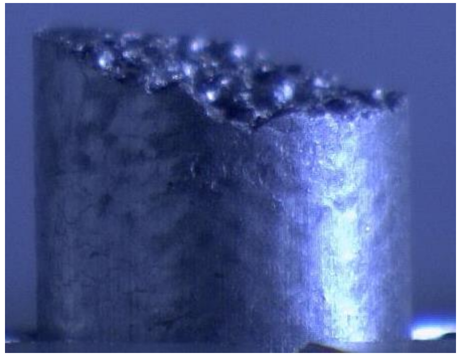




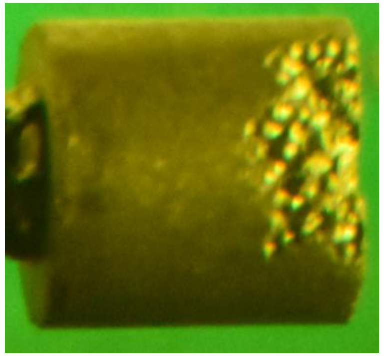
1. Petrie et al. 2023. J. Nucl. Mater, Vol 580, 155419.
2. Schappel et al. 2023. J Nucl. Mater. Vol 586, 154691.

Accidental compact PIE damage shows that not all exposed particles will be broken. Particles away from the fractures will be unaffected.

- Compact accidentally broken by a power tool during irradiation test train disassembly. Estimated 3-6% of all particles exposed. Not all of those exposed will be broken. The other 94-97% are intact and will retain fission products as usual.

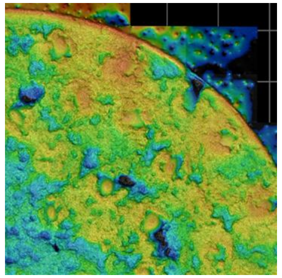



- AGR-3/4 Compact 7-3 was accidentally chipped. Chemical analysis showed that none of the ~30 exposed particles was broken! It is possible some particles could have been broken in the missing piece that was not recovered, however.



Stempien, J.D. and L. Cai. 2024. INL/RPT-24-77357.

- Particles at the surface with little or no matrix cover are the most vulnerable during normal handling

Seemingly harsh PIE does not typically challenge particle integrity. Consider the main steps of the deconsolidation-leach-burn-leach (DLBL) process.

1. Electrolytic deconsolidation (5 watts DC) to break up compact matrix in 6-8 M nitric acid
2. Loose TRISO particles recovered from deconsolidation are then leached for 48 h in near-boiling 16 M nitric acid
3. Then the particles are oxidized in air at 750°C for 72 h to remove the OPyC layer
4. The exposed-SiC particles are then leached for 48 h in near-boiling 16 M nitric acid
5. Irradiated particles survive these processes with rare exception

Idaho National Laboratory

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2024 Workshop on Storage and Transportation of TRISO and Metal Spent Nuclear Fuels, December 3-5

Tanner Mauseth

Post-Irradiation Examination (PIE) Research Scientist

Micro-Tensile Properties of Irradiated AGR-2 TRISO Fuel Pyrolytic Carbon (PyC) and Silicon Carbide (SiC) Coatings

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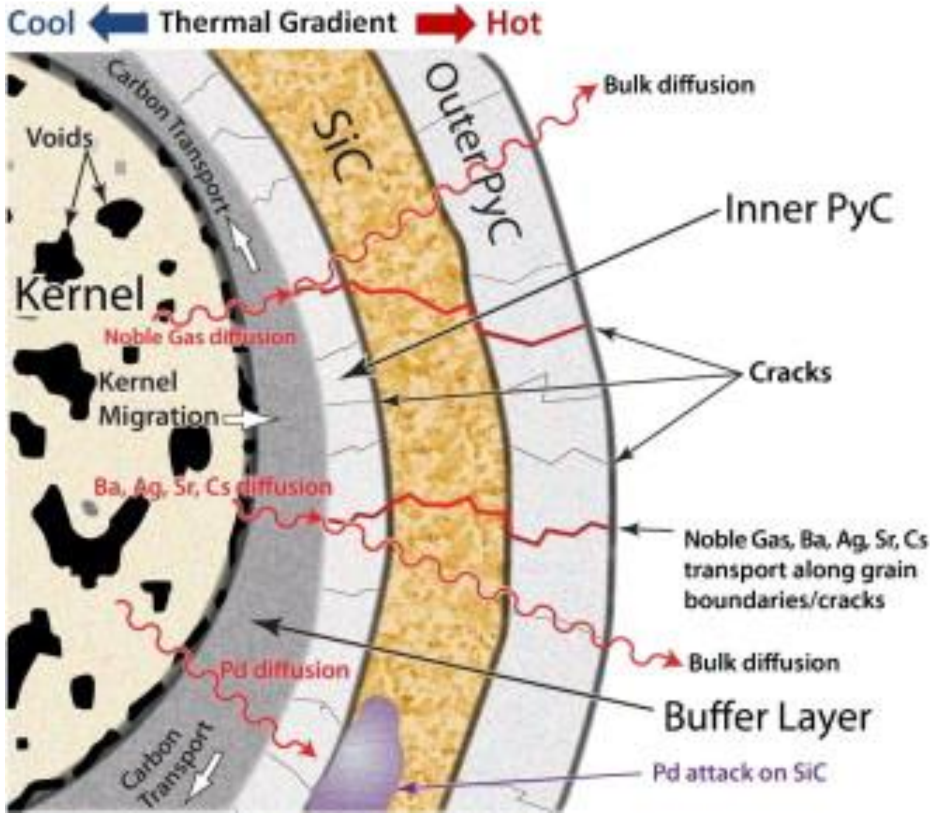
Idaho National Laboratory

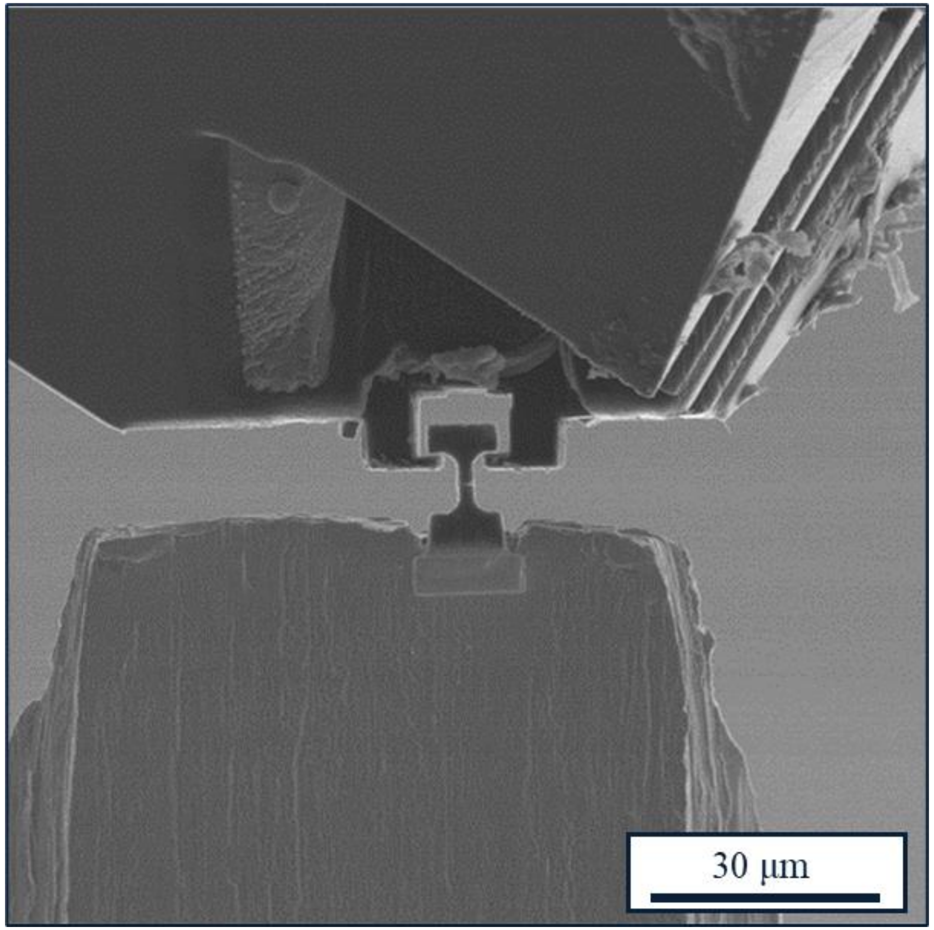
Introduction and PyC Creep Overview

- **Role of PyC and SiC Layers:**
  - The inner and outer PyC layers primarily provide structural support to the SiC layer and help contain fission gases.
  - The SiC layer is crucial for retaining fission products, providing mechanical strength, and ensuring chemical stability under high-temperature and irradiation conditions.
- **PyC Creep Phenomenon:**
  - PyC creep, which is the deformation of pyrolytic carbon layers under irradiation due to stress and high temperatures, can exert additional stresses on the SiC layer, potentially causing cracking and structural degradation that compromise the TRISO particle's ability to retain fission products.
  - However, this failure mode has not been observed in the AGR program; SiC failures are more likely due to fission product attacks when IPyC cracking exposes the intact SiC layer.



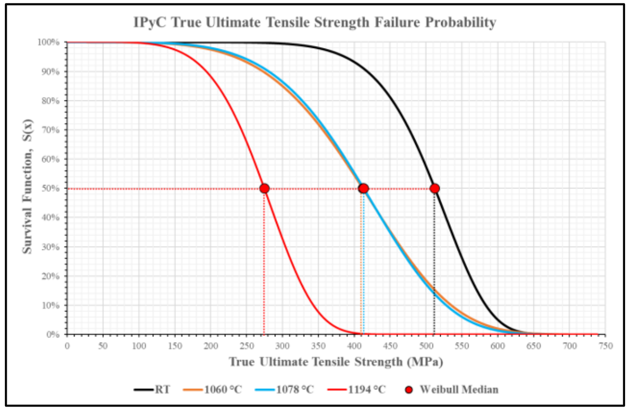
Micro-Tensile Testing Methodology

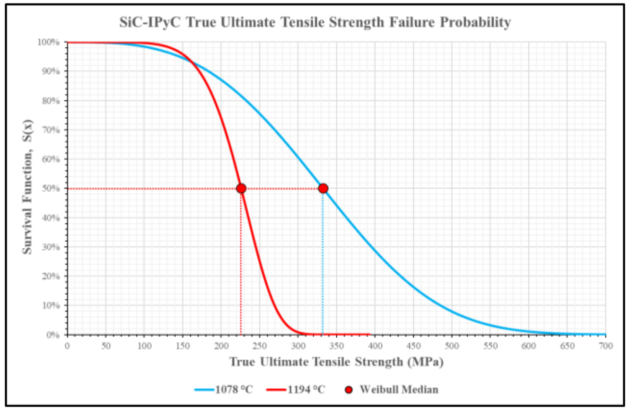
- **Testing Approach:**
  - A Bruker Hysitron PI 88 SEM PicoIndenter, typically used for nanoindentation, was retrofitted with a diamond gripper for in situ micro-tensile testing in conjunction with an SEM.
- **Sample Preparation:**
  - The gallium FEI Quanta 3D Dual Beam and Thermo G3 Plasma Dual Beam FIB SEM systems at IMCL were used to fabricate micro-tensile samples, following the study by Mauseth et al. (2023) with specific modifications for the FIB instruments used.
- **Relevance to SiC Layer Failure:**
  - Micro-tensile testing provides critical insights into the mechanical properties and failure mechanisms of the SiC layer in TRISO fuel particles, including the effects of PyC creep and operational stresses, thereby enhancing our understanding of SiC integrity and performance under realistic conditions.
  - This testing assesses the SiC layer's structural integrity, informs predictive models, enhances safety and performance strategies, and guides improvements in TRISO fuel design, ensuring the reliability and longevity of TRISO fuel particles.

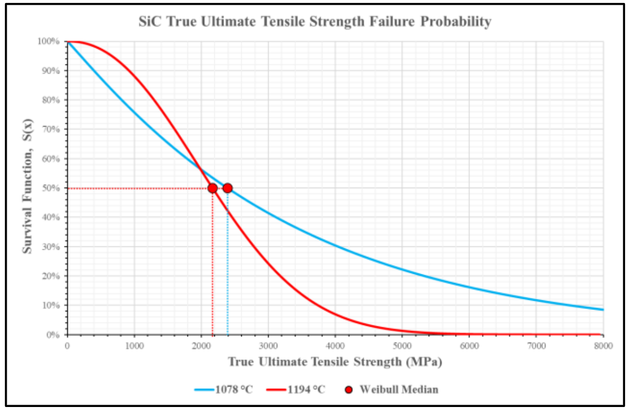


Key Findings from Micro-Tensile Testing

- **Tensile Properties:**
  - After irradiation, both IPyC and SiC show a decrease in tensile strength with increasing Time Average Volume Average (TAVA) temperature.
  - The SiC-IPyC interface exhibits lower tensile strength than the individual layers, indicating the interface as a potential weak point.
- **Implications for TRISO Fuel:**
  - The decrease in tensile strength of IPyC and SiC with increasing TAVA, particularly at the SiC-IPyC interface, suggests potential weak points in TRISO fuel particles that could lead to delamination or cracking under operational stresses, compromising the containment of fission products.
  - Even at low temperatures and with minimal neutron exposure, these weak points could compromise TRISO fuel particle integrity, stability, and fission product containment during storage and transportation due to mechanical stresses and long-term material degradation.







Discussion Points and Questions

- Research Needs:
  - What additional research is needed to better understand PyC creep and SiC layer failure?
- Data Sufficiency :
  - Are current data sufficient to model PyC creep and SiC layer failure during normal and off-normal conditions?
- Design Optimization:
  - How can design features be optimized to prevent SiC layer failure?
- Collaboration:
  - How can industry and regulators collaborate to ensure safety and compliance?

Oxidation Behavior of the SiC Coating of TRISO Fuel Particles in Air

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Oxidation of SiC in oxygen

- **Two oxidation mechanisms**
  - **Passive oxidation**: at low temperature and high oxygen partial pressure

$$\boldsymbol{SiC_{(s)} + \frac{3}{2}O_{2(g)} \rightarrow SiO_{2(s)} + CO_{(g)}}$$

- Deal-Grove oxidation behavior: $x^2 + Ax = Bt$

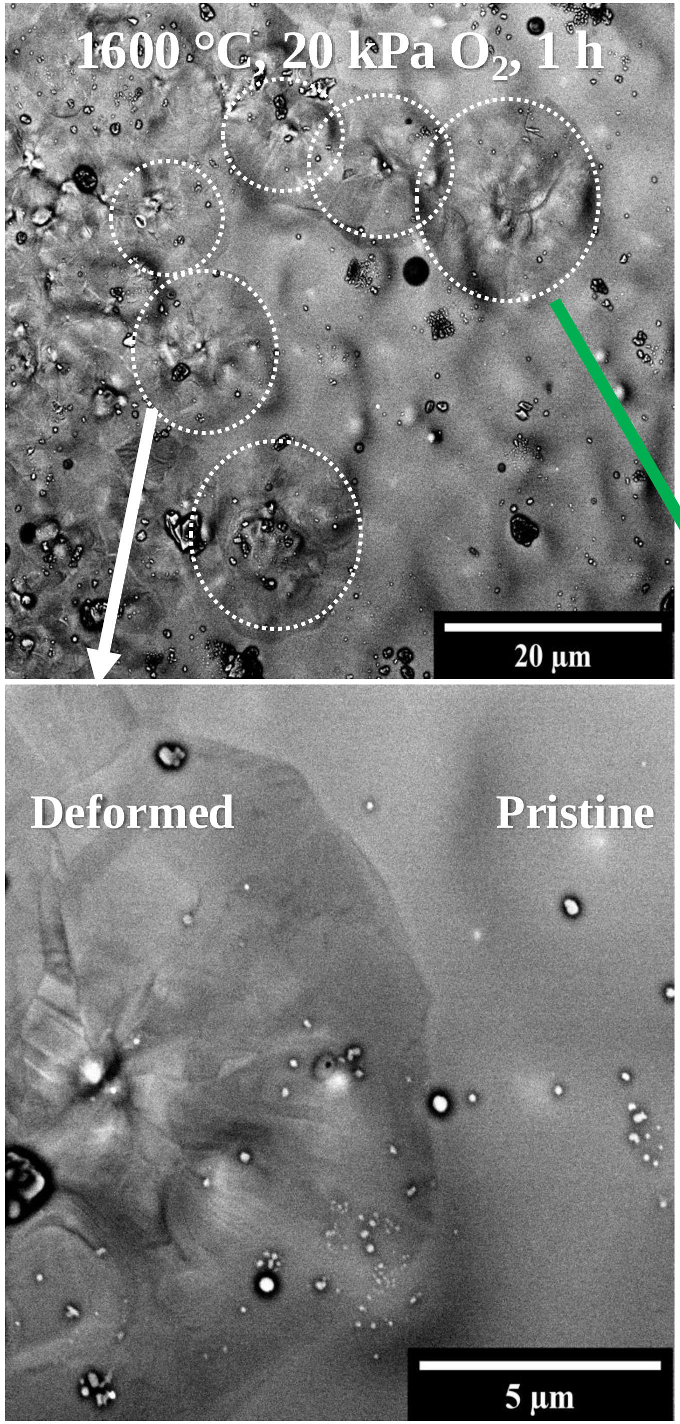
Where x=oxide layer thickness, B = parabolic rate constant, B/A = linear rate constant

- Arrhenius dependence: $B = B_0 \exp\left(\frac{-Q}{RT}\right)$

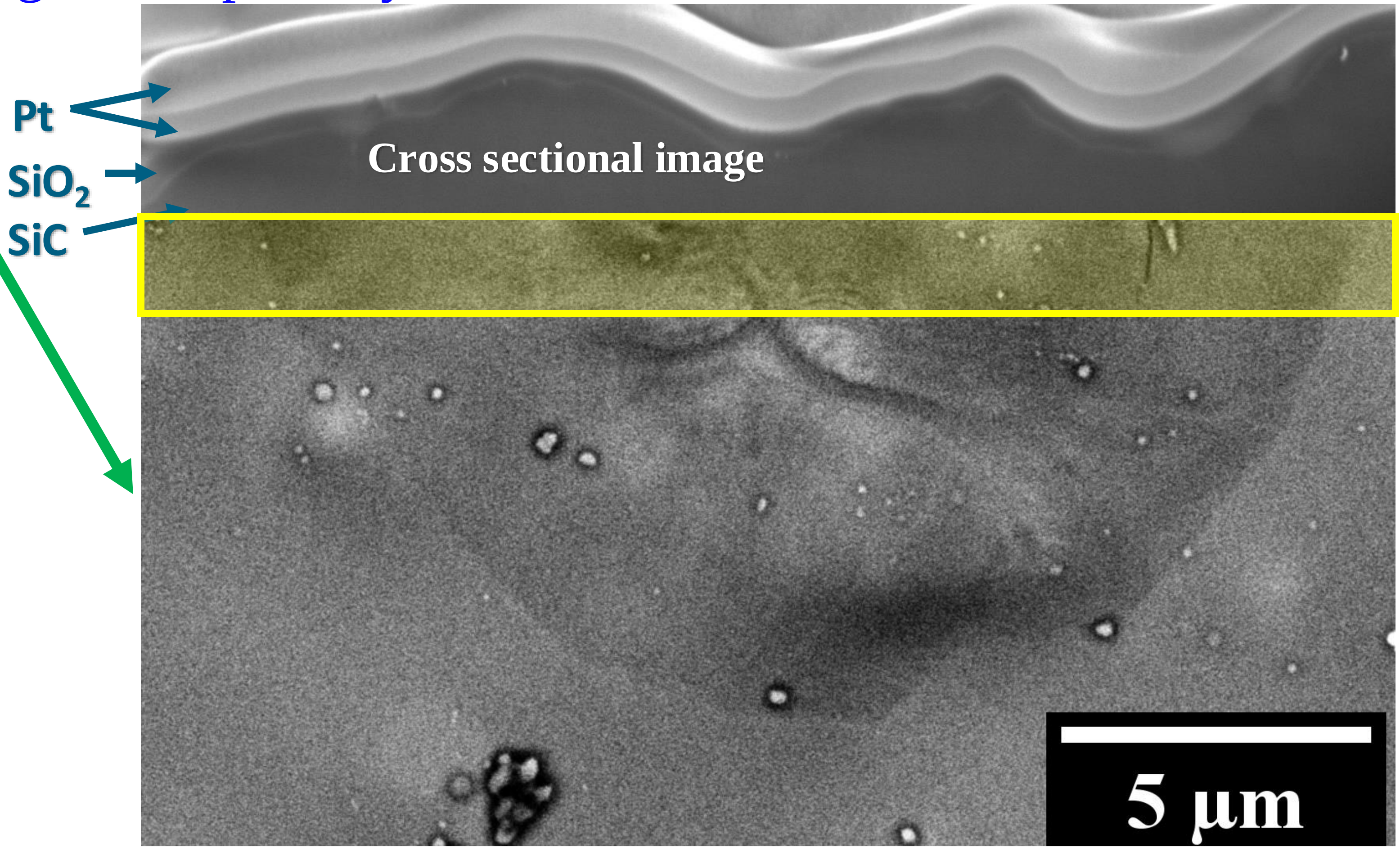
  - **Active oxidation**: at high temperature and low oxygen partial pressure

$$\boldsymbol{SiC_{(s)} + O_{2(g)} \rightarrow SiO_{(g)} + CO_{(g)}}$$

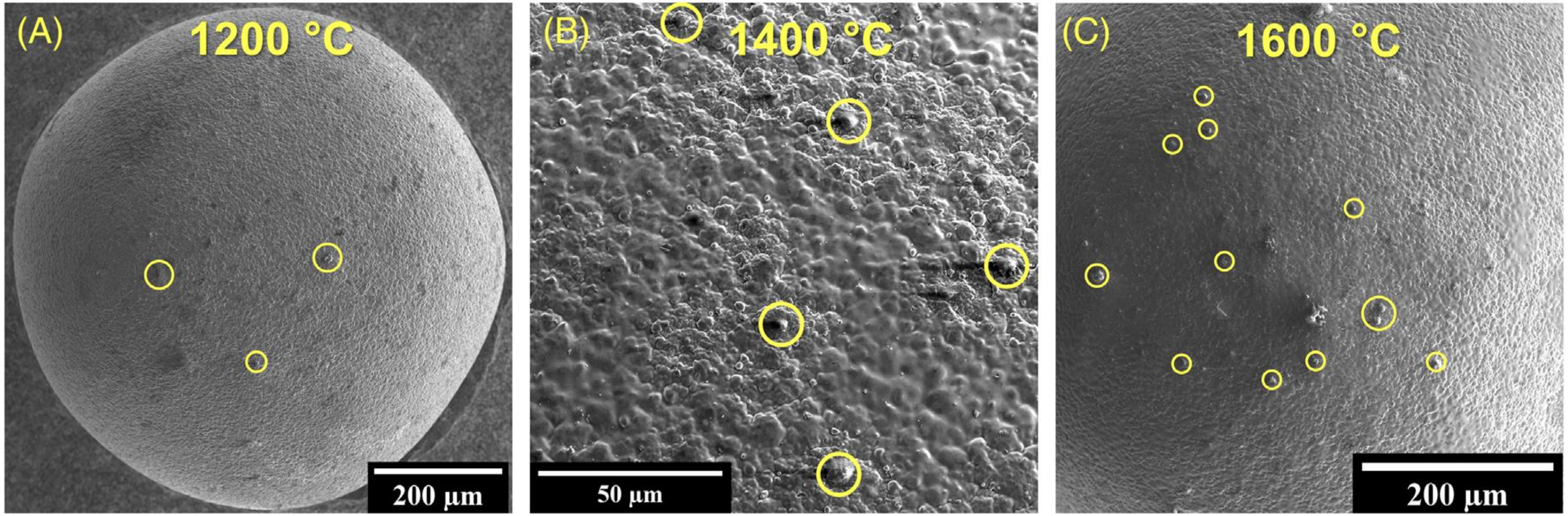
Oxidation in Flowing 20 kPa O_2 at 1600 °C



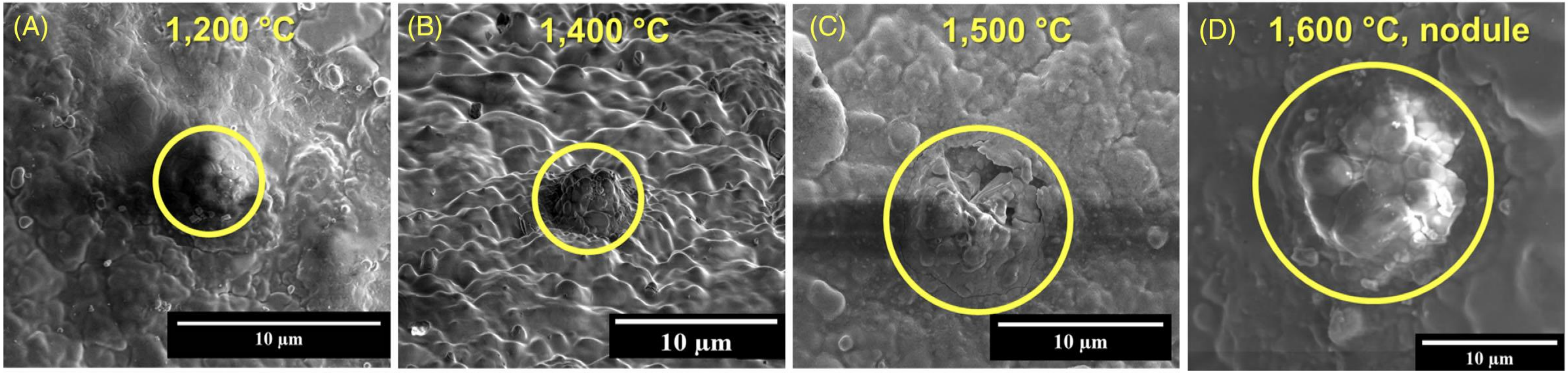
- Two regions: deformed (spherulitic) and pristine (amorphous)
- Spherulitic cracks from devitrification of SiO_2
- No significant variations in oxide thickness across the spherulitic region; no porosity



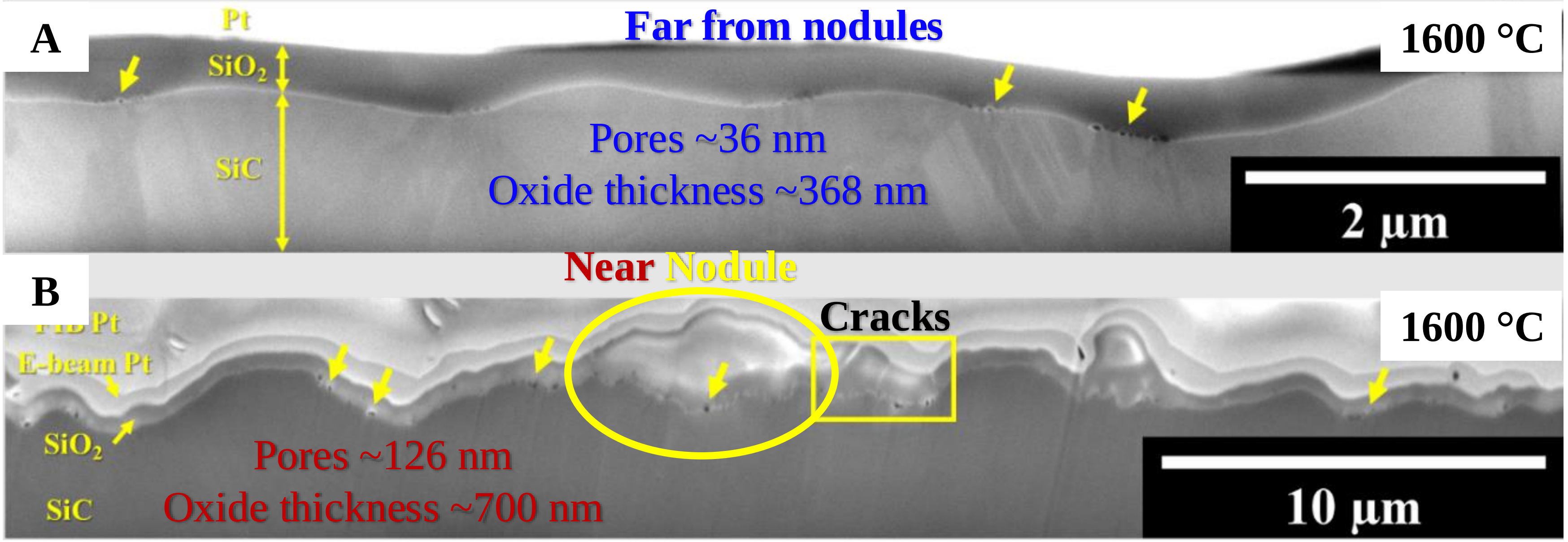
Oxidation in 0.2 kPa O_2 – surface morphology



- Raised nodules on surface
- Number density increase with oxidation temperature

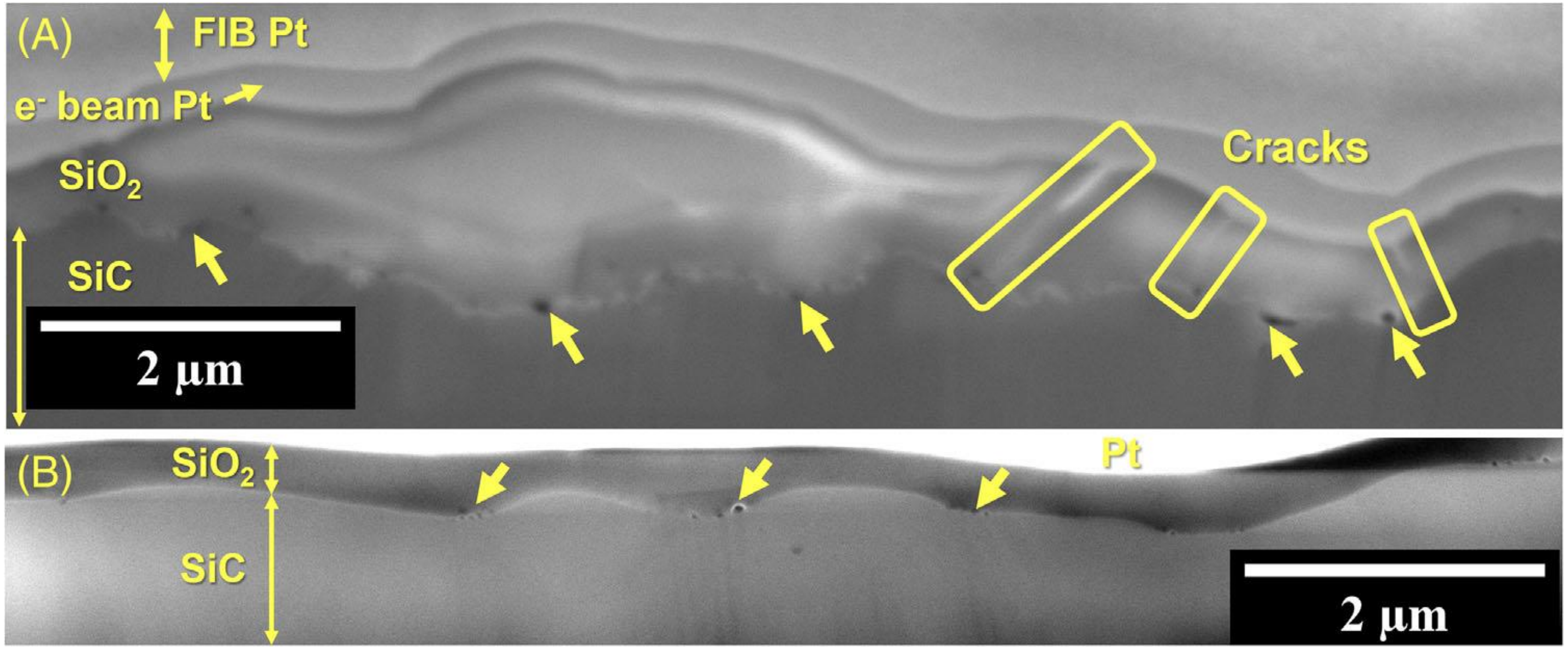


Oxidation in 0.2 kPa O_2 – cross section



- Oxide thickness and interfacial pores near nodules larger than far from nodules

Bratten A, Wen HM, et al., "High-Temperature Oxidation Behavior of the SiC Layer of TRISO Particles in Low-Pressure Oxygen," *Journal of the American Ceramic Society* 106 (2023) 3922-3933.

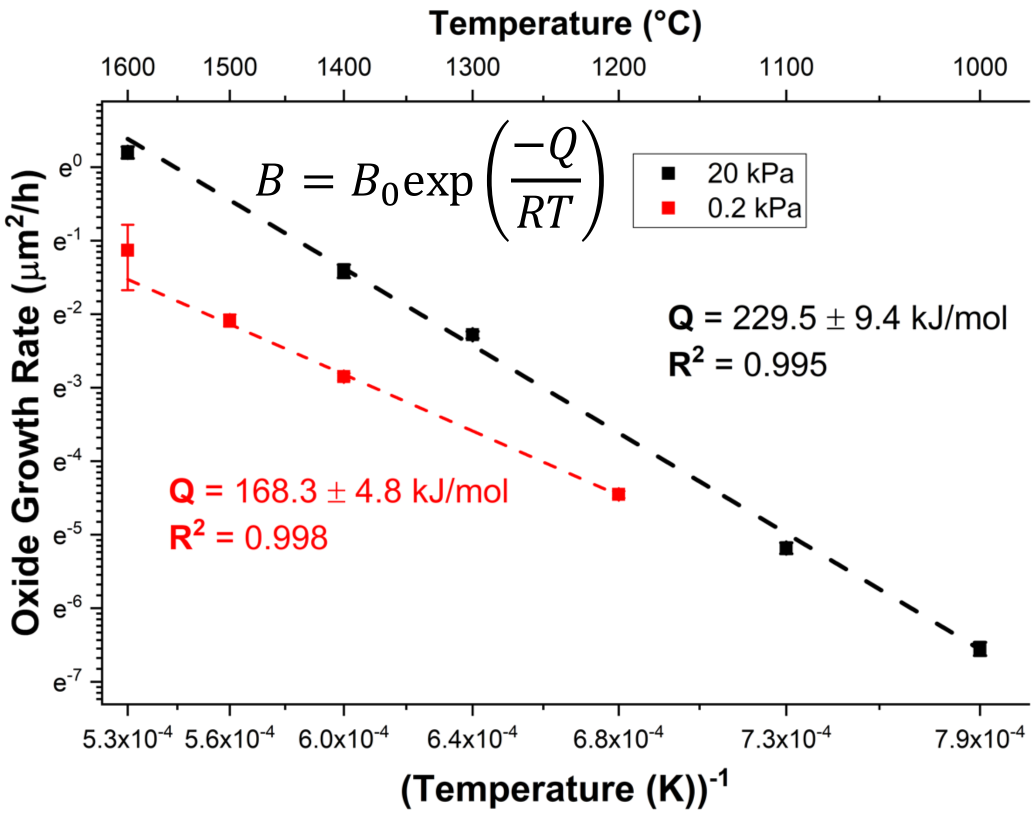


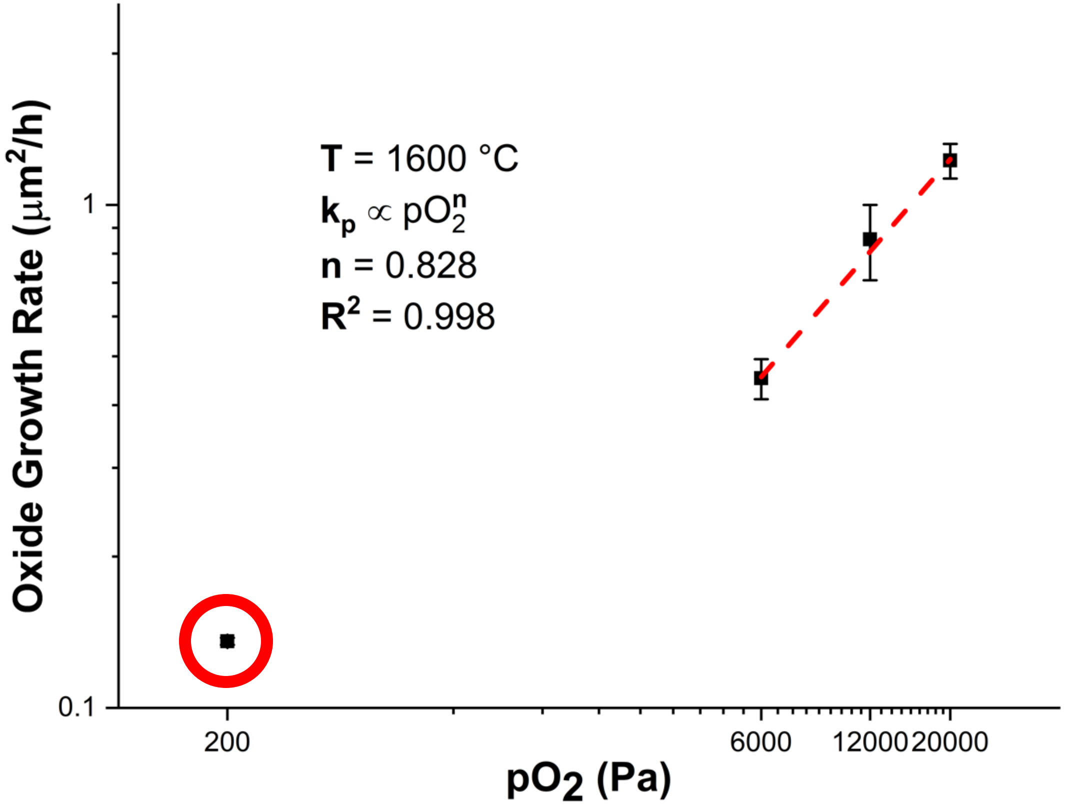
Through the nodules

Away from the nodules

Oxidation Kinetics in O_2 Environments

- Oxide growth mechanisms consistent across all temperatures; different at 20 kPa vs 0.2 kPa O_2
  - Passive oxidation at 20 kPa O_2 from 1000 – 1600 °C
  - Passive oxidation, as well as active oxidation + redeposition by $SiO_{(g)} + \frac{1}{2}O_{2(g)} \rightarrow SiO_{2(s)}$ at 0.2 kPa O_2 from 1200 – 1600 °C
- Change in oxide growth mechanism between 0.2 and 6 kPa O_2 based on pO_2 dependence



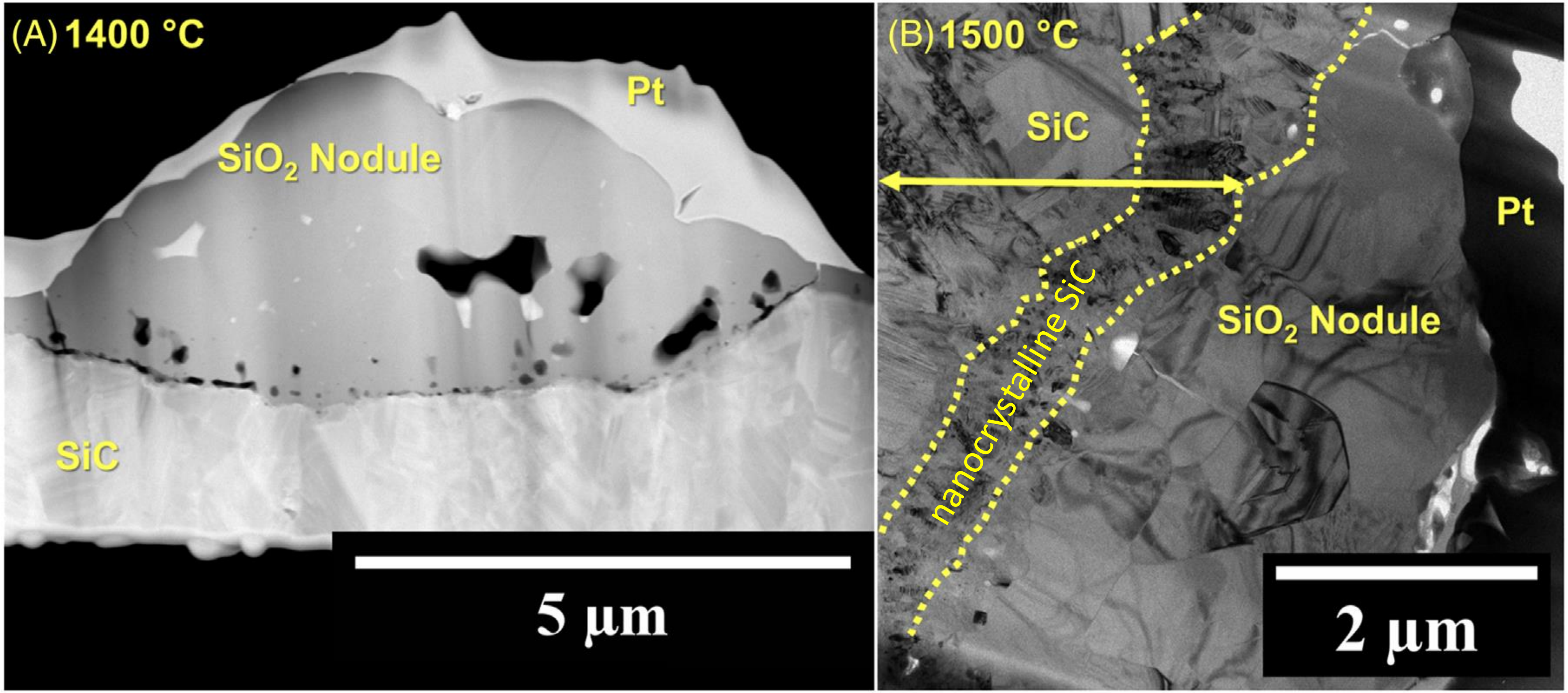


(S)TEM Analysis of Nodules

0.2 kPa O_2, 2 h oxidation

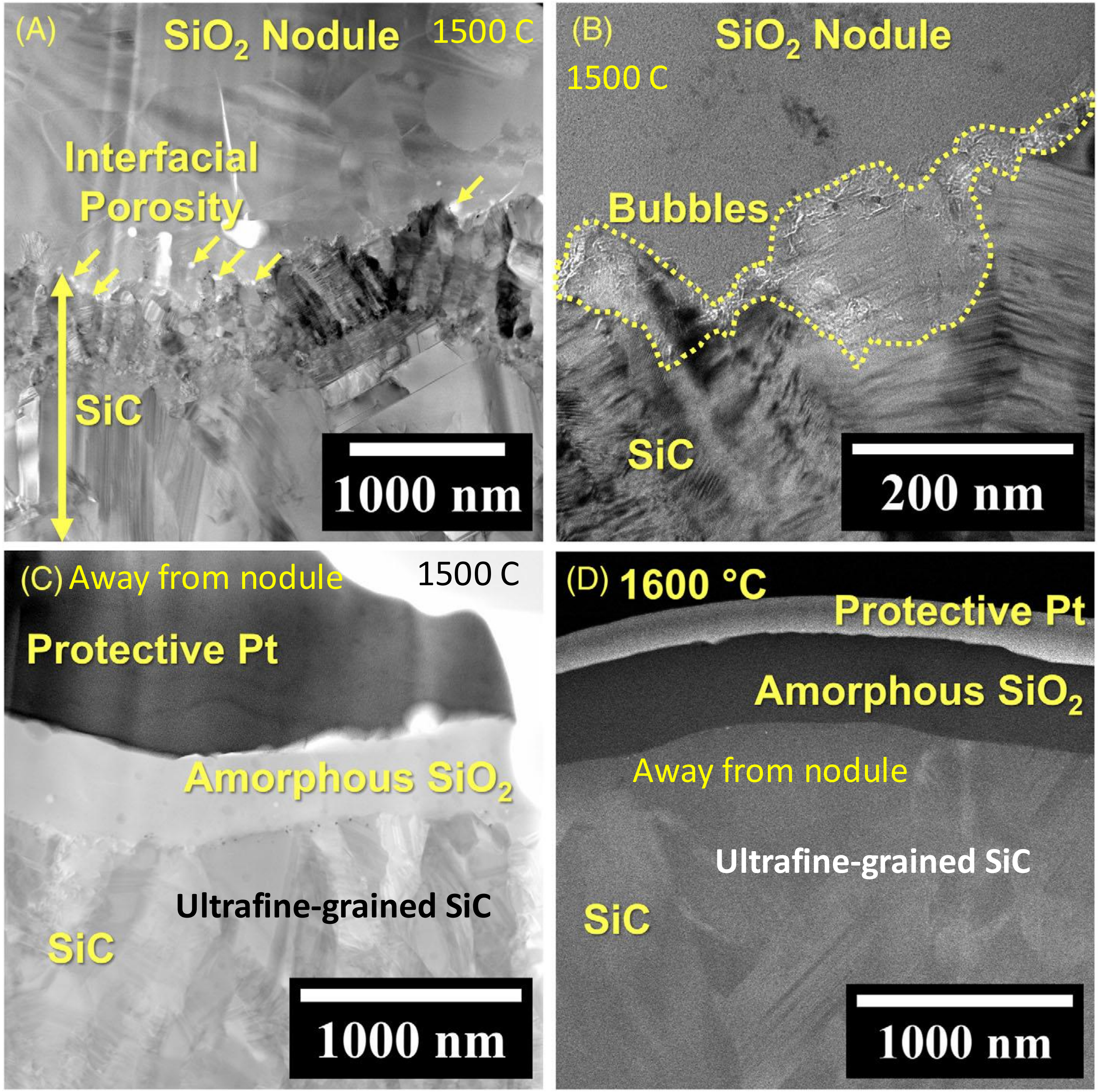
STEM image

TEM image



- Crystalline SiO_2 in nodule, cracks and pores in nodule
- Nanocrystalline SiC under nodules
- Significant porosity at interface between oxide nodule and nanocrystalline SiC

STEM Analysis of SiO_2



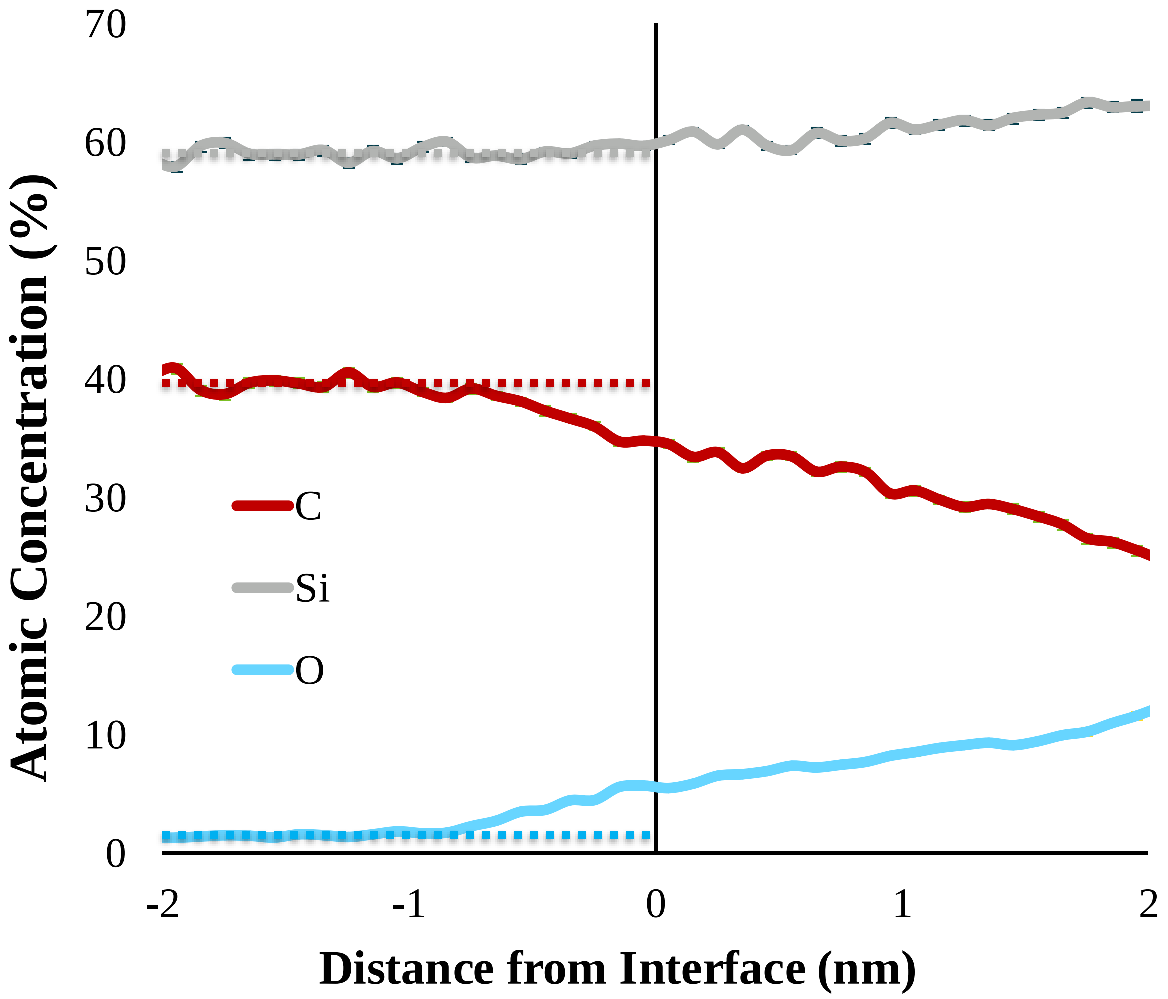
0.2 kPa O_2 oxidation

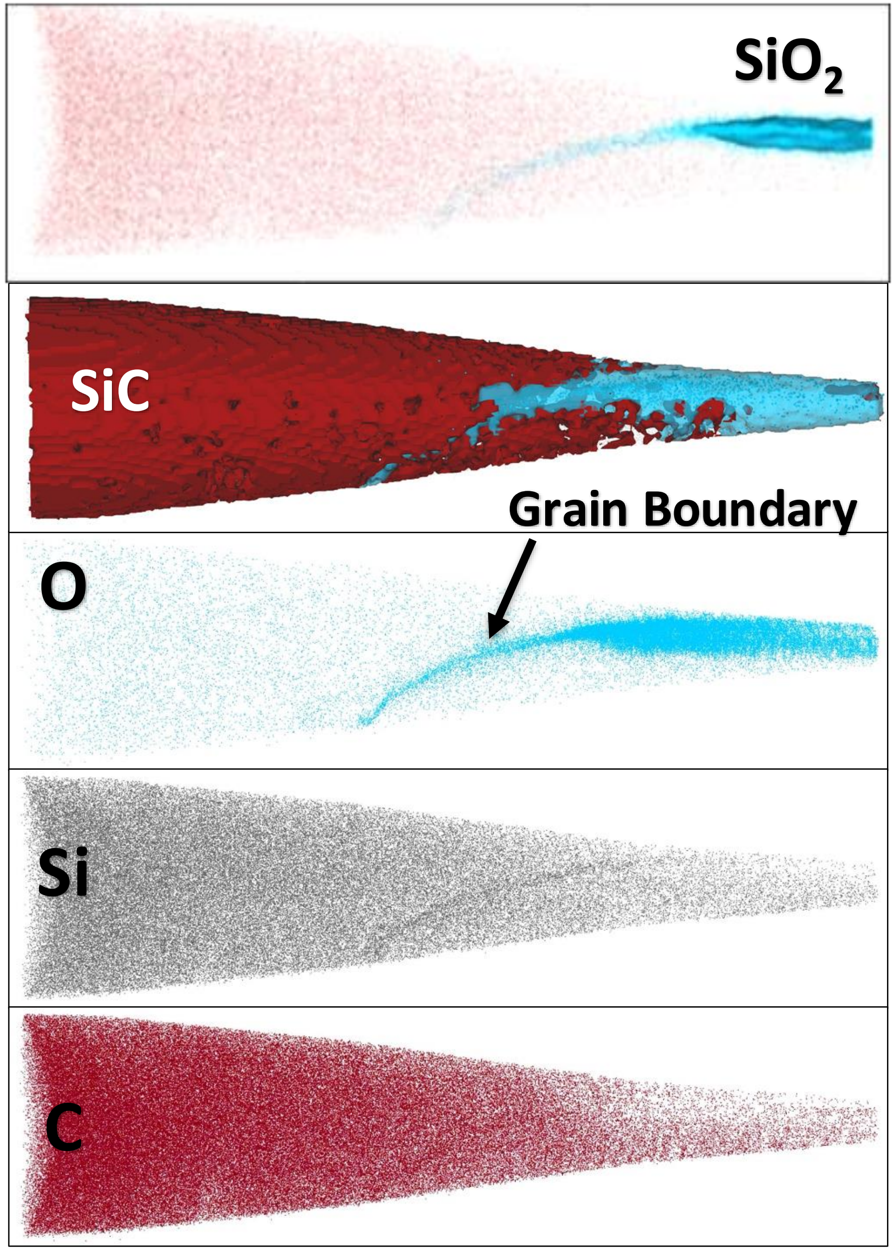
- Crystalline SiO_2 in nodule, with interfacial porosity/bubbles
- Nanocrystalline SiC within ~750 nm of SiC-SiO_2 interface under nodules
- *Continuous and amorphous SiO_2 away from nodules; significantly larger SiC grains (ultrafine-grained) underneath.*
- Nanocrystalline SiC facilitate formation of crystalline SiO_2 nodules

Bratten A, Wen HM, et al., "High-Temperature Oxidation Behavior of the SiC Layer of TRISO Particles in Low-Pressure Oxygen," *Journal of the American Ceramic Society* 106 (2023) 3922-3933.

Oxidation in 1600 °C, 0.2 kPa O_2

- Carbon removed from grain boundary region
- Directly correlated with oxygen penetration along GBs





Summary: Oxidation of SiC of Surrogate TRISO Particles in O_2 containing environment

- Oxidation in 20 kPa produces oxide scales with uniform thickness
  - Behavior effectively described as passive oxidation
  - Amorphous scale forms first, devitrification then occurs, producing spherulitic regions/cracks
  - Time needed for crystallization decreases with increasing temperature
  - Longer time oxidation leads to cracking of the crystalline oxide scale
- Oxidation in 0.2 kPa O_2 produces nonuniform oxide layer
  - SiC oxidation involves both passive and active oxidation
  - Nanocrystalline SiC promotes active oxidation, followed by redeposition of SiO_2 to form crystalline nodules.
  - Above ultrafine-grained SiC, only passive oxidation occurs.
- Enhanced O diffusion along grain boundaries in nanocrystalline region of SiC may cause extraction of C and formation of CO, which builds up at SiC-SiO_2 interface and promotes active oxidation, followed by redeposition of SiO_2

December 3-5, 2024

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PyC Creep and SiC Fracture

Out-of-pile PyC creep should be zero as should SiC fracture

2024 Workshop on Storage and Transportation of TRISO and Metal Spent Nuclear Fuels

Pyrolytic Carbon (PyC) Creep

- **Thermal creep** of PyC has been regarded as "probably not significant" for temperatures below ~2000°C.

  Prados, J.W. and J.L. Scott. 1967. "The influence of pyrolytic carbon creep on coated particle fuel performance." *Nuclear Applications*. Vol. 3. pp. 488-494.

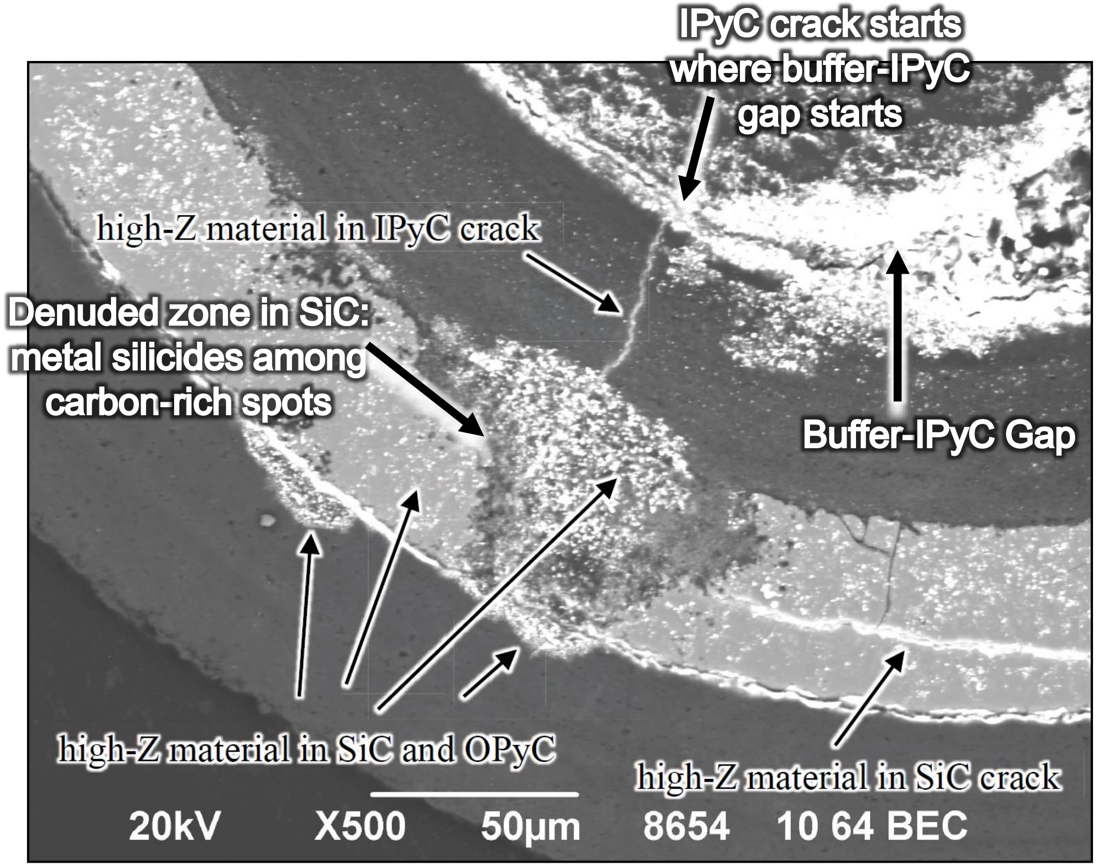
- **Irradiation creep** of PyC occurs in-pile, and it requires two things:
  - Stress
  - Fast neutrons >0.1 MeV

  Price, R.J. and J.C. Bokros. 1967. "Mechanical properties of neutron-irradiated pyrolytic carbons." *J. Nucl. Mater.* Vol 21. pp. 158-174.

- Irradiation creep will stop once the fuel has been removed from the reactor because the only fast neutrons are from spontaneous fission of U, Pu, and greater actinides
- The stress state in IPyC may be lower out of pile because low SNF temperatures reduce internal gas pressure (for α-decay see slide 4).

SiC Failure via Mechanical Fracture not Observed in the U.S. UCO TRISO Program

- Observed SiC failures have been via a multistep process:
  1. Buffer densification pulls on IPyC
  2. IPyC breaks leaving the intact SiC coating exposed to the particle interior
     - IPyC break may appear elastic
     - Or IPyC break may appear to be more of a gradual tear
  3. Fission products (Pd, Ag) and some U migrate to the IPyC/SiC interface to varying degrees
  4. With an IPyC break, Pd, Ag, and U react chemically with SiC, making silicides and leaving carbon-rich zones that traverse the SiC layer. SiC near denuded zone may also crack.



Micrograph from Hunn et al., 2019. ORNL/TM-2019/1201. Three additional labels were added here.

SiC Failure and SNF

- With no appreciable neutrons and low temperatures in SNF, the multistep failure mode for SiC shown on the previous slide cannot happen.
- Low SNF temps protect against pressure vessel failure by reducing existing internal pressure and trapping gas (such as helium from α-decay) in the kernel
- Even with IPyC and SiC failure, the OPyC most often remains intact and will retain fission gas!
  - Observed in-pile SiC failure rate for AGR-1&2 [1]: 1.9E-5
  - Observed in-pile TRISO failure (failure of IPyC, SiC, and OPyC) [1]: ≤9.7E-6
- Metallic fission products *may* migrate out of a failed SiC layer, BUT that is a thermally driven process requiring high temperatures
- With irradiation temperatures averaging ~850–950°C for 369 EFPD in ATR, 90-95% of the Cs from exposed kernels in AGR-3/4 was still retained in the compacts after irradiation [2]

1. *Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance: Topical Report EPRI-AR-1(NP).* EPRI, Palo Alto, CA: 2019. 3002015750.
2. Stempien, J.D. and L. Cai. 2024. “Radial Deconsolidation and Leach-burn-leach of Eight As-irradiated AGR-3/4 TRISO fuel Compacts.” INL/RPT-24-77357. Idaho National Laboratory.

Time-Dependent Failure Analysis of TRISO Fuel

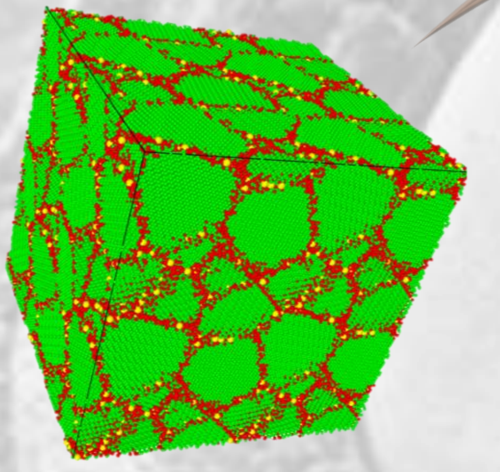
Wen Jiang [a,b]

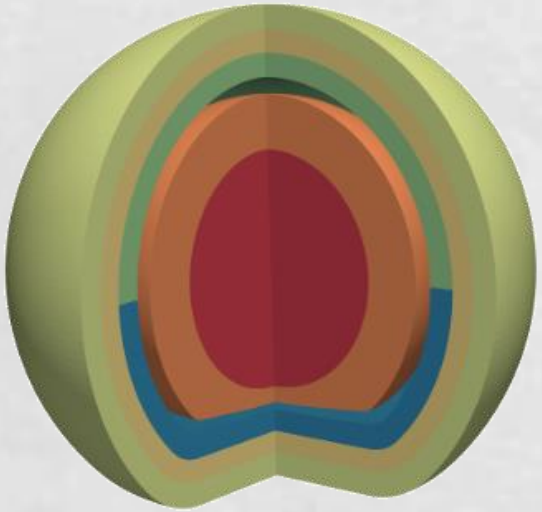
[a] Department of Nuclear Engineering, NC State University

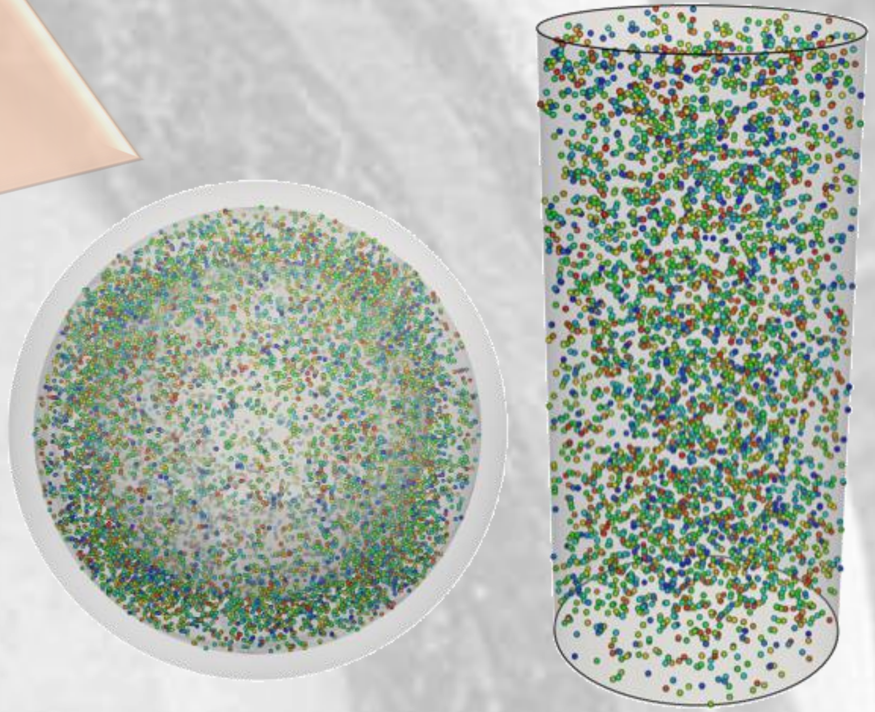
[b] Idaho National Laboratory

Multi-scale TRISO Modeling overview

BISON







Lower-length scale modeling

- **Fission gas release model:** Xe, Kr diffusivity in UCO
- **Fission product diffusivity:** Silver diffusion in SiC, Pd Penetration

TRISO particle

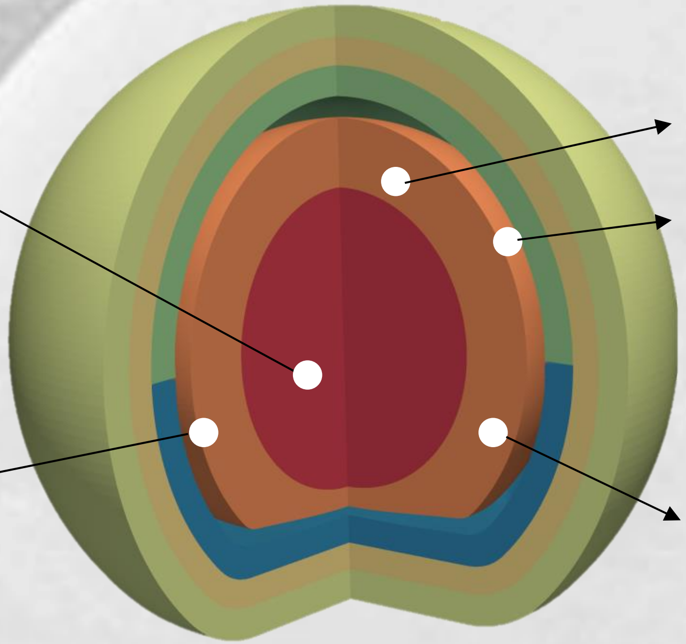
- **Thermal-mechanical modeling**
  - **Failure analysis:** asphericity, IPyC cracking and debonding
- **Fission product diffusion through layers**

Pebble and Compact modeling

- **Failure probability calculation**: Monte Carlo and Fast Integration Approach
- **Fission product diffusion through matrix**
- **Particle-Matrix interaction**

TRISO Materials Models

Kernel (350-500 μm):
- UO_2 or UCO;
- Retention of fission products
  - ❖ Thermal
  - ❖ Burnup
  - ❖ Elasticity Tensor
  - ❖ Volumetric Swelling
  - ❖ Fission Gas Release



IPyC (~40 μm)
- Protects kernel from chlorine during SiC deposition
- Surface for SiC deposition
- Contributes to fission gas retention
- Irradiation shrinkage contributes to compression in SiC layer
  - ❖ Thermal Expansion
  - ❖ Elasticity Tensor
  - ❖ Creep
  - ❖ IIDC

OPyC (~40 μm)
- Contributes to fission gas retention
- Surface for bonding to matrix
- Protects SiC layer during handling

Buffer (~100 μm)
- ~50% dense pyrolytic carbon
- Provides space for fission gas and CO(g) accumulation
- Accommodates fission recoils
  - ❖ Thermal
  - ❖ Elasticity Tensor
  - ❖ Creep
  - ❖ Thermal Expansion
  - ❖ IIDC

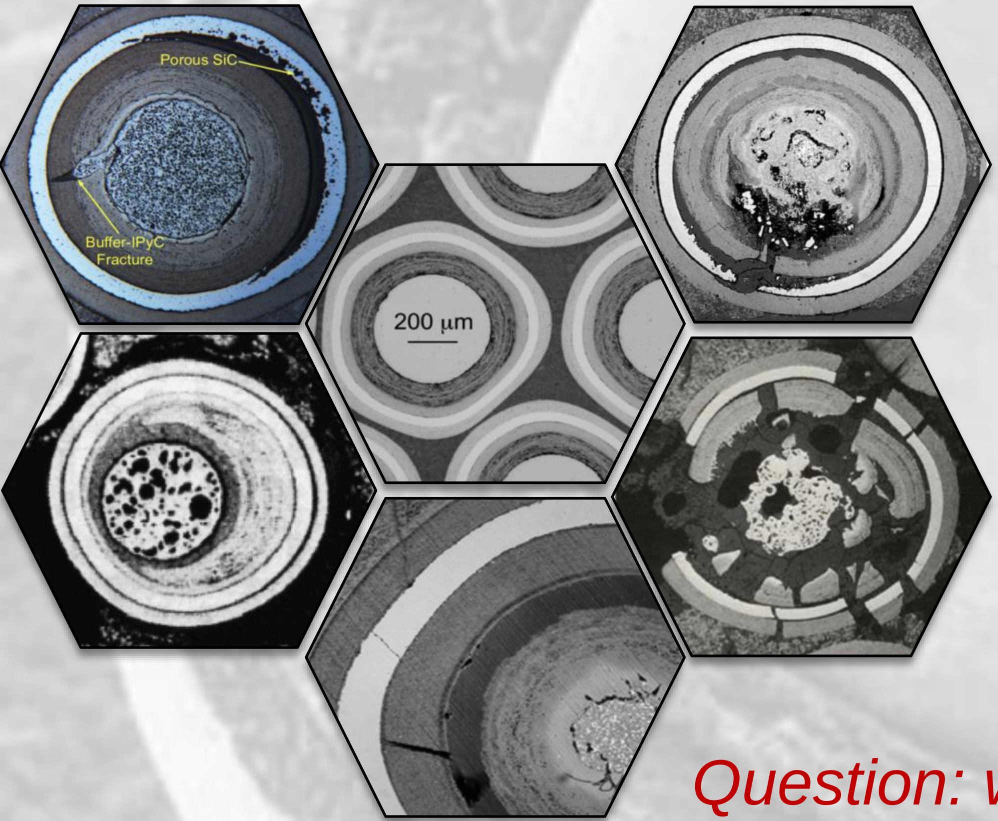
SiC (~35 μm)
- Main structural layer
- Primary coating layer for retaining non-gaseous fission products
  - ❖ Elasticity Tensor
  - ❖ Thermal Expansion

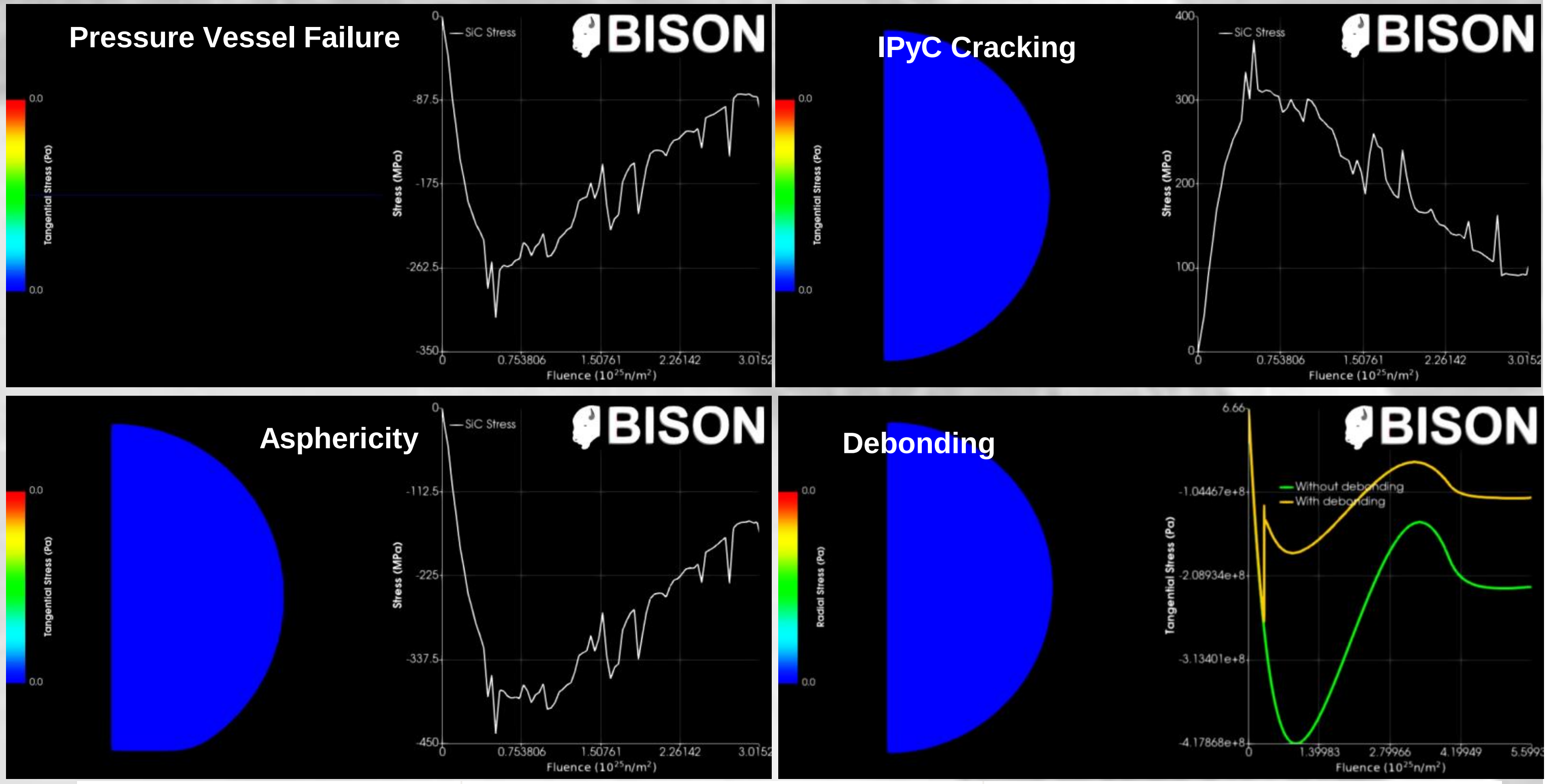
Questions: are these models still valid during long term storage?
what are the boundary and loading conditions?

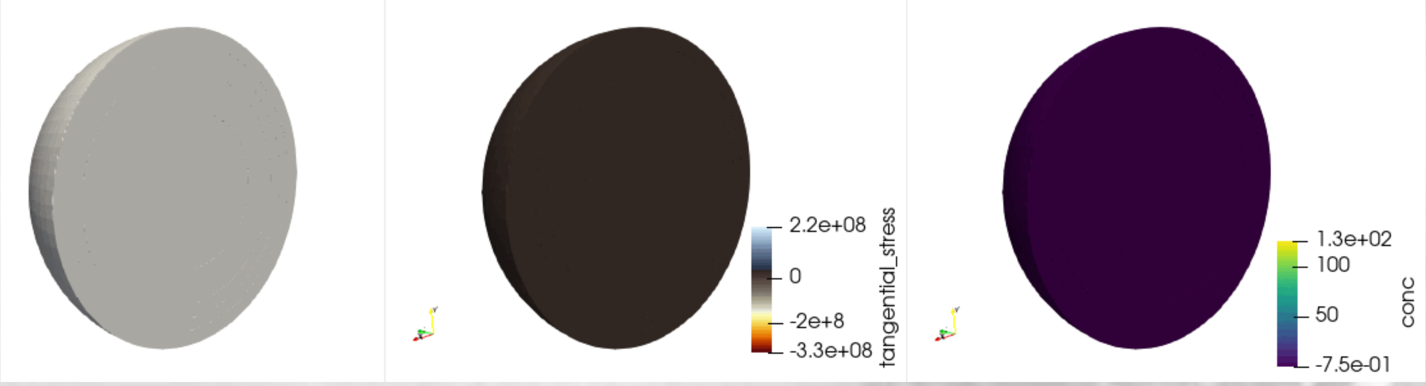
TRISO Failure Analysis

Mechanical
- Pressure vessel failure
- Irradiation-induced PyC failure leading to SiC cracking
- IPyC-SiC / Buffer-IPyC partial debonding

Thermochemical
- Kernel migration
- SiC thermal decomposition
- Fission product attack of SiC
- Corrosion of SiC by CO







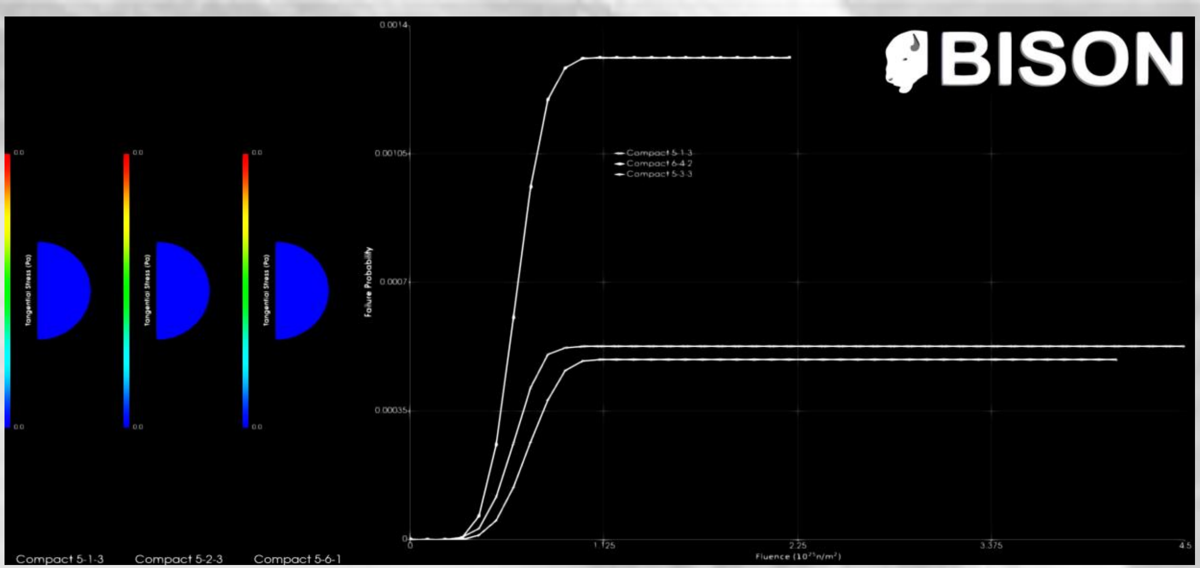
Question: what are failure modes during long term storage?

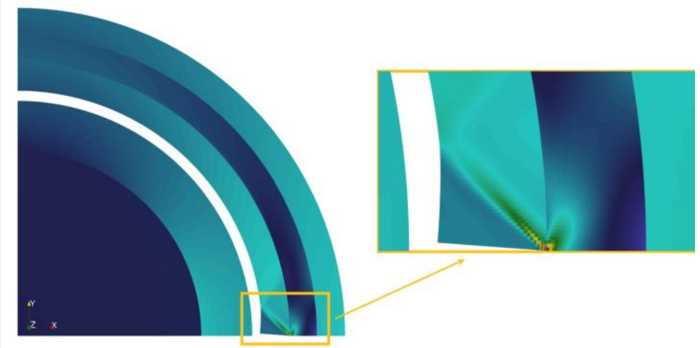
Failure Analysis

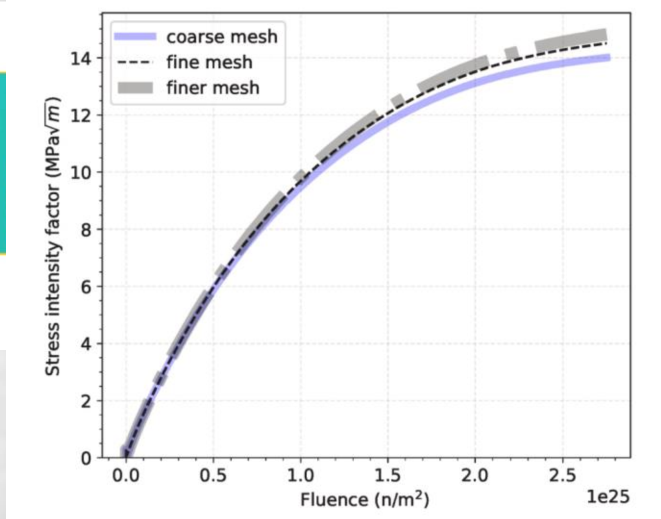
- In the Weibull theory, the failure probability is:

$$P_f = 1 - \exp\left(-\int \left(\frac{\sigma_c}{\sigma_{ms}} \right)^m dV \right)$$

- Fracture-mechanics based criterion
  - Stress intensity factor (K)
  - Energy release rate (J)
- Subcritical crack growth
  - Crack growth when stress is under the fracture strength
  - Fatigue? environmental-assist crack growth? Oxidation?







Question: how to determine failure during long term storage?

$$\frac{da}{dN} = C(\Delta K)^m$$

$$p_{f,t}(t;\alpha,m_t) = \left(\frac{m_t}{\alpha}\right)\left(\frac{t}{\alpha}\right)^{m_t-1} \cdot e^{-\left(\frac{t}{\alpha}\right)^{m_t}}$$

GAS-COOLED REACTOR
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December 3, 2024

Safety Considerations for Irradiated Graphite

Rebecca Smith

Idaho National Engineering Laboratory

2024 Workshop on Storage and Transportation of TRISO and Metal Spent Nuclear Fuels

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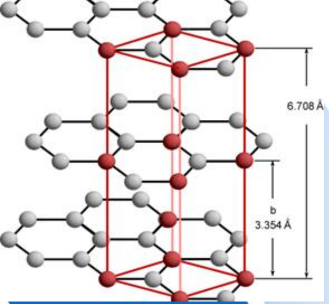
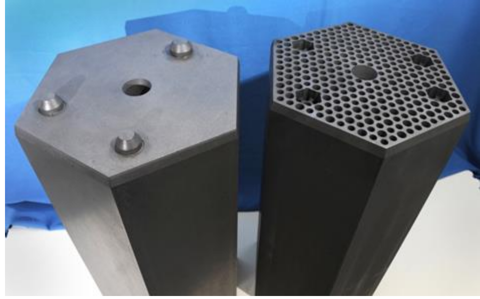


Atomic Carbon (~2Å)

$$C(s) + \left(1 - \tfrac{1}{2}x\right)O_2(g) \leftrightarrow xCO(g) + (1-x)CO_2(g)$$

$$C(s) + O_2(g) \leftrightarrow CO_2(g)$$

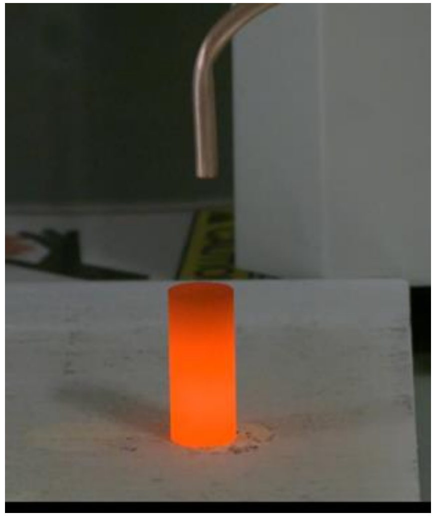
$$\Delta H_{298K} = -393.5\ \text{kJ/mol}$$



Graphite Reactor Components (cm to m)

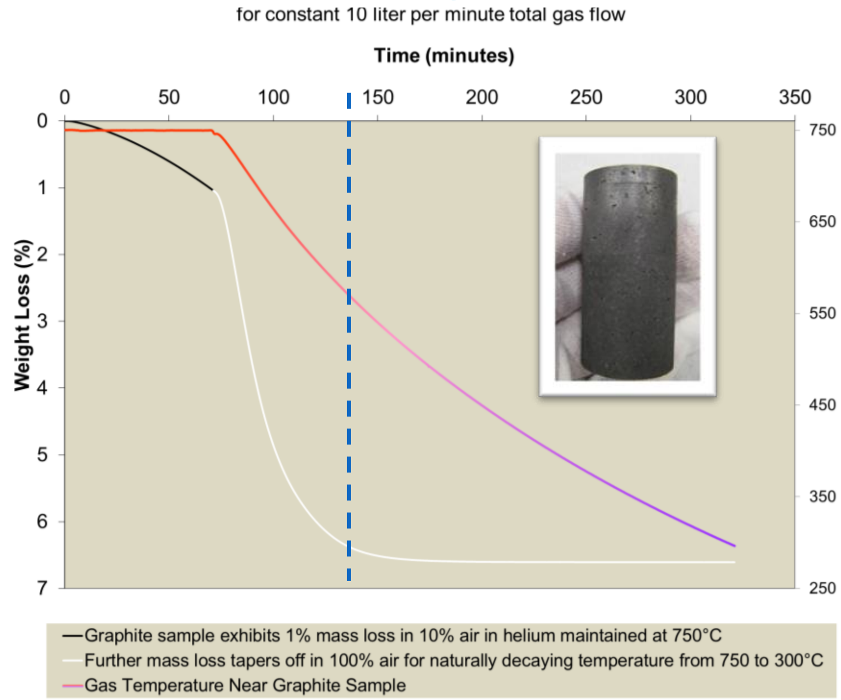
Graphite Does Not Burn

Graphite as a Class D Fire Extinguishing Agent[1,2]

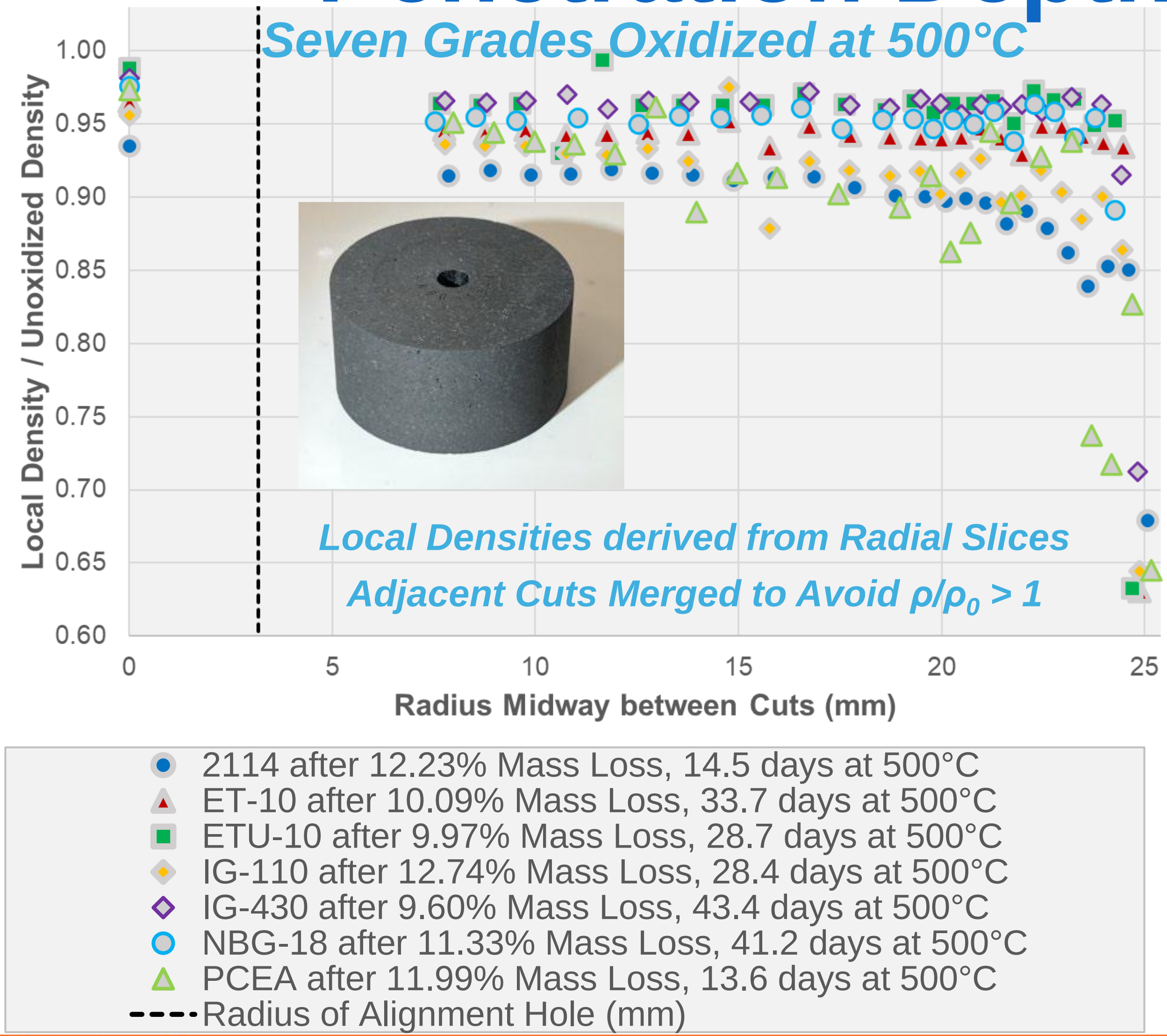


Radiant Heat Transfer

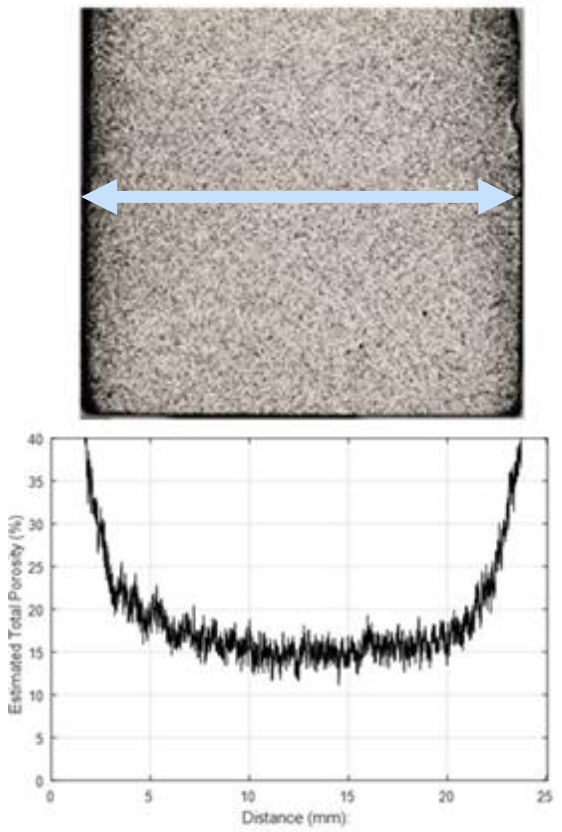
Self-Limiting Oxidation



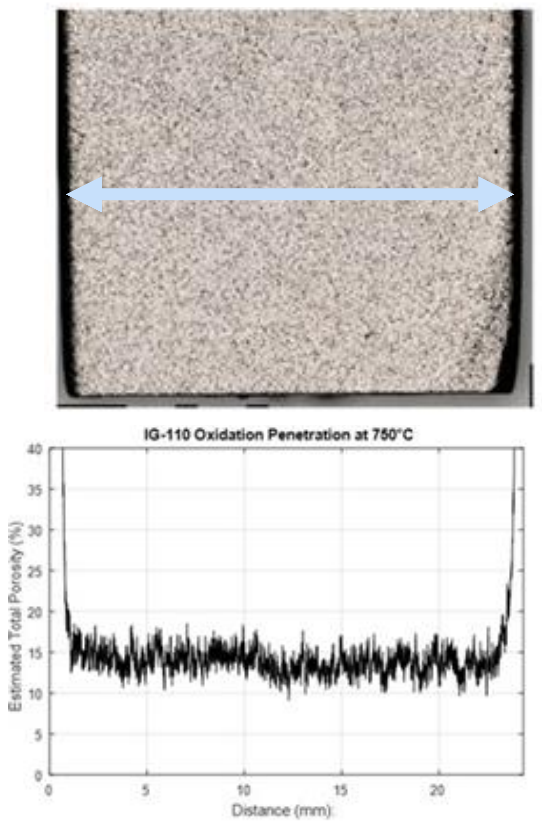
Oxidation Temperature and Penetration Depth



IG-110 Image Oxidized at 650°C

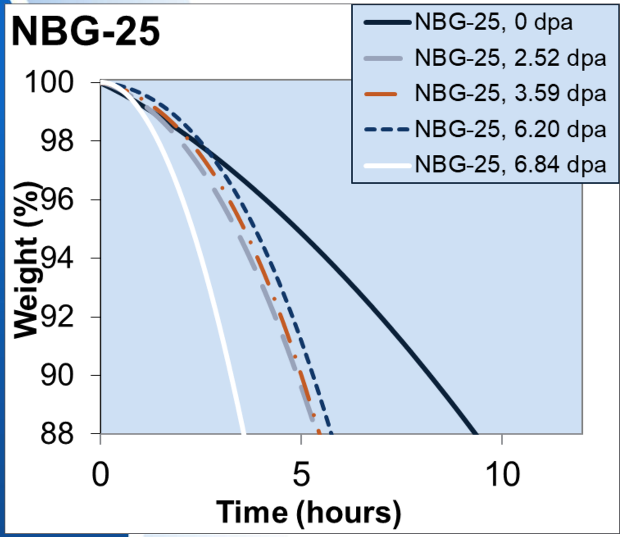


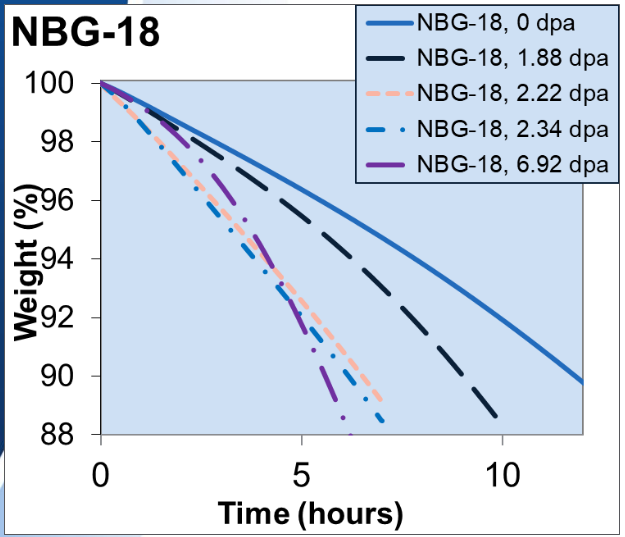
IG-110 Image Oxidized at 750°C



Oxidation Rates after Irradiation

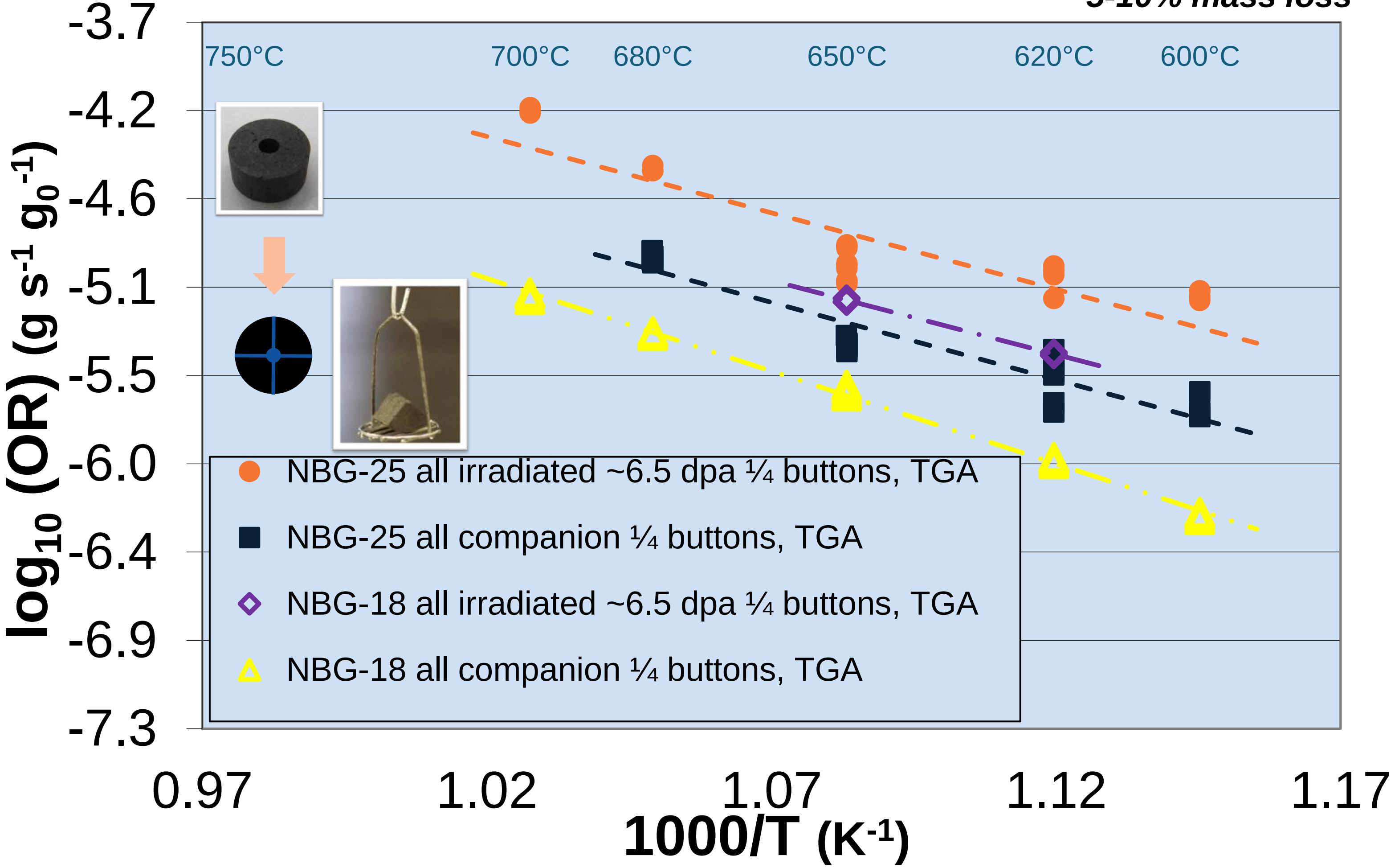
Mass Normalized Split Sample TGA Data





Dose Dependency at 650°C

5-10% mass loss

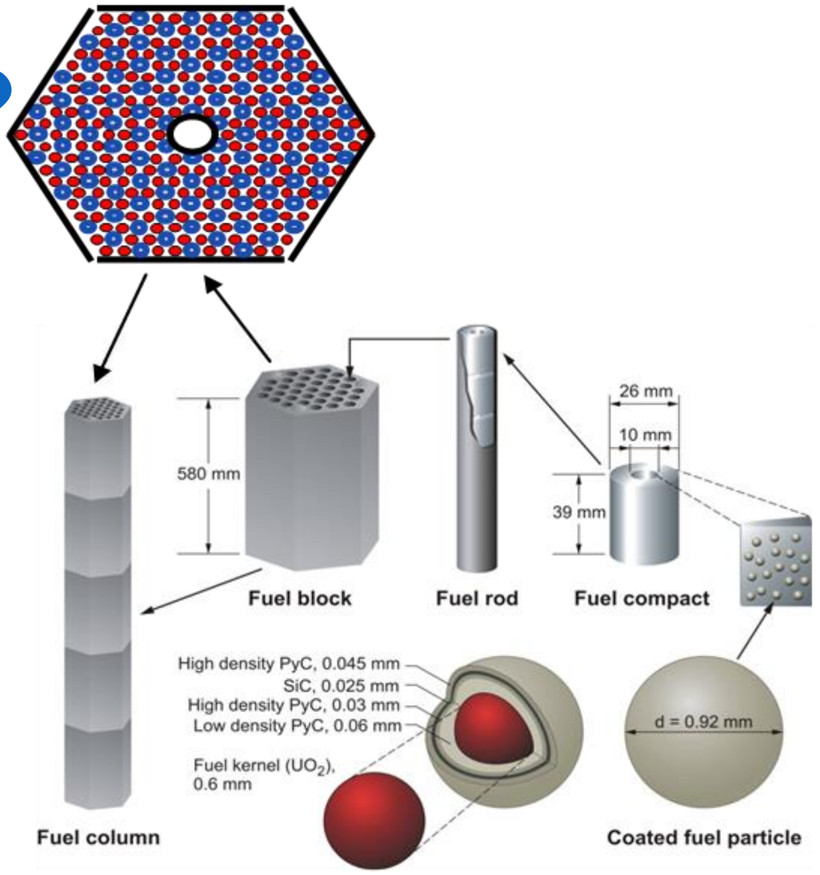


Arrhenius Relationship

$k = A \bullet \exp(-E_a/RT)$

So, What's the Problem?

ASME Guidance, ASTM Test Standards



- Oxidation Effects (1% and 10% rules)
  Strength & Thermal Properties
- Irradiation Facilitates (kinetic regime) Oxidation
  (rate to 10% mass loss doubles at 6 dpa)
- Relative Performance of Matrix
  (an order of magnitude faster than graphite)
- Considering (boundary layer diffusion regime) Complete Oxidation for Disposal[3]
  - Rate depends on T, with increase in CO/CO_2 ratio above 1000°C
  - Efficiency depends on air flow rate
  - Particle size didn't affect much
  (nibble & vacuum 2 mm – 6 mm range tested)
  - Virgin and irradiated graphite performed similarly

Just because it has graphite in it doesn't mean it will perform as well as graphite!

At 500°C, fully graphitized material takes 24 hours or more to oxidize to 1% mass loss!

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Thank you

Rebecca Smith

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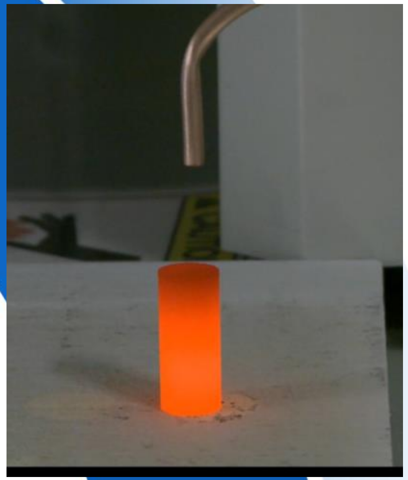
U.S. DEPARTMENT OF ENERGY

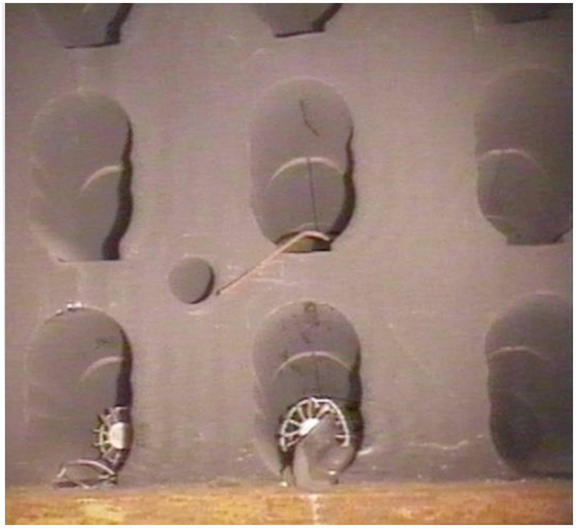


But What About…

Fires After the Accident at Chernobyl?[4-12]

- Explosion expelled tons of hot nuclear material (graphite + fuel)
- Glow of red-hot material (>650°C)
- Fires on bitumen coated rooftops and among debris and vegetation nearby
- Reports of **"a column of white combustion products (white smoke) rising several hundreds of meters into the sky"** not consistent with the colorless, odorless CO_2 and CO expected from graphite
- INSAG-7 (updated from INSAG-1) excludes all mention of the misnomer "graphite fires"

And What Happened at Windscale?[13-17]



Windscale reactor internals before (left) and after accident (right)

- Wigner Energy anneal at 250°C in air
- Hot spots ignited metal fuel rods
- Metal fire did not spread to the graphite moderator

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December 2, 2024

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2024 Workshop on
STORAGE AND TRANSPORTATION OF TRISO AND METAL SPENT NUCLEAR FUELS

Determining the Oxidation Behavior of Matrix Graphite

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Experimental Methods

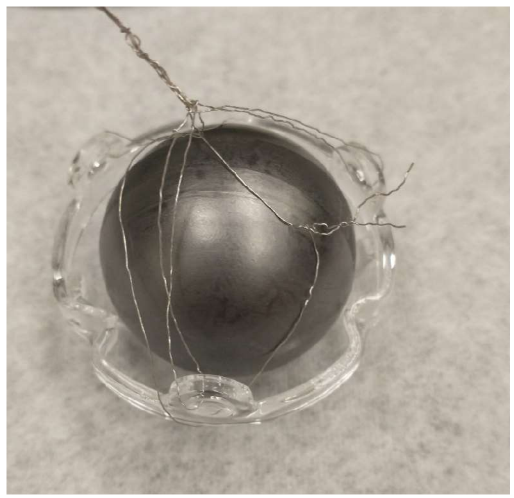
Designation: D7542 – 21

Standard Test Method for Air Oxidation of Carbon and Graphite in the Kinetic Regime

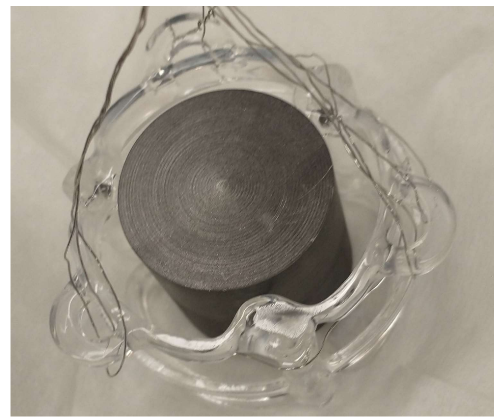
Vertical Furnace

Sample Geometries:

Vertical Furnace

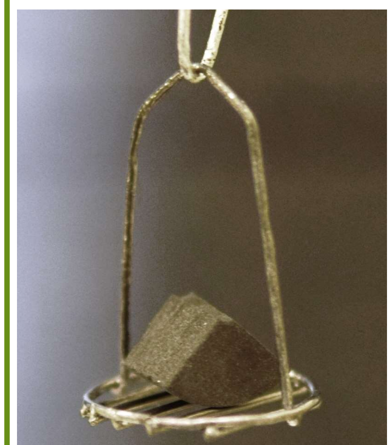


Pebble: D=40 mm (only matrix graphite)



Standardized cylinder D=H=25.4 mm

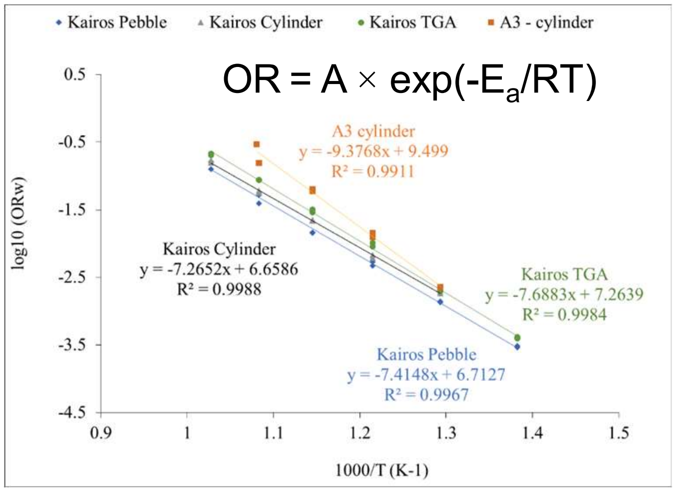
TGA



Quadrant of a disk (D~12 mm, H~6mm)

Oxidation Results (unirradiated) – Kairos Pebble and Cylinder vs. A3

Oxidation Rate (OR_w) normalized by Weight
Temperature range: 450°C – 700°C



Sample	Sample geometry	E_a (kJ/mol)	Pre-exponent ($g\ h^{-1}\ g^{-1}$)
Kairos	cylinder D/H = 25.4 mm	139.1	4.6E+06
Kairos	pebble D = ~40 mm	142.0	5.2E+06
Kairos	A quarter of a disc	147.2	1.8E+07
A3	cylinder D/H = 25.4 mm	179.6	3.2E+09

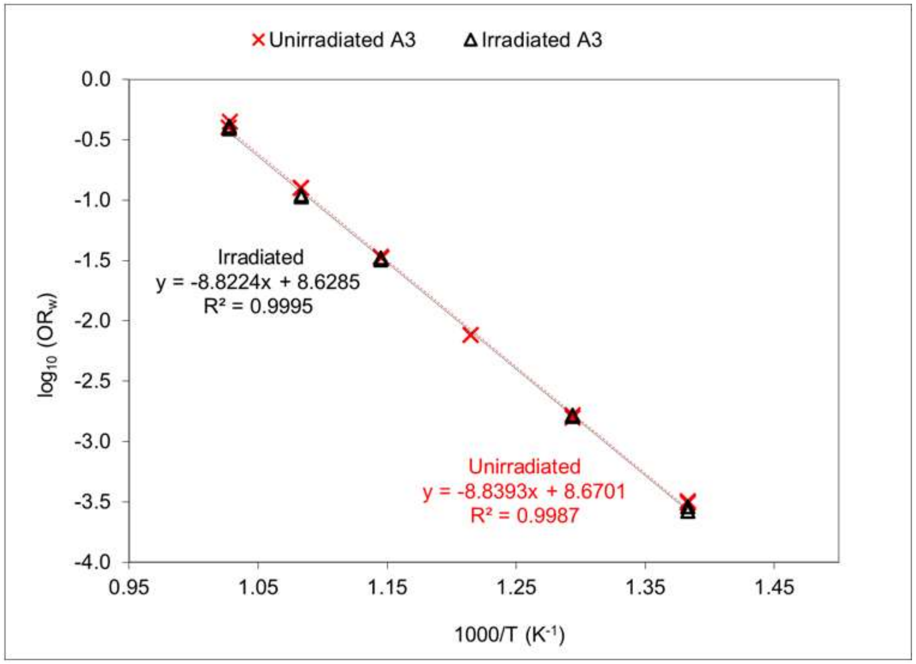
- **Followed Arrhenius equation at temperatures of 450–700°C (kinetic-controlled regime)**
- **Limited sample geometry effects**
- **Lower activation energy (less oxidation resistance) than nuclear graphite**
- **Oxidation rates at 500°C or below are extremely low**

E_a of nuclear graphite (IG-110, NBG-25, purified PCEA, BAN, NBG-17, NBG-18): **188 – 213** KJ/mol

Ref. 1. L.Cai, Journal of Nuclear Materials 589 (2024) 154849
2. R.E. Smith, Journal of Nuclear Materials 545 (2021)152648

Oxidation Results (irradiated) –A3

a disc-quadrant geometry in TGA



ID	Dose (dpa)	Irradiation Temp (°C)	Sample Batch	Oxidation Test Temperature (°C)
H-7-1	6.31	692.3	AGC-1	700
H-17-1	6.52	706.9	AGC-1	600 and 650
H-9-1	6.55	694.7	AGC-1	450 and 500

- No irradiation effects are observed.

Conclusion

- The oxidation rates of matrix graphite follow the Arrhenius equation at 450–700°C with lower activation energy than nuclear graphite.
- Oxidation rates at 500°C or below are extremely low.
- An identical oxidation rate for both unirradiated and irradiated A3 matrix graphite indicated that the non-graphitic carbon dominates the oxidation behavior and that no accelerated oxidation occurred due to irradiation damage at a weight loss of up to 10%.

Acknowledge

We gratefully acknowledge that funding for this work was provided by the U.S. Department of Energy's Advanced Reactor Technologies Program under the DOE Idaho Operations Office, Contract DE-AC07-05ID14517, with Battelle Energy Alliance, LLC.

Oxidation of graphitic components under accident conditions

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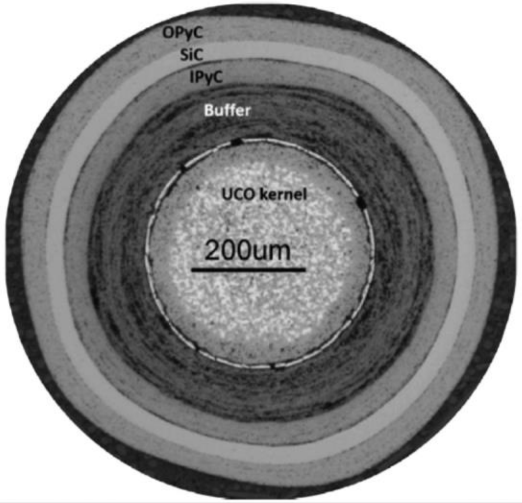
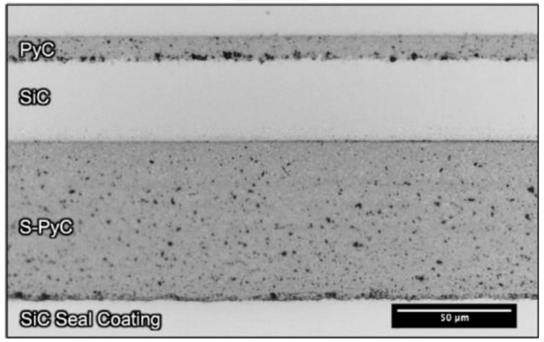
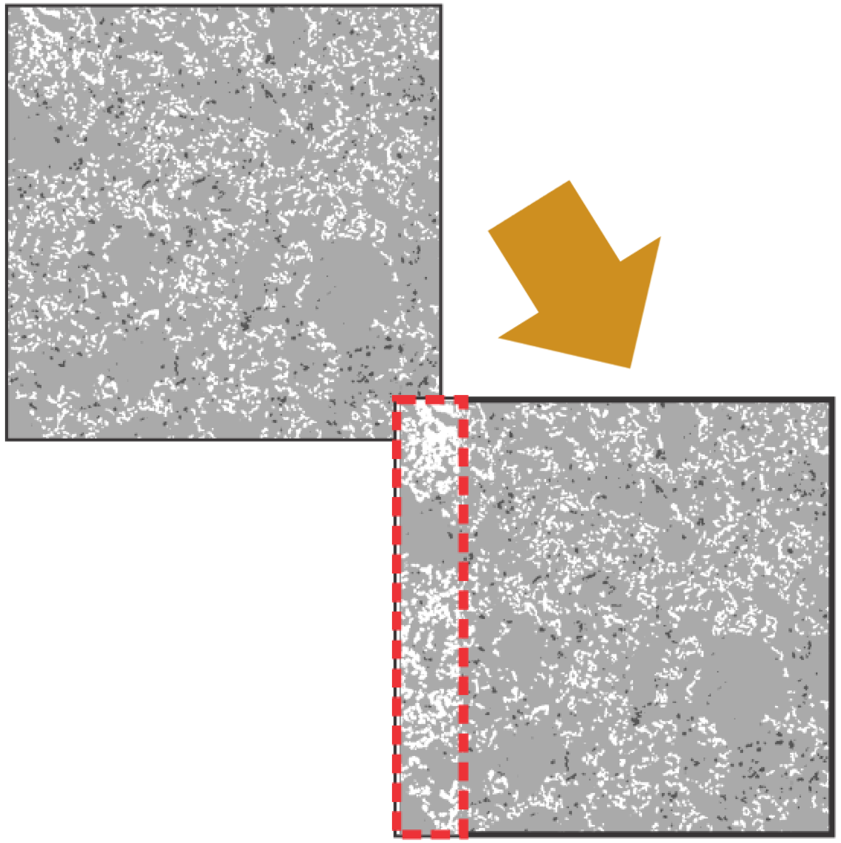
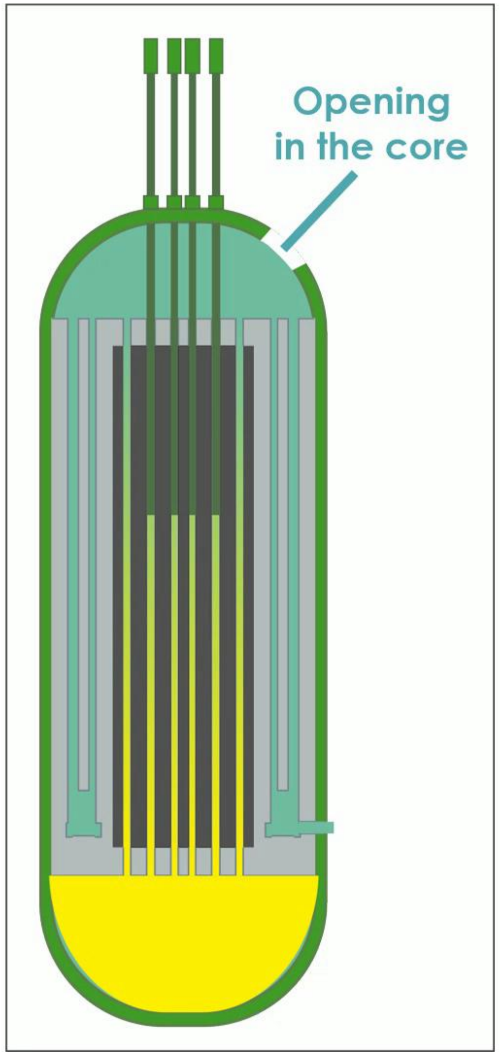
Oxidation of graphitic components under accident conditions

- **Acute oxidation** occurs during the accidental ingress of air into the core
- **Acute oxidation** damage might be localized at the specimen's surface









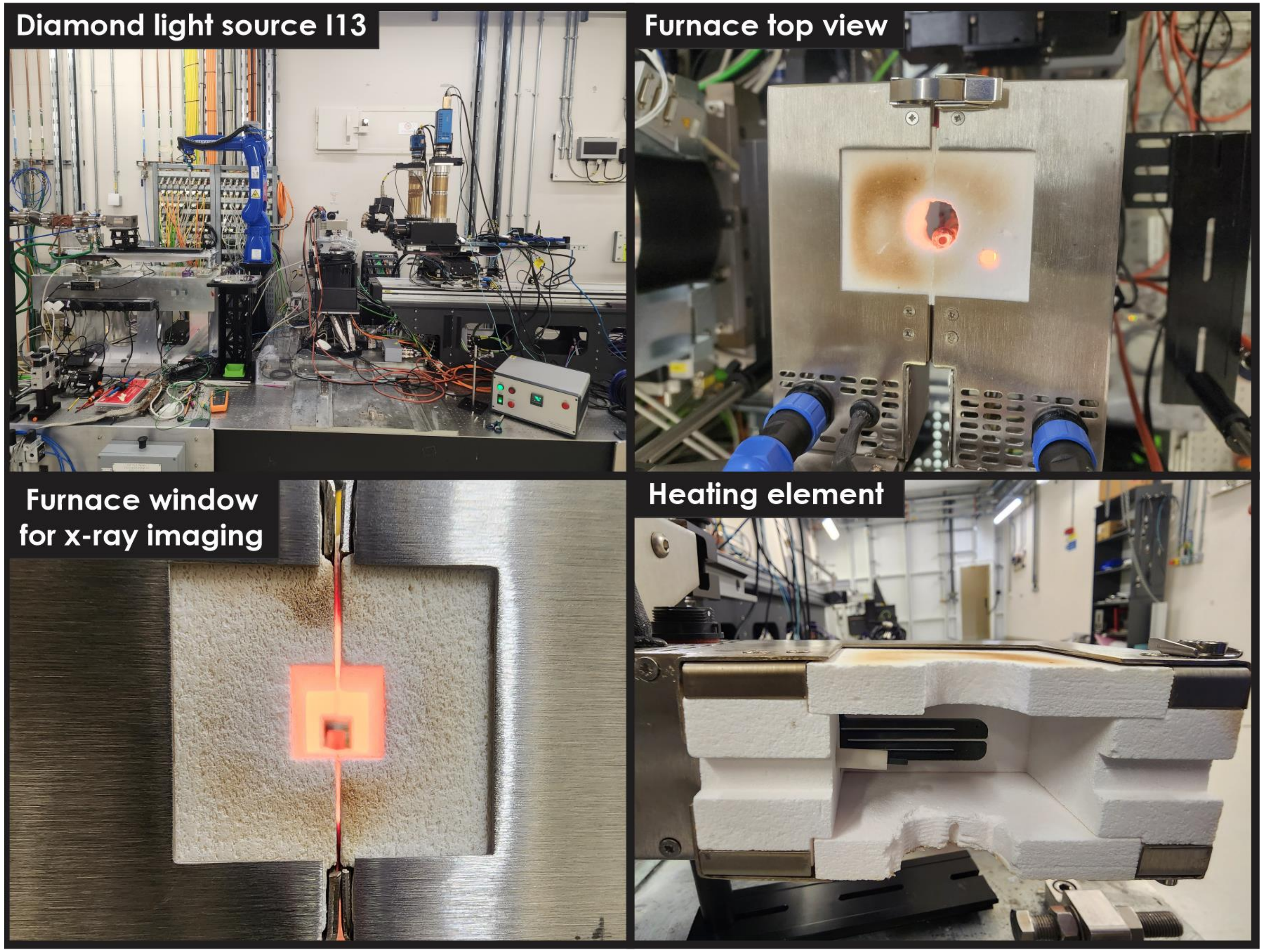
Acute oxidation

Simulation accident conditions in graphite – *In situ* oxidation monitored with XCT

- Samples of IG-110, NBG-18 and PCEA were oxidized at temperatures above 660°C and characterized via synchrotron XCT

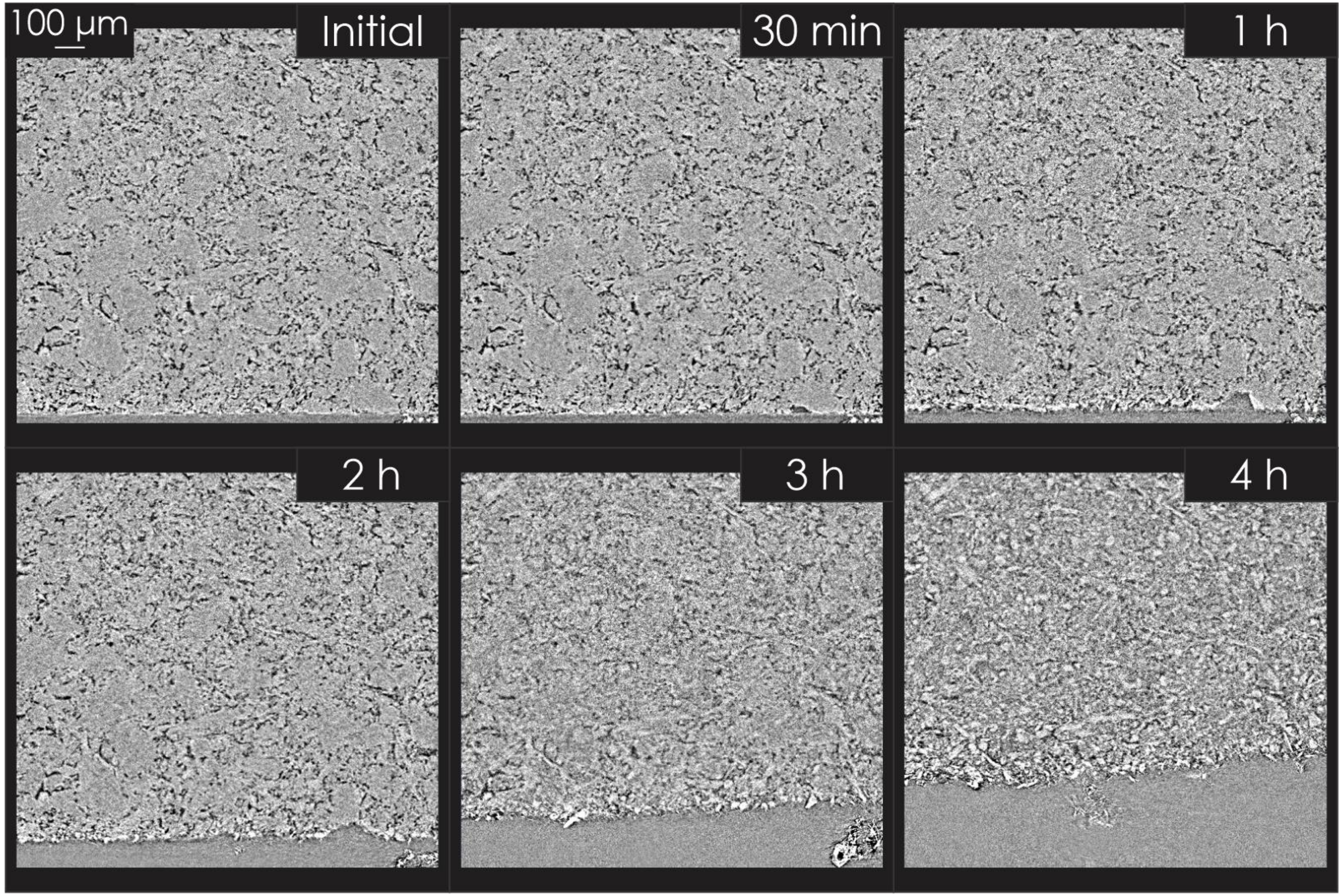
- These experiments were conducted to simulate **oxidation under accident conditions**

- The experiments were performed at the **Diamond Light Source Synchrotron**

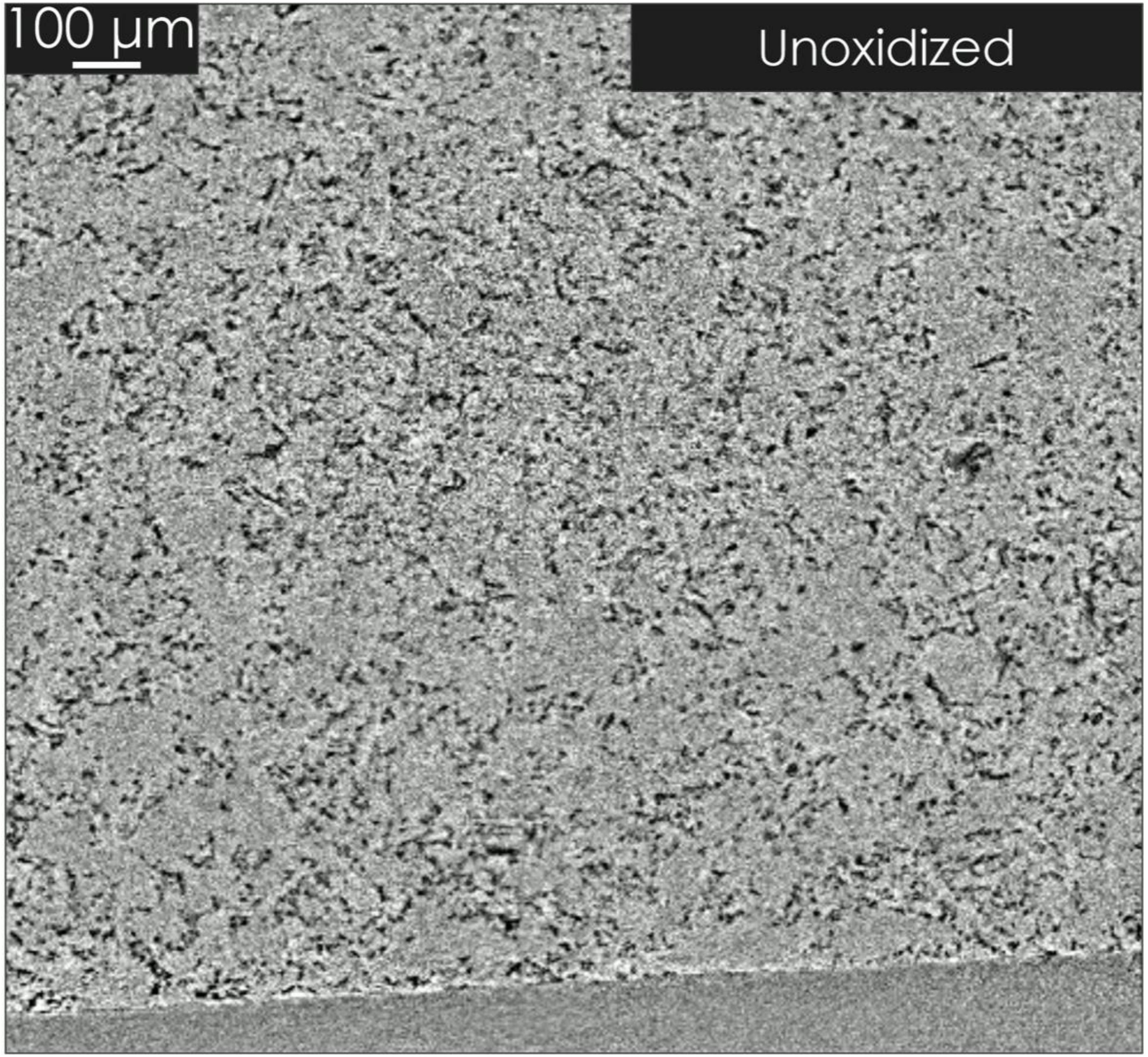


IG-110 oxidation at 660°C

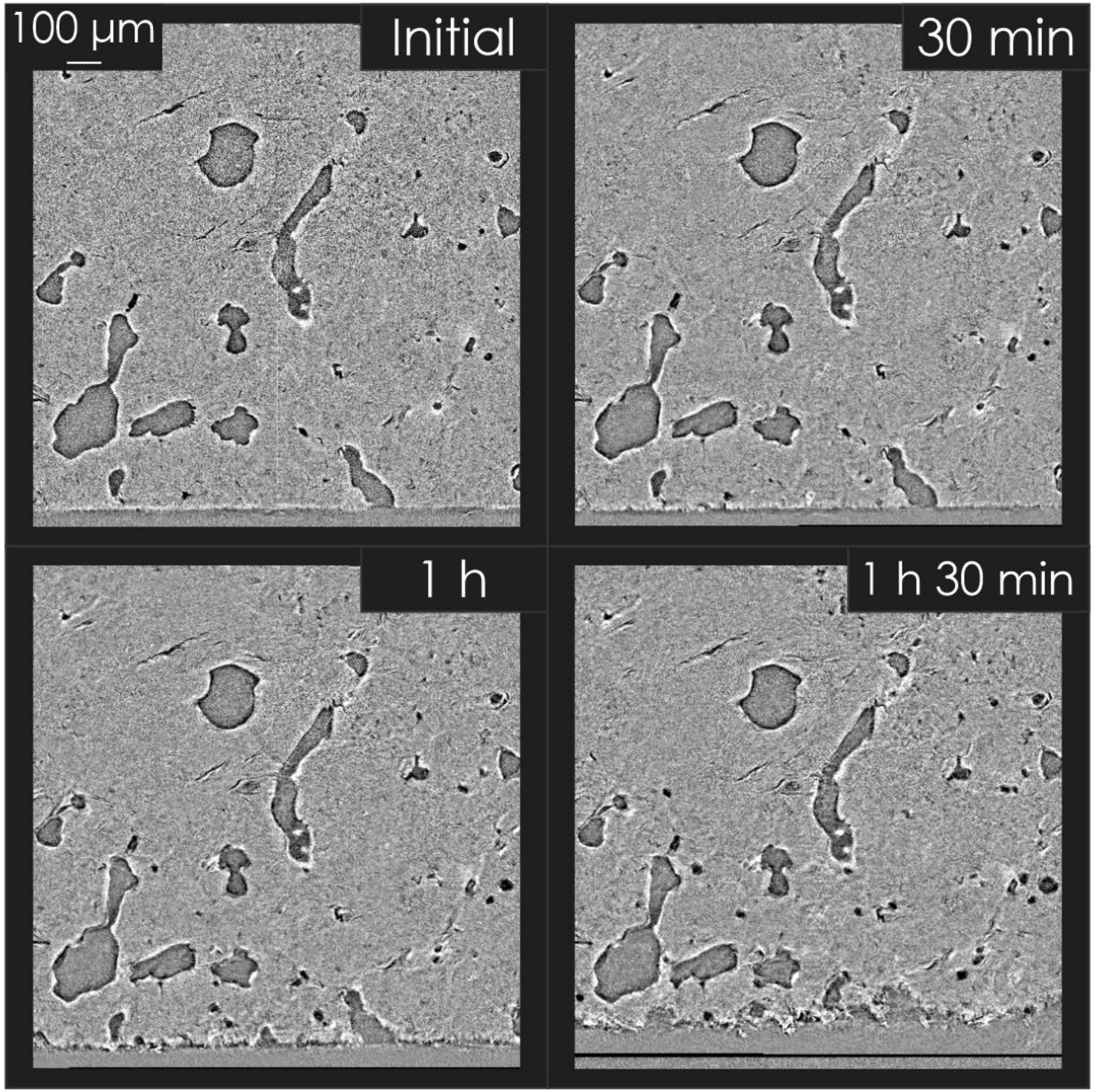
- IG-110 is a graphite with small pores <15 μm
- Degradation occurred at the bottom edge of the sample
- After the second hour, the density of the imaged region began to decrease



In situ oxidation experiment – IG-110 results

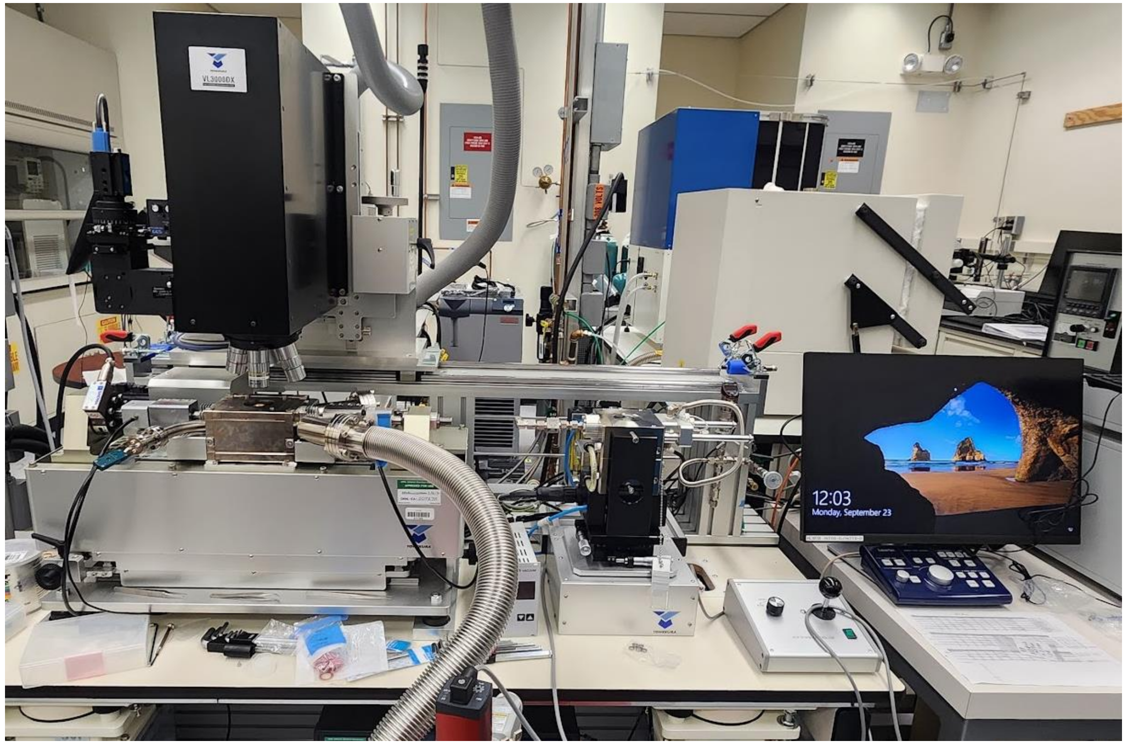


PCEA oxidation at 700°C



Same sample close to full oxidation

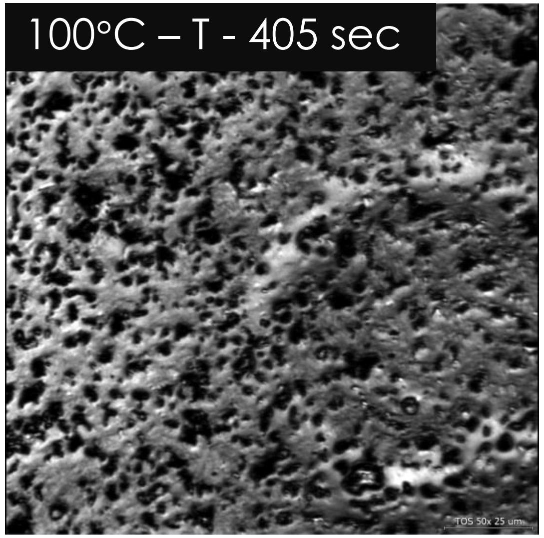
Simulation accident conditions in graphite – *In situ* oxidation monitored with Laser microscopy

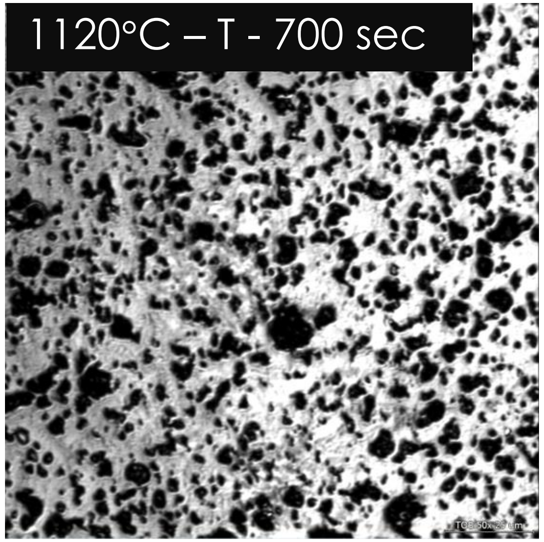


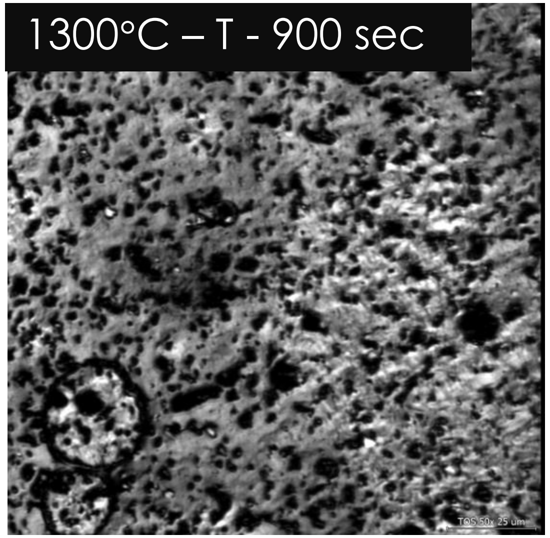
Laser microscope used for the experiments

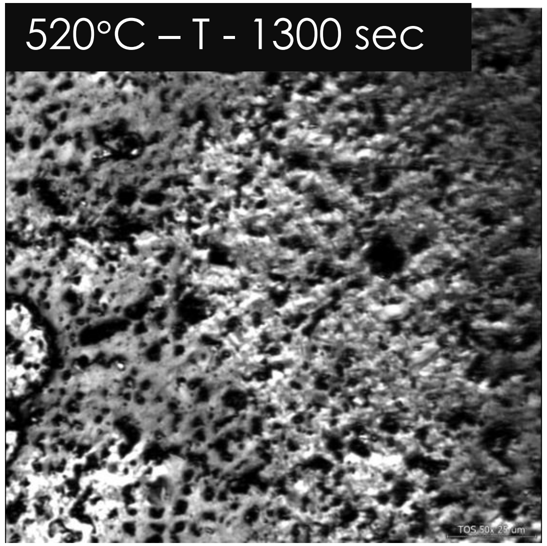
Temperature range 20 to 1600°C

The instrument will be enabled for steam and other environments



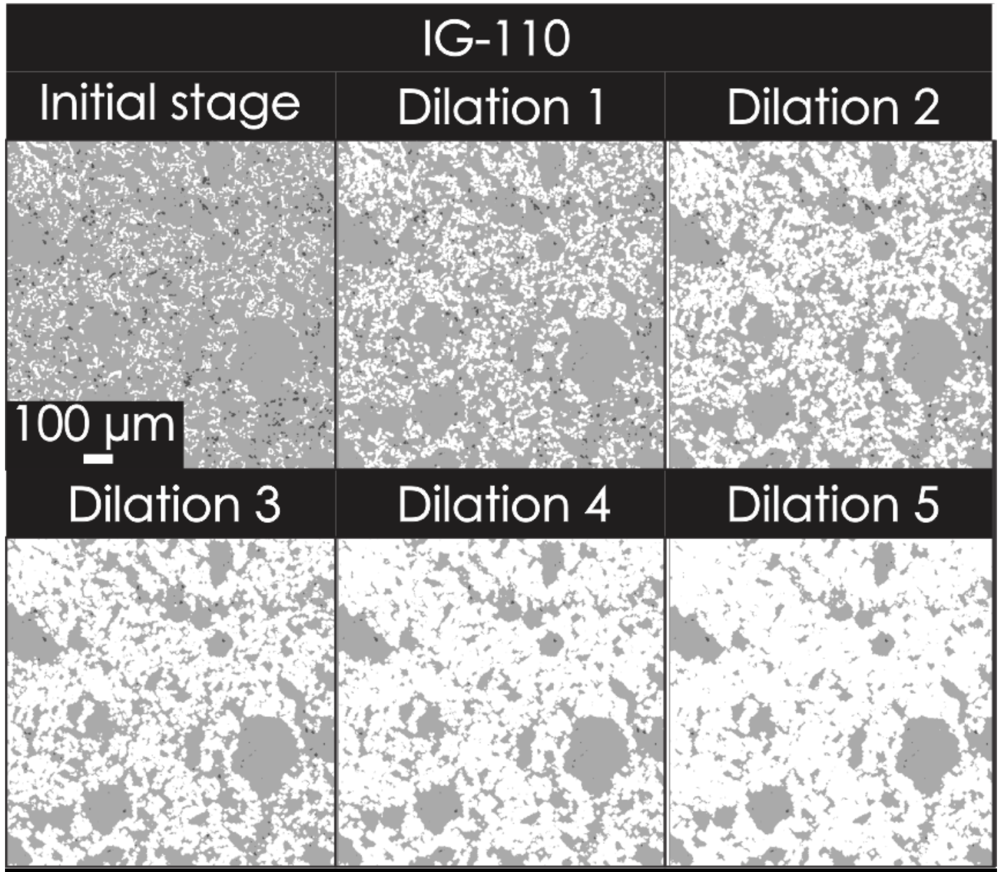






Oxidation of graphite with low concentrations of oxygen

XCT - Oxidation simulations

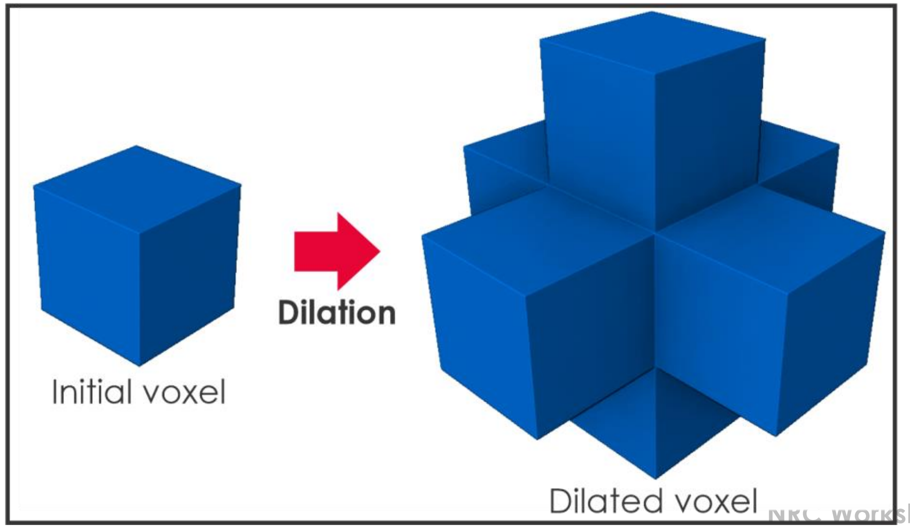


Uniform oxidation simulations

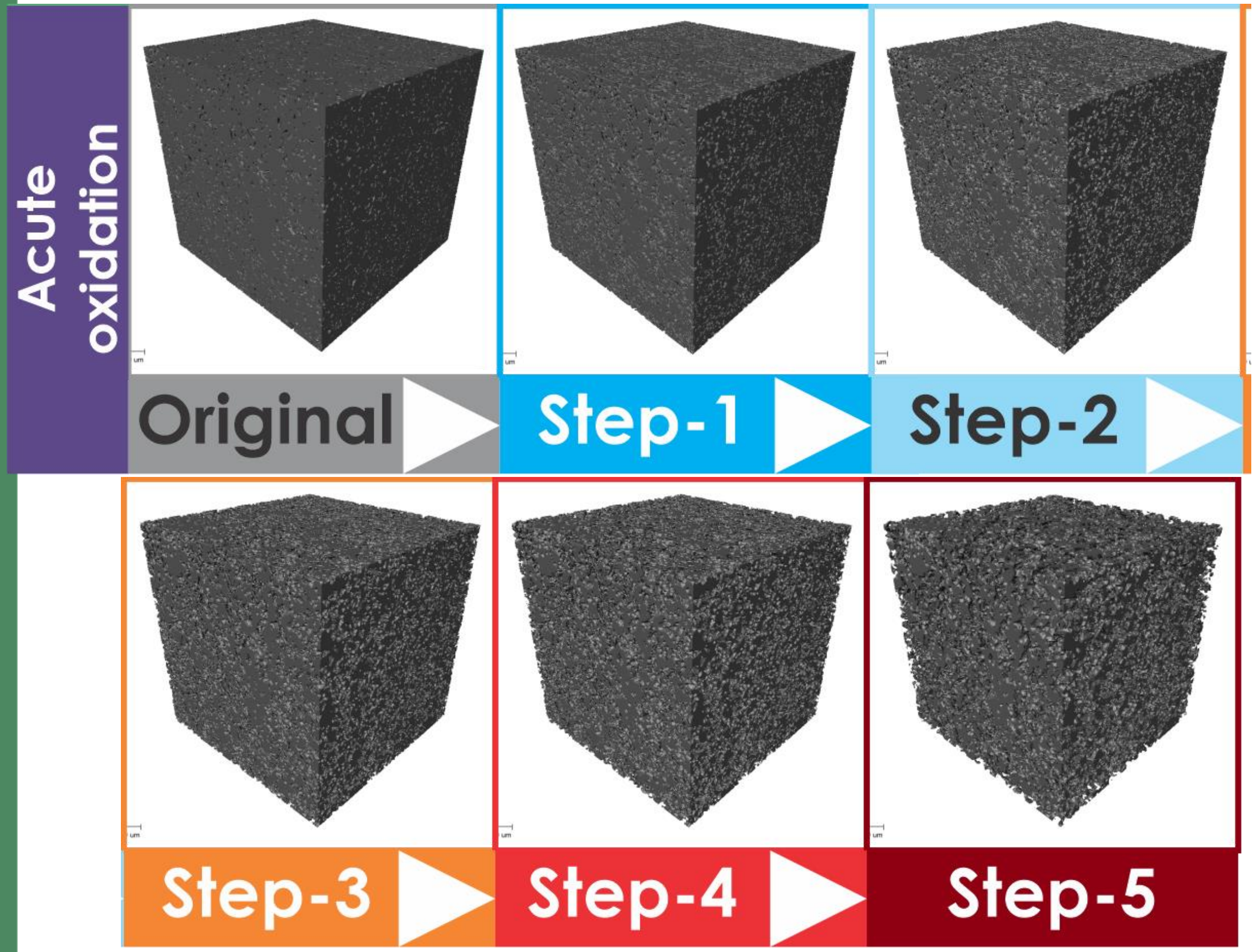
- The oxidation simulations are based on the gradual dilation of porosity

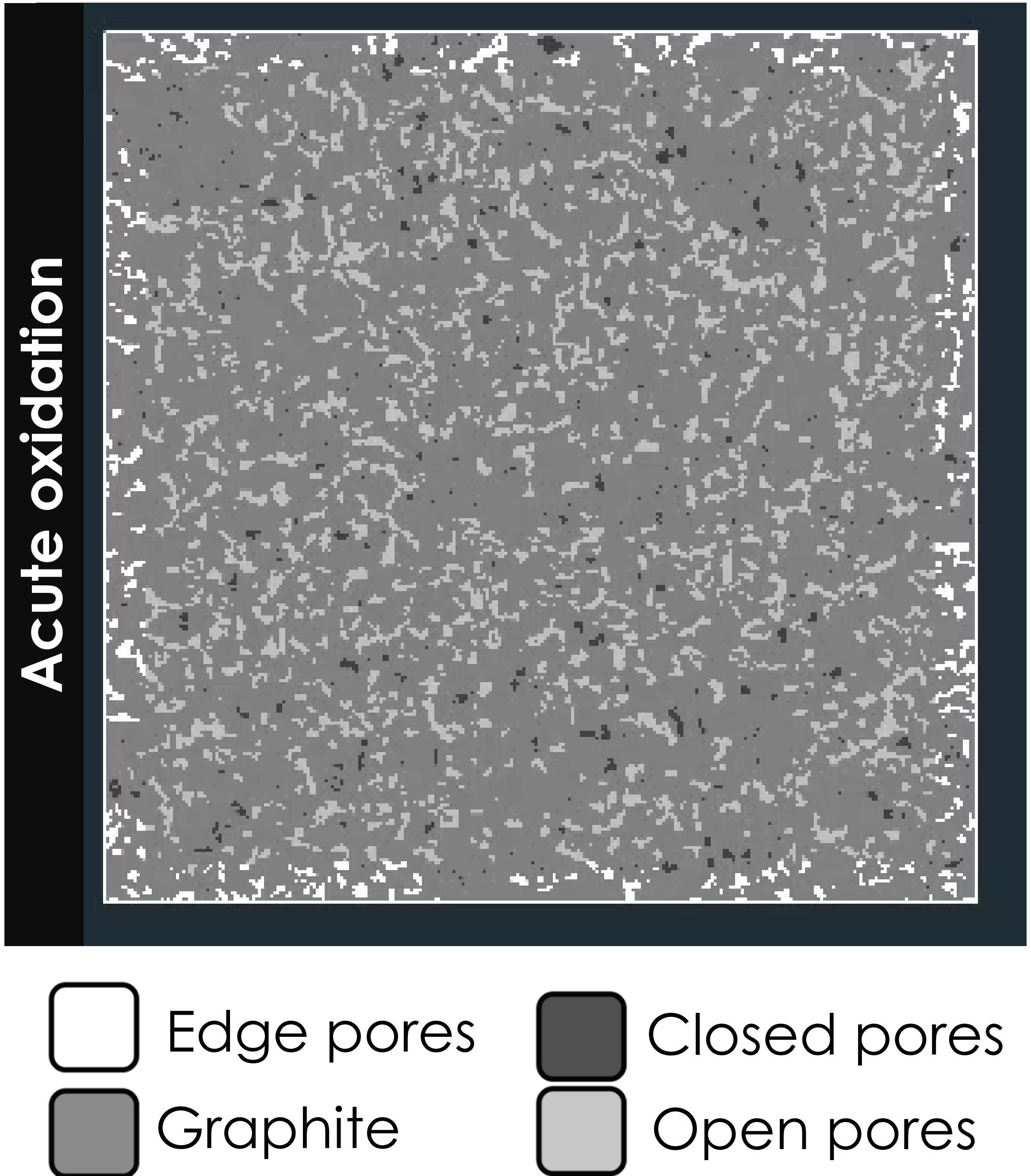
- For **acute oxidation**, the voxel dilation is limited to 10% of the pores near the surface

Ryan M. Paul, Jose D. Arregui-Mena, Cristian I. Contescu, Nidia C. Gallego, Effect of microstructure and temperature on nuclear graphite oxidation using the 3D Random Pore Model, Carbon, Volume 191, 2022



Oxidation evolution - Simulations





Summary

- The implementation of *microstructural characterization* can provide a better insight into oxidation effects of graphitic and ceramic materials

- XCT can be a new approach to characterize the oxidation effects in TRISO and matrix graphite. This data can be used for stress simulations

- Microstructural based simulations can be a valuable tool to predict the resistance of a material to oxidation accident conditions

Acknowledgments

- This work was supported by the U. S. Department of Energy, Office of Nuclear Energy, under the Advanced Reactor Technologies (ART) program. A portion of this work was supported by the Office of Nuclear Energy under DOE Idaho Operations Office Contract DE-AC07051D14517 as part of a Nuclear Science User Facilities experiment. This research used the resources of the Low Activation Materials Development and Analysis Laboratory (LAMDA), a DOE Office of Science research facility operated by the Oak Ridge National Laboratory (ORNL). Oak Ridge National Laboratory is managed by UT-Battelle under contract DE-AC05-00OR22725.
- This work was supported by the U.S. Department of Energy, Office of Nuclear Energy under DOE Idaho Operations Office Contract DE-AC07-051D14517 as part of a Nuclear Science User Facilities experiment
- "Diamond Light Source Ltd is acknowledged for providing beamtime (proposal MG31774-1) at the I13-2 Diamond Manchester Imaging beamline."

Storage and Transportation of TRISO and Metal Spent Nuclear Fuels

NRC's simulation capabilities supporting criticality, reactor physics, decay heat, and shielding for TRISO-particle fueled non-LWRs

Andy Bielen, Ph.D.

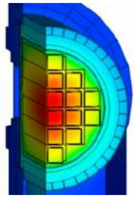
Office of Nuclear Regulatory Research
Division of Systems Analysis
Fuel & Source Term Code Development Branch

Objectives

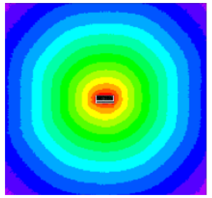
- *NRC's simulation capabilities supporting nuclear fuel safety for TRISO-particle fuel designs*
  - *Decay Heat*
  - *Neutron Multiplication & Criticality*
  - *Shielding and Radiation Protection*

- *Overview of data availability, gaps, and where additional data would be beneficial*

Nuclear Physics Considerations for TRISO/SFR Spent Fuel Safety



Decay Heat



Shielding and Radiation Protection

Neutron Multiplication and Criticality

NRC Regulations limit radiation dose under all phases of the fuel cycle:

- Direct radiation dose
- Radioactive material releases
- Inadvertent criticality

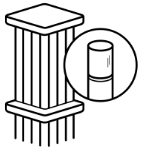
Computer codes used to determine:

- Irradiated fuel composition for nuclides that contribute to:
  - Direct radiation dose and dose from radioactive material releases
  - Decay heat
  - Determination of criticality safety (k_{eff})
- Radiation dose and k_{eff}

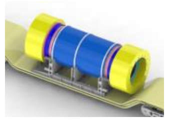
Codes must be validated against measured irradiated fuel data

10 CFR 20 – Radiation Protection

10 CFR 50/52 – Power Plants



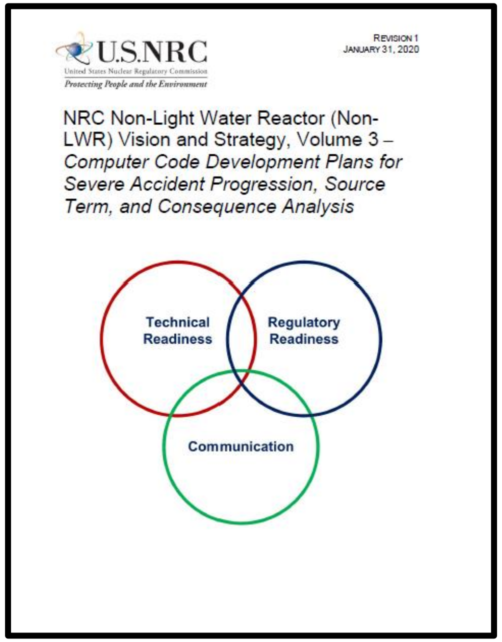
10 CFR 70 – Fuel Cycle Facilities



10 CFR 71 – Transportation

10 CFR 72 – SNF Storage

Non-LWR Source Term & Fuel Cycle Demonstration Projects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JANUARY 31, 2020

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 3 – *Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis*

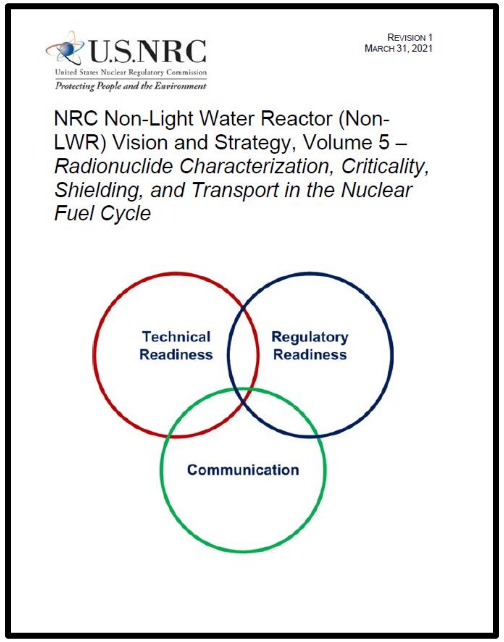
[Source Term]

NRC's comprehensive neutronics package

- Cross-section processing
- Decay heat analyses
- Criticality safety
- Radiation shielding
- Radionuclide inventory & depletion generation
- Reactor core physics

NRC's comprehensive severe accident progression and source term code

- Accident progression
- Thermal-hydraulic response
- Core heat-up, degradation, and relocation
- Fission product release and transport behavior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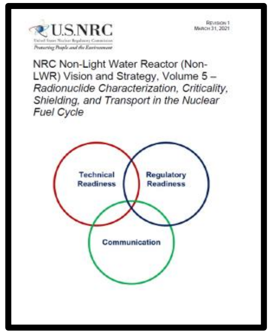
MARCH 31, 2021

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 5 – *Radionuclide Characterization, Criticality, Shielding, and Transport in the Nuclear Fuel Cycle*

[Fuel Cycle]

Non-LWR demonstration projects improve and validate SCALE & MELCOR for simulating non-LWRs for severe accident progression and fuel cycle analyses.

Decay Heat, Criticality Safety, and Radiation Shielding / Dose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MARCH 31, 2021

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 5 – *Radionuclide Characterization, Criticality, Shielding, and Transport in the Nuclear Fuel Cycle*

[*Fuel Cycle*]

Decay Heat

- SCALE/TRITON is used to generate specific ORIGEN reactor libraries; functionally bounds fuel enrichment and burnup.
- SCALE/ORIGAMI is used to obtain the spent fuel inventories; uses ORIGEN to compute detailed irradiated and decayed isotopic compositions.

Criticality Safety

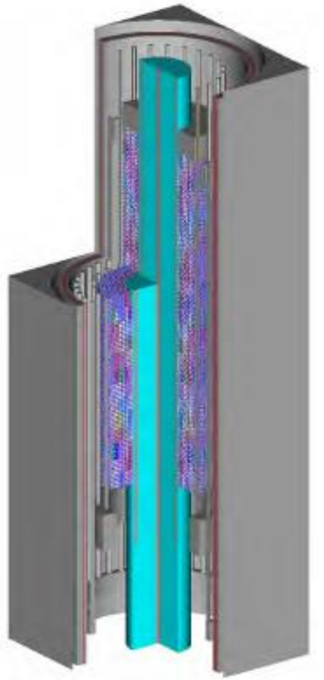
- SCALE/CSAS is used to perform criticality safety analyses. CSAS is a sequence that uses Monte Carlo transport codes KENO or Shift.
- Used to determine the multiplication factor of any system.

Shielding & Dose

- SCALE/MAVRIC is used to perform the shielding and dose analyses.
- *Uses the radiation source term & radionuclide inventories generated from SCALE/TRITON or SCALE/ORIGAMI.*

Non-LWR Reference Models

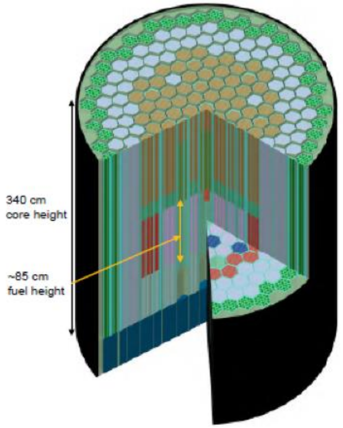
High-Temp. Gas Cooled Reactor



PBMR-400

- 400 MWth reactor, graphite moderated
- Helium-cooled & TRISO-particle pebble-fueled at 10 wt.% U-235
- Fuel discharged at high burnup (90 GWd/MTIHM)

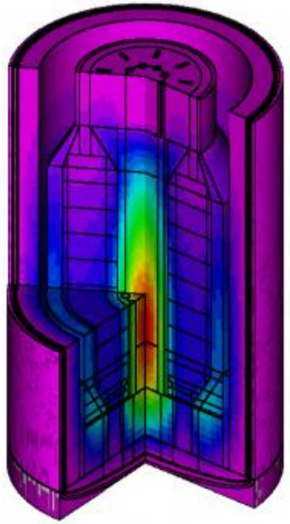
Sodium-Cooled Fast Reactor



ABTR

- 250 MWth pool-type reactor, utilizing metallic U / HT-9 fuel rods
- Reactor fueled with U-Pu-Zr fuel slugs
- Liquid sodium coolant

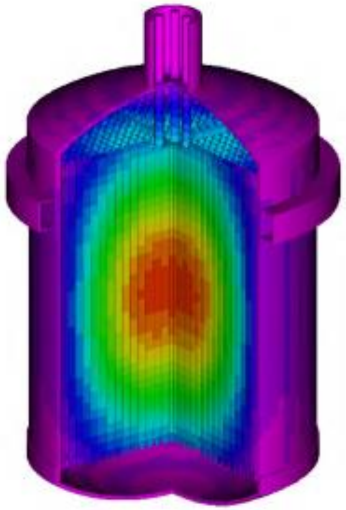
Molten Salt-Cooled Reactor



UCB Mk1 PB-FHR

- 236 MWth reactor at atmospheric pressures
- Flibe cooled & Pebble fueled (TRISO) at 19.9 wt.% U-235
- Online refueling

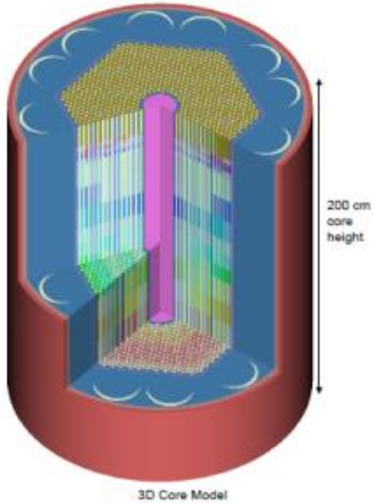
Molten Salt-Fueled Reactor



MSRE

- 10 MWth reactor, graphite moderated at near atmospheric pressures
- Reactor fueled with liquid dissolved fuel in molten salt (34.5 wt. % U-235)

Heat Pipe Reactor

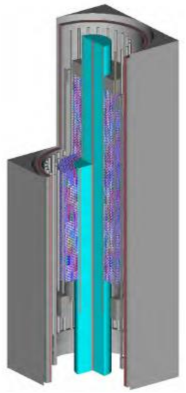


INL Design A

- 5 MWth with a 5-year operating lifetime
- 1,134 heat pipes fueled with UO_2 fuel (19.75 wt.% U-235)
- Reactivity controlled via control drums

Pebble Bed Reactor Workflows

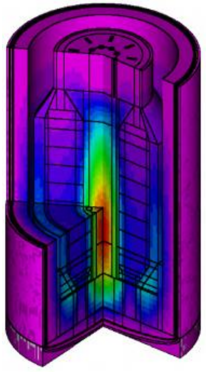
High-Temp. Gas Cooled Reactor



PBMR-400

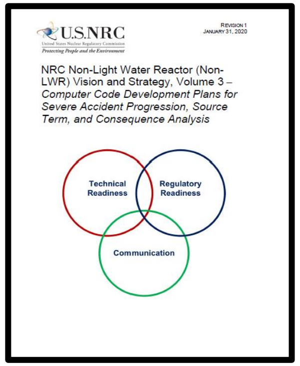
- 400 MWth reactor, graphite moderated
- Helium-cooled & TRISO-particle pebble-fueled at 10 wt.% U-235
- Fuel discharged at high burnup (90 GWd/MTIHM)

Molten Salt-Cooled Reactor



UCB Mk1 PB-FHR

- 236 MWth reactor at atmospheric pressures
- Flibe cooled & Pebble fueled (TRISO) at 19.9 wt.% U-235
- Online refueling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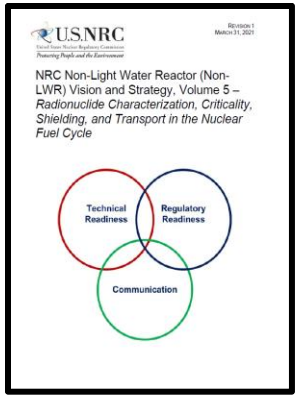
Revision 1
January 31, 2020

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 3 – *Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis*

Technical Readiness
Regulatory Readiness
Communication

Source Term


U.S.NRC

Revision 1
March 31, 2021

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 5 – *Radionuclide Characterization, Criticality, Shielding, and Transport in the Nuclear Fuel Cycle*

Technical Readiness
Regulatory Readiness
Communication

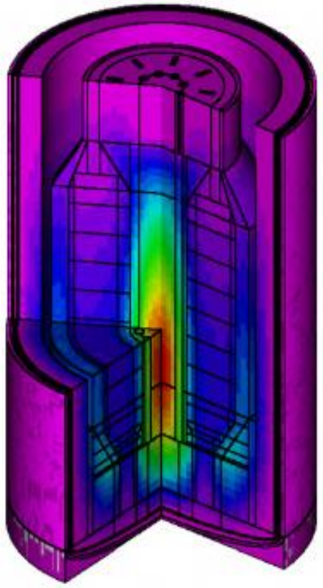
Fuel Cycle

Fuel Depletion, Decay Heat, & Radionuclide Inventory Generation

- Decay Heat
- Criticality Safety
- Shielding & Dose

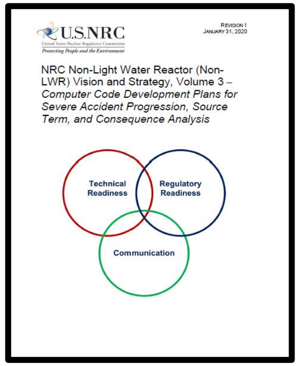
Fuel Depletion, Decay Heat, and Nuclide Inventory Generation

Molten Salt-Cooled Reactor



UCB Mk1 PB-FHR

- 236 MWth reactor at atmospheric pressures
- Flibe cooled & Pebble fueled (TRISO) at 19.9 wt.% U-235
- Online refueling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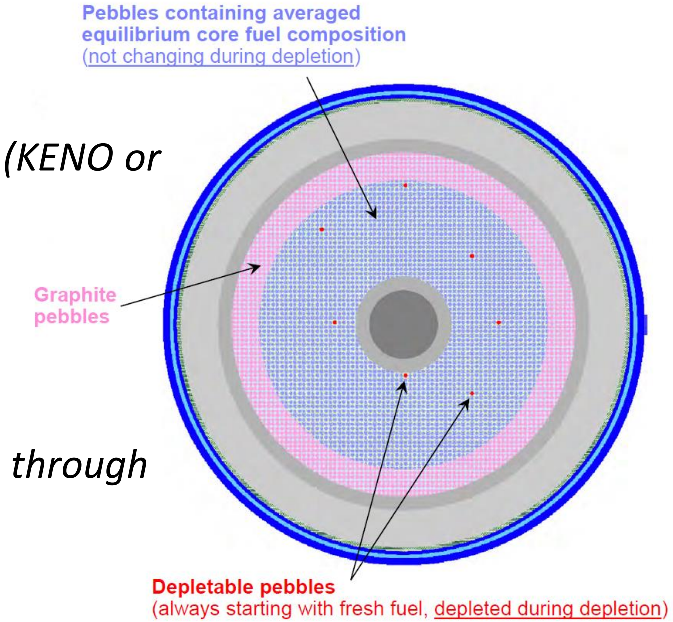
JANUARY 31, 2020

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 3 – *Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis*

Fuel Depletion, Decay Heat, & Radionuclide Inventory Generation

Source Term

- SCALE/TRITON used for fuel depletion
  - *Continuous energy Monte Carlo physics or MG methods available (KENO or Shift)*
  - *MG methods utilize SCALE's double-het methods*
  - *Equilibrium inventories generated via SLICE method*
    - *SCALE Leap-In Method for Cores at Equilibrium*
    - *Generates region-average fuel inventories*
    - *Accounts for average behavior of pebbles as they transverse through the core*
- *Radionuclide inventories used to support downstream analyses.*
  - *MELCOR for severe accident progression & radionuclide transport*
  - *ORIGAMI for decay heat analyses; utilizes the ORIGEN libraries from TRITON*



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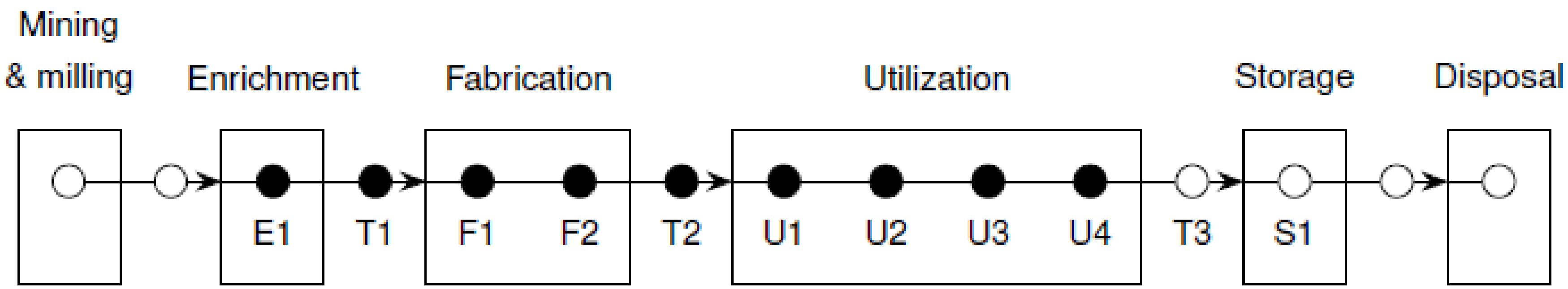
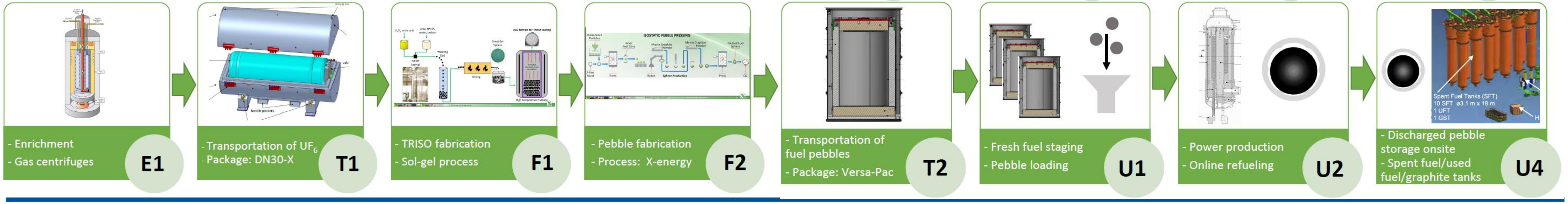
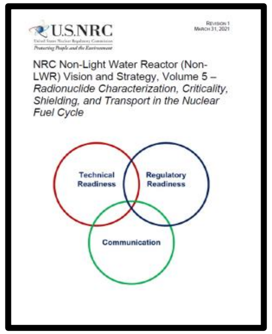
High Temperature Gas-Cooled Reactors Fuel Cycle



E1 – UF_6 enrichment
T1 – Transportation of UF_6 to fabrication facility
F1 – Fuel fabrication
F2 – Fuel assembly/pebble fabrication
T2 – Transportation of assemblies/pebbles/salt to plant
U1 – Fresh fuel staging/preparation/loading
U2 – Power production
U3 – Spent fuel pool/shuffle operations
U4 – On-site dry cask storage
T3 – Transportation of spent fuel to off-site storage
S1 – Off-site storage

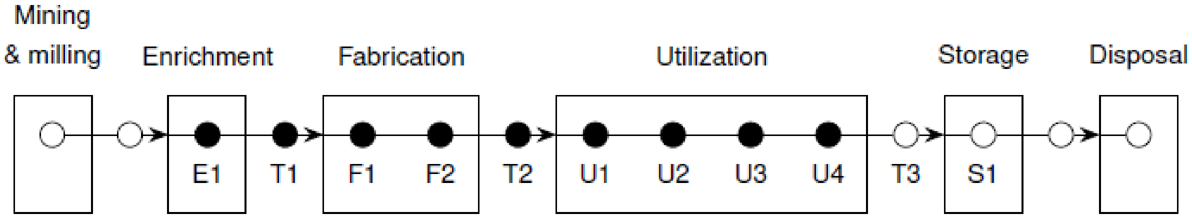


Decay Heat, Criticality Safety, and Radiation Shielding / Dose



Fuel Cycle

Decay Heat | Criticality Safety | Shielding & Dose



E1 – UF_6 enrichment
T1 – Transportation of UF_6 to fabrication facility
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T2 – Transportation of assemblies/pebbles/salt to plant
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T3 – Transportation of spent fuel to off-site storage
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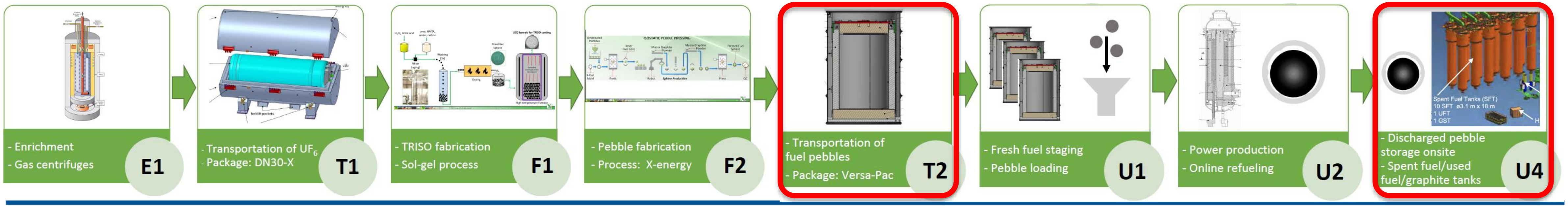
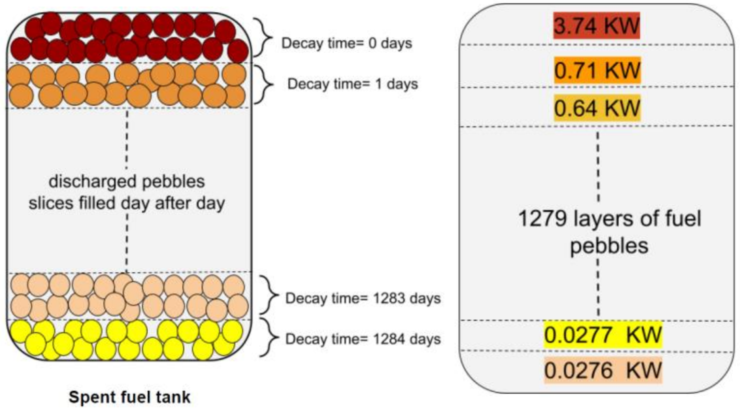
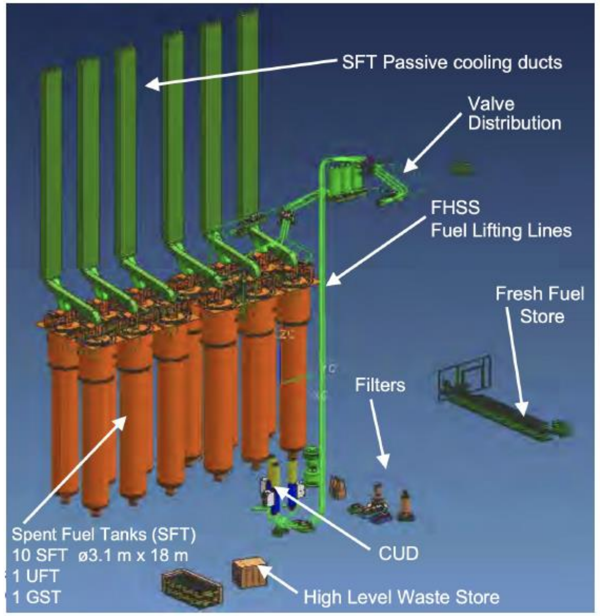


Decay Heat Analyses for TRISO-based Fuels

- *Vehicle / collision strike with a spent nuclear fuel storage tank loaded with spent TRISO-pebbles.*
  - *Once burnup limits are reached, pebble is moved into a spent fuel tank, with a capacity of holding ~620K pebbles.*
  - *Discharge rate – 483 pebbles / day; 1,284 days to fill spent fuel tank.*





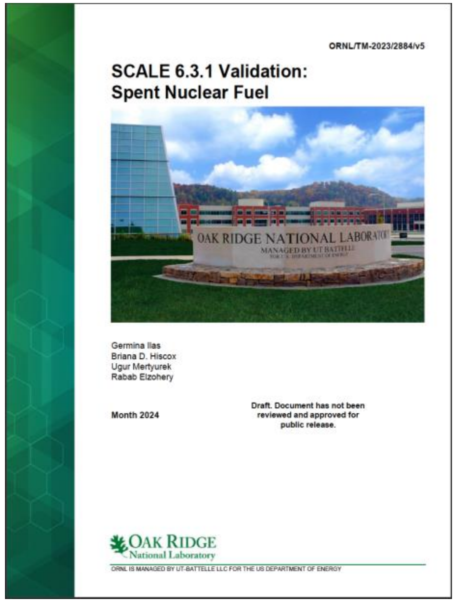
PBMR-400 FHSS with pebble storage tanks*

- *Determine average spent fuel pebble inventory after discharge*
  - Leveraged from the non-LWR demonstration source term work (for HTGR)
  - TRITON & ORIGAMI used for generating inventories & performing decay-correction from ORIGEN reactor libraries

- *Radionuclide inventories used to support downstream analyses.*
  - MELCOR for severe accident progression & radionuclide transport
  - MAVRIC for shielding & dose analyses

NRC's Computer Codes and Validation

SCALE Validation in Four Major Areas (Criticality Safety, Radiation Shielding, Reactor Physics, and Spent Fuel Inventory)


ORNL/TM-2023/2884/v5

SCALE 6.3.1 Validation: Spent Nuclear Fuel

Germina Ilas
Briana D. Hiscox
Ugur Mertyurek
Rabab Elzohery

Month 2024

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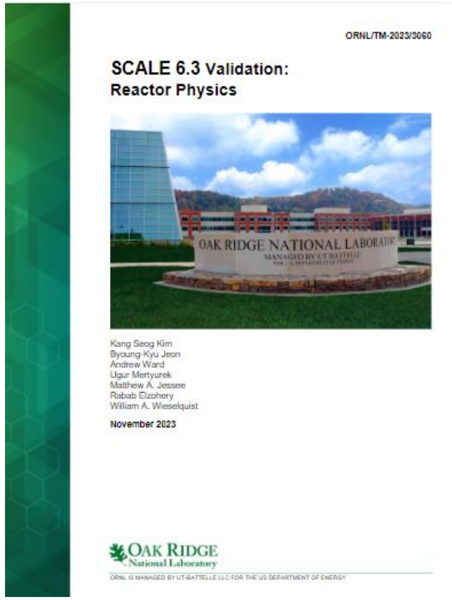
ORNL/TM-2023/3060

SCALE 6.3 Validation: Reactor Physics

Kang Seog Kim
Byoung-Kyu Jeon
Andrew Ward
Ugur Mertyurek
Matthew A. Jessee
Rabab Elzohery
William A. Wieselquist

November 2023

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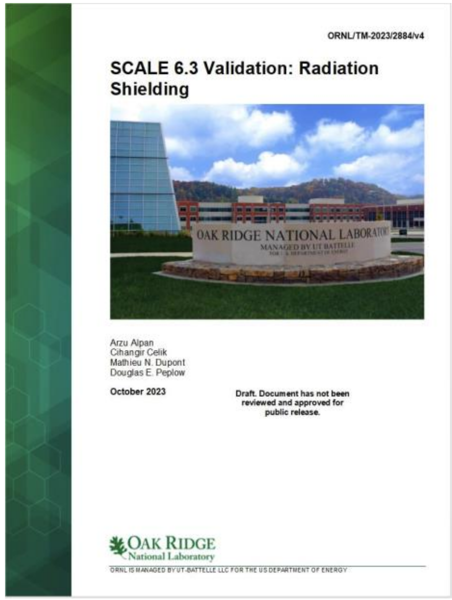
ORNL/TM-2023/2884/v4

SCALE 6.3 Validation: Radiation Shielding

Arzu Alpan
Cihangir Celik
Mathieu N. Dupont
Douglas E. Peplow

October 2023

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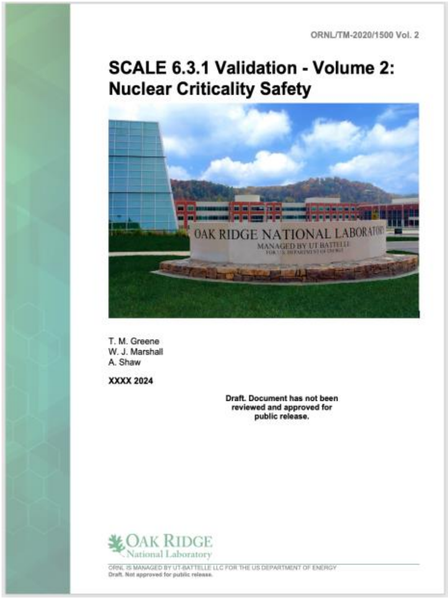
ORNL/TM-2020/1500 Vol. 2

SCALE 6.3.1 Validation - Volume 2: Nuclear Criticality Safety

T. M. Greene
W. J. Marshall
A. Shaw

XXXX 2024

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SFRs

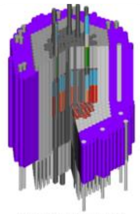

Figure 5.7. EBR-II SCALE model [62].

Table 5.4. Eigenvalue results for the high-fidelity EBR-II benchmark.

	k_{eff}	σ	Δk_{eff}^{a} (pcm)
Benchmark value [7]	1.00927	± 0.00618	reference
SCALE 6.3.1/KENO-VI CE ENDF/B-VII.1	1.00722	± 0.00010	-205 (± 618)
SCALE 6.3.1/KENO-VI CE ENDF/B-VIII.0	1.00691	± 0.00013	-236 (± 618)

[a] Calculated as 10^5(k-eff$_{calculated}$ – k-eff$_{benchmark}$).

HTGRs

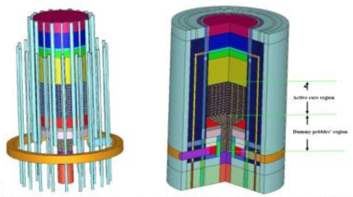

Figure 5.1. Illustration of HTR-10 benchmark model details.¶ (Channels in reflector regions [left], full reactor model [right]; images not to scale) [35].

Table 5.1. Eigenvalue results for high-fidelity HTR-10 benchmark.

	k_{eff}	σ	Δk_{eff}^{a} (pcm)
Benchmark value [6]	1.00000	0.00370	reference
SCALE/KENO-VI CE ENDF/B-VII.1	1.00303 ± 0.00041	0.99661 ± 0.00031	303 ± 370
SCALE/KENO-VI CE ENDF/B-VIII.0	1.00604 ± 0.00027	0.99919 ± 0.00026	604 ± 370
SCALE/KENO-VI 252-group ENDF/B-VII.1	1.00265 ± 0.00031	0.99595 ± 0.00025	265 ± 370
SCALE/KENO-VI 252-group ENDF/B-VIII.0	1.00376 ± 0.00027	0.99746 ± 0.00025	376 ± 370

[a] Calculated as 10^5(k-eff$_{calculated}$ – k-eff$_{benchmark}$).

MSRs

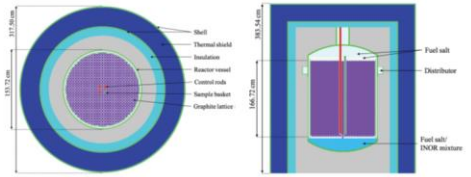

Figure 5.5. Cross-sectional illustrations of MSRE benchmark models [60].

Table 5.3. Eigenvalue results for the high-fidelity MSRE benchmark.

	k_{eff}	σ	Δk_{eff}^{a} (pcm)
Benchmark value	0.99978	± 0.00420	reference
SCALE 6.3.1/Shift CE ENDF/B-VII.1	1.019016	± 0.00010	1924 (± 420)
SCALE 6.3.1/Shift CE ENDF/B-VIII.0	1.021833	± 0.00010	2205 (± 420)

[a] Calculated as 10^5(k-eff$_{calculated}$ – k-eff$_{benchmark}$).

SCALE has been heavily validated for standard fuel designs in LWRs. SCALE 6.3 validation efforts are underway to validate SCALE for several advanced non-LWR systems.

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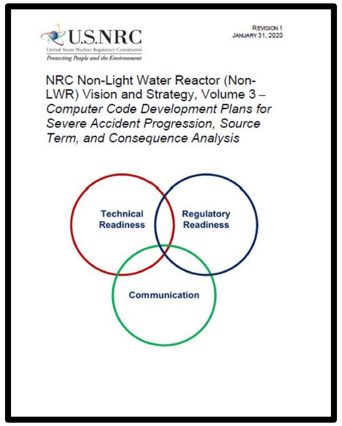
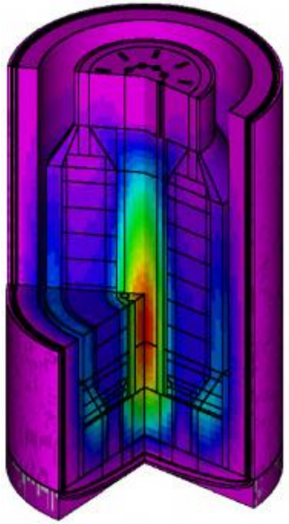
Applications of non-LWR Demonstration Project - Kairos Hermes Construction Permit Application Support



Source Term

Molten Salt-Cooled Reactor



UCB Mk1 PB-FHR

- 236 MWth reactor at atmospheric pressures
- Flibe cooled & Pebble fueled (TRISO) at 19.9 wt.% U-235
- Online refueling

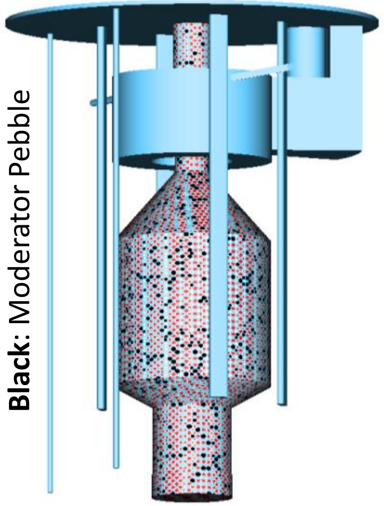
- *Generated a library of well-tested & demonstrated non-LWR reference plant models in SCALE & MELCOR.*
- *Models can be heavily leveraged to support licensing reviews.*

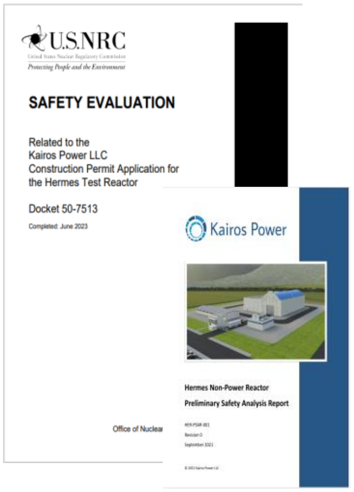
- *Leveraged the FHR model to supportrt the licensing review of Hermes I*
  - *Similarities between the UCB Mk1 & Hermes I noted*
  - *Leveraged existing models & insights from non-LWR demonstration project*
- *SCALE and MELCOR used for analyzing various scenarios (e.g., loss of forced circulation, accidental control rod withdrawal)*

Kairos Hermes I

- 35 MWth reactor at atmospheric pressures
- Flibe cooled & Pebble fueled (TRISO) at 19.75 wt.% U-235
- Online refueling

Blue: FLiBe
Red: Fuel Pebble
Black: Moderator Pebble





Non-LWR demonstration project was instrumental in an effective and efficient review of a first of a kind non-LWR.

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For More Information

SCALE/MELCOR non-LWR source term demonstration project	

• Heat-pipe reactor workshop
  ◦ Slides
  ◦ Video Recording
• SCALE report
• MELCOR report
	June 29, 2021

• High-temperature gas-cooled reactor workshop
  ◦ Slides
  ◦ Video Recording
• SCALE report
• MELCOR report
	July 20, 2021

• Fluoride-salt-cooled high-temperature reactor workshop
  ◦ Slides
  ◦ Video Recording
• SCALE report
• MELCOR report
	September 14, 2021

• Molten-salt-fueled reactor workshop
  ◦ Slides
  ◦ Video Recording
• SCALE report
• MELCOR report
	September 13, 2022

• Sodium-cooled fast reactor workshop
  ◦ Slides
  ◦ Video Recording
• SCALE report
• MELCOR report
	September 20, 2022

SCALE/MELCOR non-LWR fuel cycle demonstration project	

• High-temperature gas-cooled reactor fuel cycle workshop
  ◦ Slides
  ◦ Video Recording
  ◦ SCALE Report
  ◦ MELCOR Report
	February 28, 2023

• Sodium-cooled fast reactor fuel cycle workshop
  ◦ Slides
  ◦ Video Recording
  ◦ SCALE Report
  ◦ MELCOR Report
	September 20, 2023

• Molten salt reactor fuel cycle workshop
  ◦ Slides
  ◦ Video Recording
  ◦ SCALE Report
  ◦ MELCOR Report
	July 11, 2024

• Microreactor fuel cycle workshop	Coming in 2025
• Non-LWR Fuel Cycle Scenarios for SCALE and MELCOR Modeling Capability Demonstration ◦ Report	December 15, 2023

Public workshop videos, slides, reports at advanced reactor source term webpage

TRISO AND METAL SPENT NUCLEAR FUEL (SNF) CANISTER DECAY HEAT

LAURA PRICE

Principal Member of Technical Staff
Sandia National Laboratories

U.S. DEPARTMENT OF ENERGY | Office of NUCLEAR ENERGY | SPENT FUEL & HIGH-LEVEL WASTE DISPOSITION

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December 3-5, 2024

DISCLAIMER

This is a technical presentation that does not take into account contractual limitations or obligations under the Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste (Standard Contract) (10 CFR Part 961).

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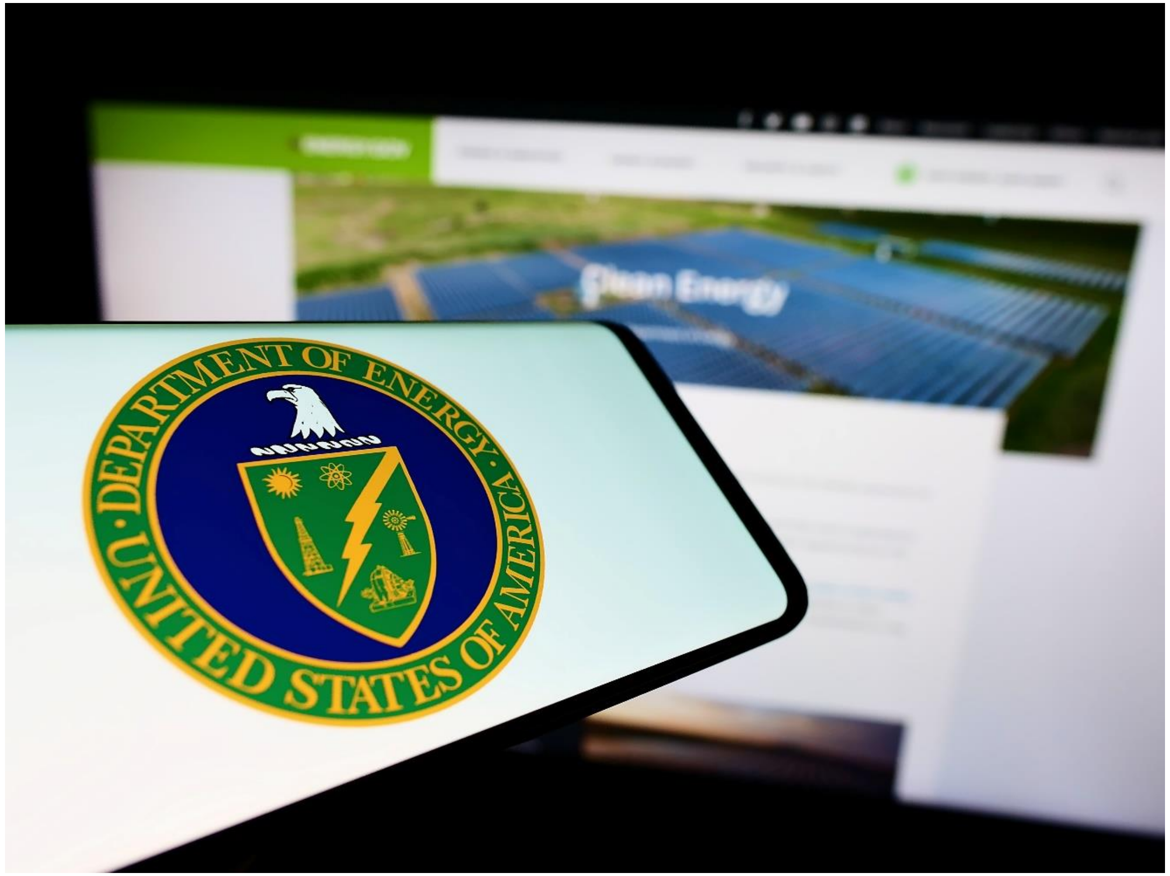
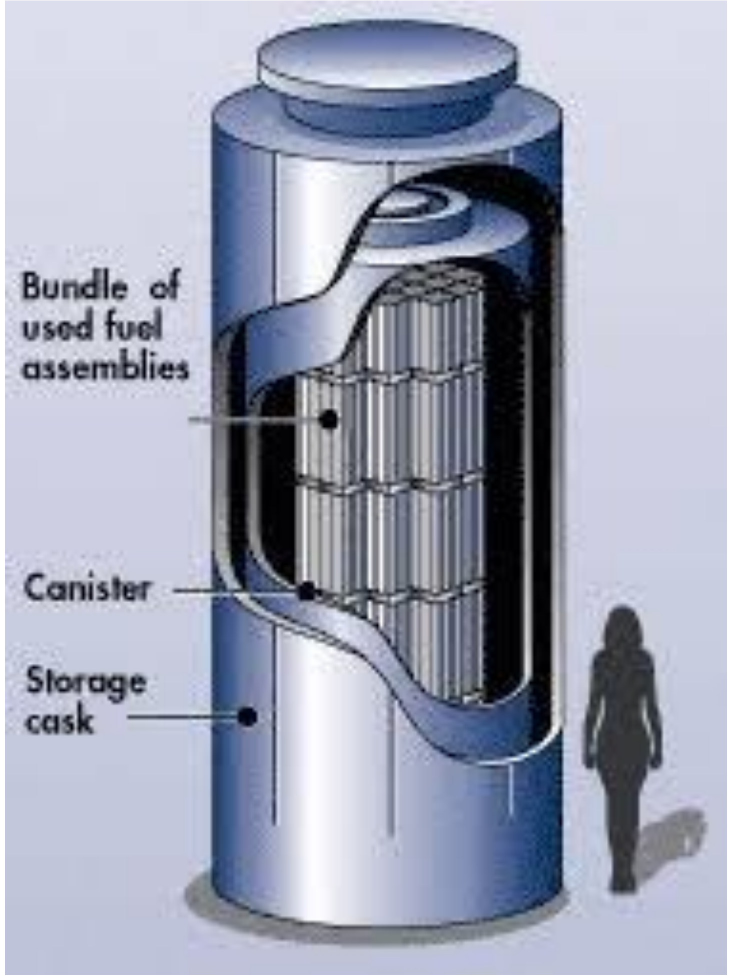
This presentation reflects technical work which could support future decision making by the U.S. Department of Energy (DOE or Department). No inferences should be drawn from this presentation regarding future actions by DOE, which are limited both by the terms of the Standard Contract and Congressional appropriations for the Department to fulfill its obligations under the Nuclear Waste Policy Act including licensing and construction of a spent nuclear fuel repository.



COMPARISON OF TYPICAL LIGHT WATER REACTOR SPENT NUCLEAR FUEL (SNF) WITH TRISO AND METALLIC SNF

- Hoffman et al. (2024) compared the characteristics of three different types of proposed irradiated TRISO fuels and one type of proposed irradiated metallic fuel (sodium free) to those of “typical” LWR SNF.
- Characteristics considered
  - Physical dimensions
  - Isotopic composition and fissile inventory
  - **Decay heat of SNF in Canisters – drives loading limits for storage, transportation, and disposal**
  - Radiation source strength
  - A2 value
  - Energy generation
- Comparison made on a basis of what could fit in a 37-PWR canister; done for comparison purposes only

TYPICAL 37-PWR CANISTER SYSTEM



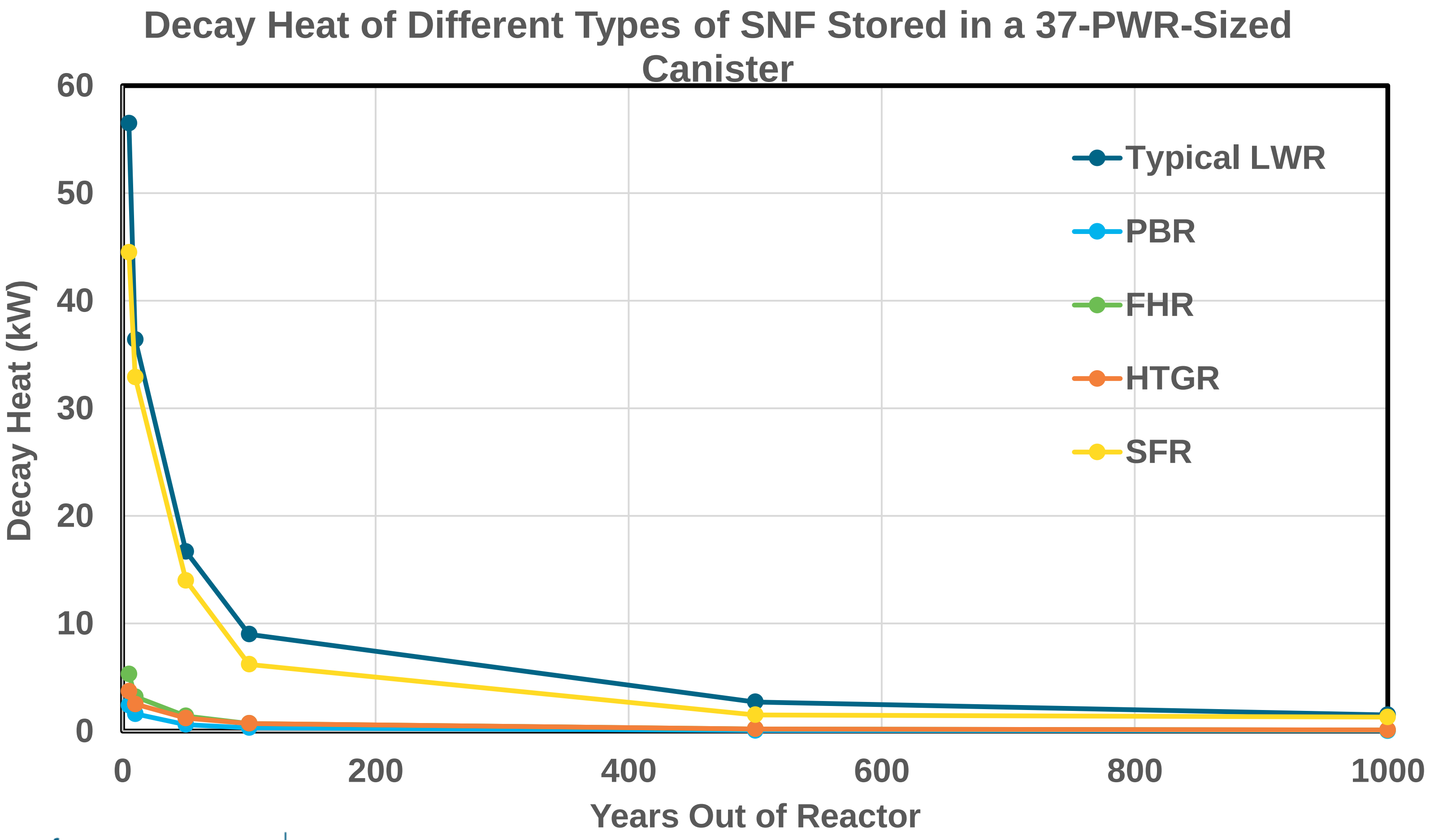
Picture on left: Courtesy of the NRC
Above picture: Courtesy of Holtec International

SELECTED SNF PARAMETERS

	Typical LWR	Pebble Bed Reactor (PBR)	Fluoride-salt-cooled high-temperature reactor (FHR)	Prismatic block high-temperature gas reactor (HTGR)	Metallic sodium-cooled fast reactor (SFR)
Heavy metal loading in 37-PWR-size canister (initial MTU)	19.9	0.3	0.5	0.6	8.3
Average proposed discharge burnup (GWd/MTU)	50	165	180	120	147.3

Source: Hoffman et al. (2023)

SNF CANISTER DECAY HEAT



REFERENCES

Hoffman, E., Kim, T.K., and Price, L., 2024. “Characteristics of Potential Significance in Waste Management from HALEU Spent Fuel – 24226,” Waste Management Conference, March, 2024,Phoenix, AZ, USA.

LEARN MORE

Office of Spent Fuel & High-Level Waste Disposition

energy.gov/ne/office-spent-fuel-and-high-level-waste-disposition

MODELING CAPABILITIES FOR TRISO AND METALLIC SPENT NUCLEAR FUEL

GORDON PETERSEN
SPENT FUEL ANALYST

2024 Workshop on Storage and Transportation of TRISO and Metal Spent Nuclear Fuels

U.S. DEPARTMENT OF ENERGY | Office of NUCLEAR ENERGY | SPENT FUEL & HIGH-LEVEL WASTE DISPOSITION

December 3-5, 2024

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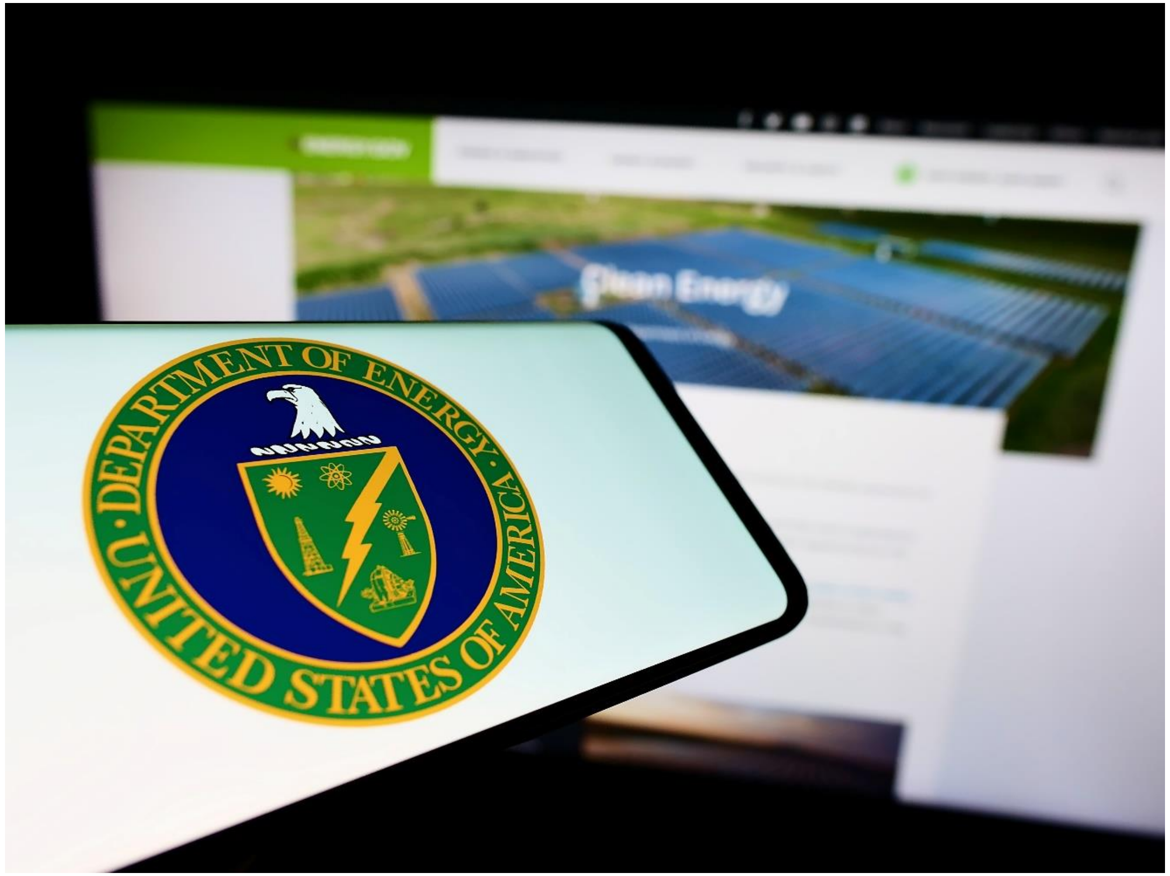


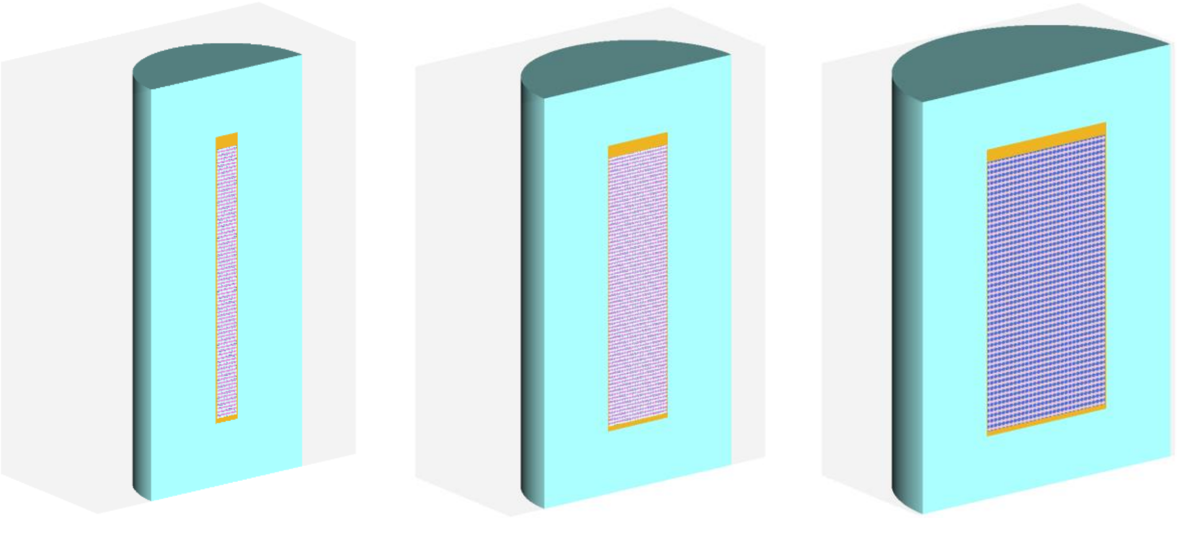
MODELING LIGHT-WATER REACTOR SPENT NUCLEAR FUEL

- Light-water reactor (LWR) spent nuclear fuel (SNF) has been packaged and modeled for decades
- Criticality Evaluations
  - Boiling-water reactor SNF models peak reactivity
  - Pressurized-water reactor utilizes burnup credit
  - Typical uncertainty between 500–800 per cent mille (pcm) for LWR [1]
- Dose/Decay Heat Evaluations
  - Decay heat uncertainty ~2% [2]

Dry storage systems in the U.S.

MODELING TRISO SNF

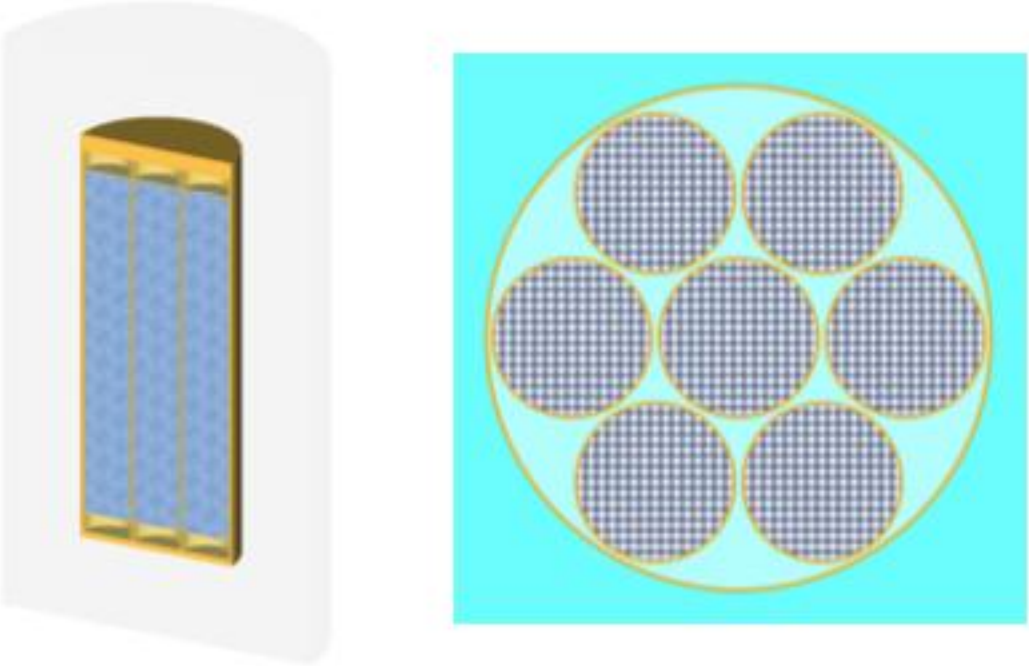
- Performed potential packaging analyses for TRi-structural ISOtropic (TRISO) SNF
- Criticality evaluations
  - Top contributors to uncertainty [1,2]
    - ^{235}U, ^{238}U, and graphite*****
  - Uncertainty expected to be less than 1,000 pcm [1,2] for SNF packages
- Dose/decay heat evaluations
  - TRISO SNF is expected to have lower dose and decay heat compared to LWR SNF.
- No significant modeling challenges expected in packaging TRISO SNF for extended storage and transportation.



12 in. Canister

36 in. Canister

72 in. Canister



Multiple DOE Standard Canisters in a large diameter canister

* Due to the large amount of graphite and other carbon non-fuel layers

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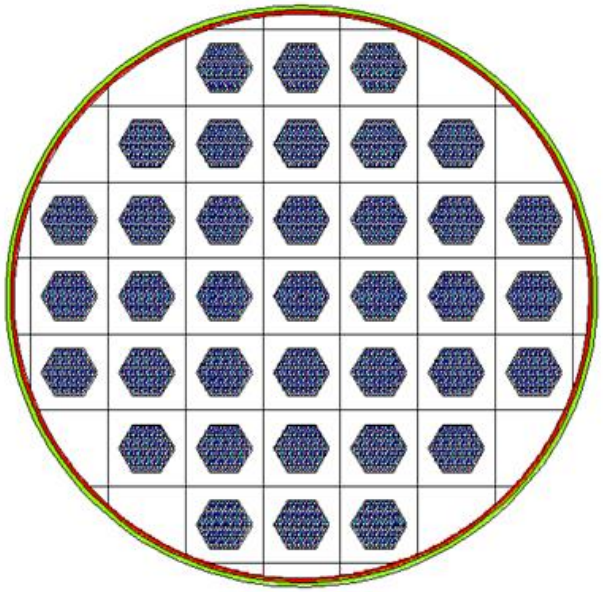
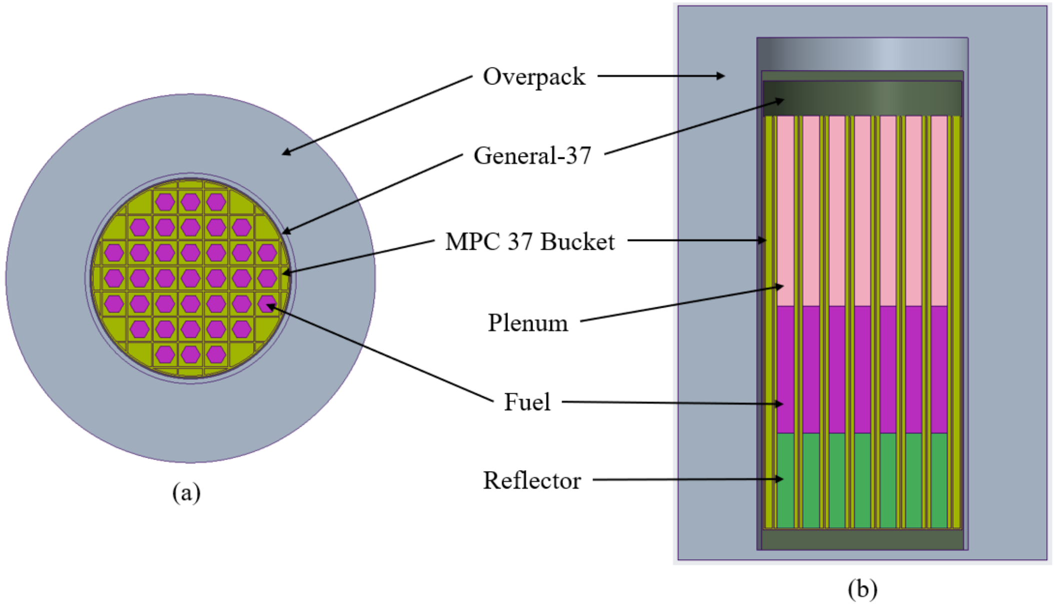
MODELING METALLIC SNF

- Performed potential packaging analyses for metallic SNF
- Criticality evaluations
  - Top contributors to uncertainty [1]
    - ^{235}U, ^{238}U, and ^{56}Fe*
    - Uncertainty of sodium fast reactor expected to be between 1000-1500 pcm.
- Dose/decay heat evaluations
  - Decay heat and dose largely depends on configuration.
- No significant modeling challenges expected in packaging metallic SNF for extended storage and transportation.

Radial and axial representations for metallic SNF from a sodium fast reactor.





* Due to the structural materials being composed of iron-based alloys

REFERENCES

1. Bostelmann, F., Illas, G., Wieselquist, W. "Key Nuclear Data Impacting Reactivity in Advanced Reactors." ORNL/TM-2020/1557, June 2020.

2. Ebiwonjumi, B., Kong, C., Zhang, P., Cherezov, A., Lee, D. "Uncertainty Quantification of PWR Spent Fuel Due to Nuclear Data and Modeling Parameters." Nuclear Engineering and Technology, Volume 53, Issue 3, 2021.

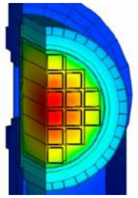
3. Wing, J., Maldonado, G. I., Petersen, G., Joseph, R. "Uncertainty Quantification for Pebble Bed Reactor Fuels Burnup Credit." Transactions, Volume 130, Number 1, June 2024, Pages 168-171.

LICENSING EXPERIENCE WITH TRISO SPENT FUEL – A HISTORICAL PERSPECTIVE: FORT ST. VRAIN INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)

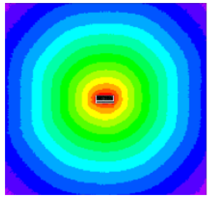
WORKSHOP ON STORAGE AND TRANSPORTATION OF TRISO AND METAL SPENT NUCLEAR FUELS DECEMBER 3-5, 2024

Drew Barto
U.S. NRC
NMSS Division of Fuel Management

Nuclear Physics Considerations for TRISO/SFR Spent Fuel Safety



Decay Heat



Shielding and Radiation Protection

Neutron Multiplication and Criticality

NRC Regulations limit radiation dose under all phases of the fuel cycle:

- Direct radiation dose
- Radioactive material releases
- Inadvertent criticality

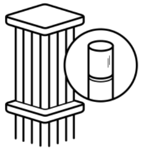
Computer codes used to determine:

- Irradiated fuel composition for nuclides that contribute to:
  - Direct radiation dose and dose from radioactive material releases
  - Decay heat
  - Determination of criticality safety (k_{eff})
- Radiation dose and k_{eff}

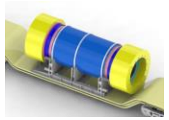
Codes must be validated against measured irradiated fuel data

10 CFR 20 – Radiation Protection

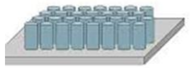
10 CFR 50/52 – Power Plants



10 CFR 70 – Fuel Cycle Facilities



10 CFR 71 – Transportation



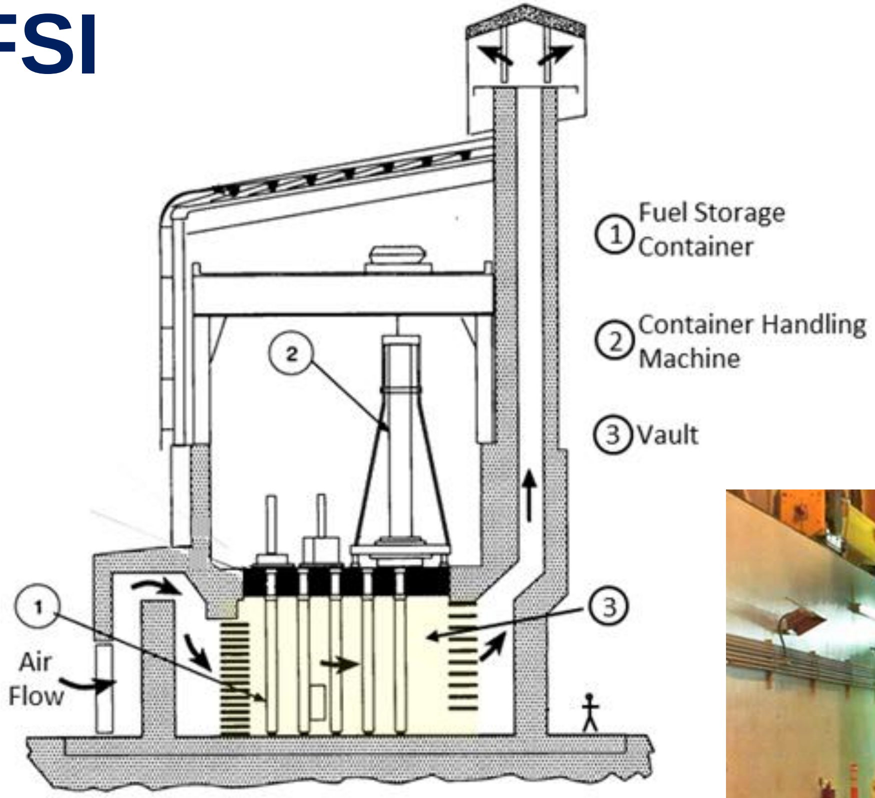
10 CFR 72 – SNF Storage

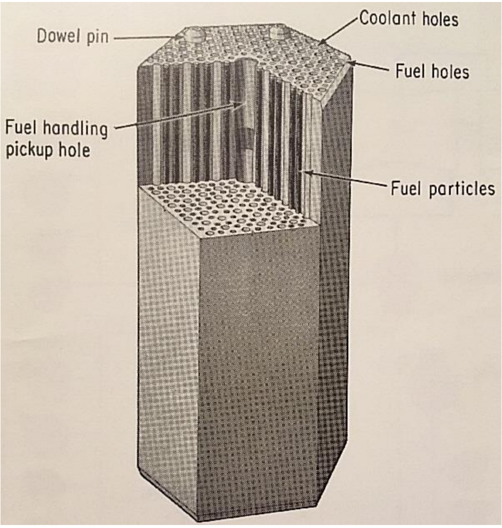
Fort St. Vrain ISFSI

- FSV reactor was a 330 Mwe TRISO fuel HTGR that operated from 1979 – 1989
- HEU (93.5 wt%) and thorium carbide fuel in cylindrical graphite compacts
- Compacts inserted into hexagonal graphite fuel elements
- ISFSI licensed by NRC in 1991

Fort St. Vrain ISFSI

- ISFSI is a "Modular Vault Dry Store" system
- Fuel stored in cylindrical steel fuel storage containers
- Each storage location holds up to six spent fuel elements
- Air cooled





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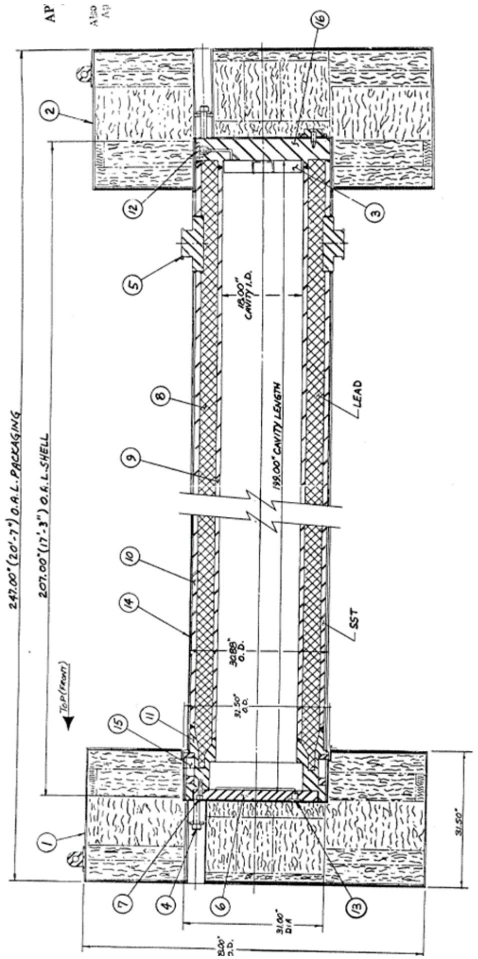
Fort St. Vrain Fuel Transportation

- USA/9253/B(U)F-96: TN-FSV Package
- Originally certified in 1993
- Ships one FSV fuel storage container
- Legal weight truck package – 47,000 lbs. loaded



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Key Messages

- TRISO spent fuel not entirely new, and has been licensed in storage and transportation
- Licensed using old codes and data to determine criticality safety, decay heat, and radiation dose, with additional margins for uncertainties in codes and data
- Today’s codes well capable of evaluating TRISO fuel in prismatic and pebble configurations
- Margins can be reduced when code validation data becomes more available

Storage and Transportation of TRISO and Metal Spent Nuclear Fuels

NRC's simulation capabilities supporting materials performance for TRISO-particle fueled non-LWRs

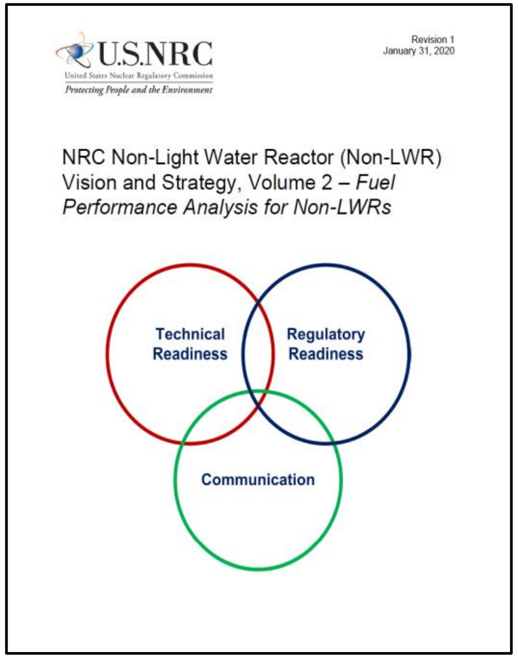
James Corson, Ph.D.

Office of Nuclear Regulatory Research
Division of Systems Analysis
Fuel & Source Term Code Development Branch

Objectives

- *NRC's simulation capabilities for modeling TRISO-particle fuel forms*
- *Overview of data availability, gaps, and where additional data would be beneficial*

Non-LWR Fuel Performance Analyses



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NRC's Fuel Analysis under Steady-state and Transients (FAST) code

- *Models the thermal-mechanical response of nuclear fuel*
- *Is used for normal operations, anticipated operational occurrences, accident conditions, and spent fuel storage*
- *Is used for LWR & non-LWR fuel types*

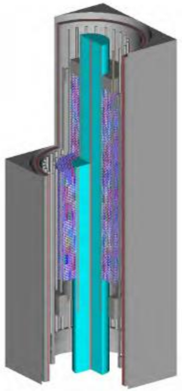
Non-LWR demonstration project for fuel performance

- *Developed new models for TRISO and metallic fuel designs*
- *Performed assessments & validation activities with available experimental data*

Non-LWR demonstration project for fuel performance has improved and validated FAST for simulating non-LWR fuel designs, including metallic fuel designs for use in SFRs.

Non-LWR Fuel Designs & Phenomena Relevant to Safety

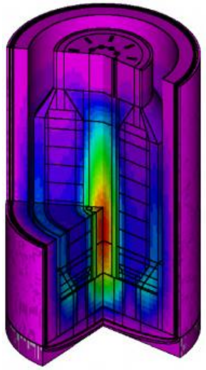
High-Temp. Gas Cooled Reactor



HTGRs

- Pebble-bed core, fueled with TRISO-pebbles, moderated with graphite and cooled with helium.
- Operated HTGRs in the US Peach Bottom Unit 1 and Fort St. Vrain
- *Potential Future Designs - X-energy*

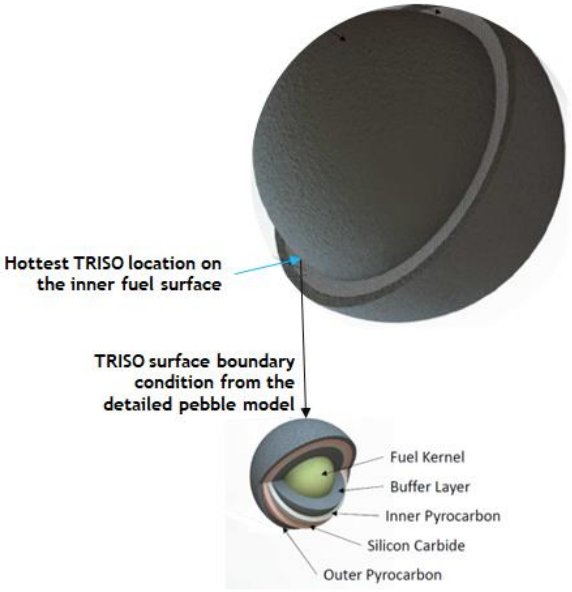
Molten Salt-Cooled Reactor



FHRs

- Pebble-bed core, fueled with TRISO-pebbles, moderated with graphite, and cooled with liquid salts.
- *Potential Future Designs – Kairos Power*

TRISO Particles



- TRISO – Tri-structural ISOtropic particle fuel; embedded in graphite pebbles or compacts
- Multi-later spherical fuel particle; consisting of kernel, buffer, pyrolytic carbon and cladding
- Kernel is the fissionable fuel; typically, UO2 or UCO

Phenomena Relevant to Safety

- Fission product migration to and attack of SiC layer
- Oxygen and carbon monoxide release from kernel
- Pressure buildup inside the iPyC due to fission products, carbon monoxide, and free oxygen
- Mechanical stress analysis of pressure / kernel swelling of structural layers
- Impact of temperature and irradiation on material properties
- Impact of manufacturing defects on fission product migration

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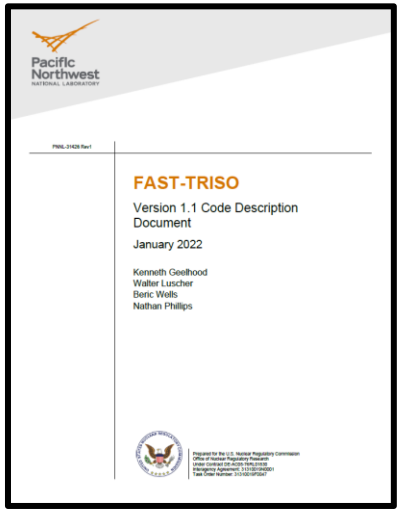
TRISO Fuel Modeling with FAST

- New Standalone 1D code for TRISO fuel performance
  - Leverages the framework of NRC's fuel performance code FAST
  - Focuses on uranium oxycarbide (UCO) kernels surrounded by buffer, inner pyrocarbon (IPyC), silicon carbide (SiC), and outer pyrocarbon (OPyC) layers

- FAST-TRISO includes the following capabilities
  - Heat transfer from the kernel to the particle surface
  - Stresses in PyC and SiC layers
  - Fission product transport from the kernel through the layers
  - Monte Carlo analysis for layer failure probabilities



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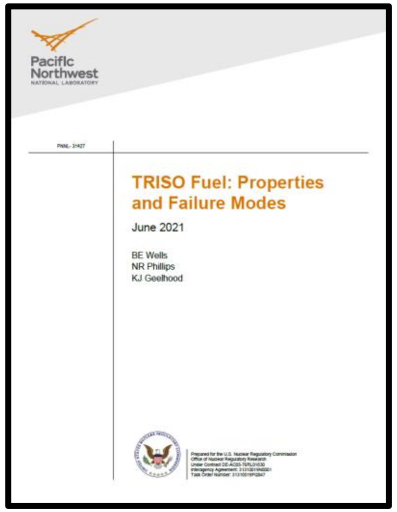

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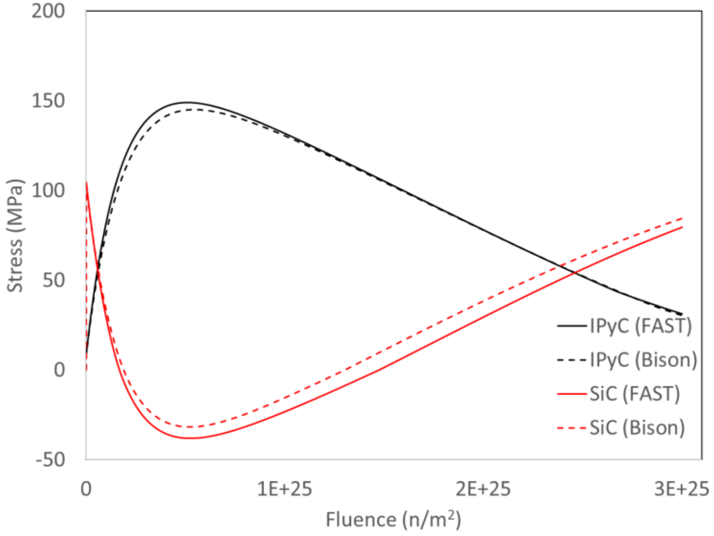
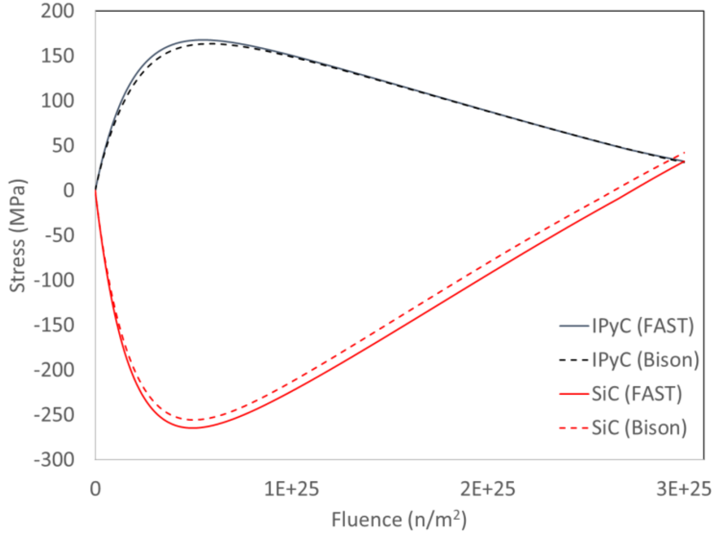
NRC's fuel performance code FAST has been extended to model the steady-state response of individual TRISO particles. FAST-TRISO models the particle's temperatures, pressures, and deformation.

Ongoing Code Development & Validation Efforts in FAST-TRISO

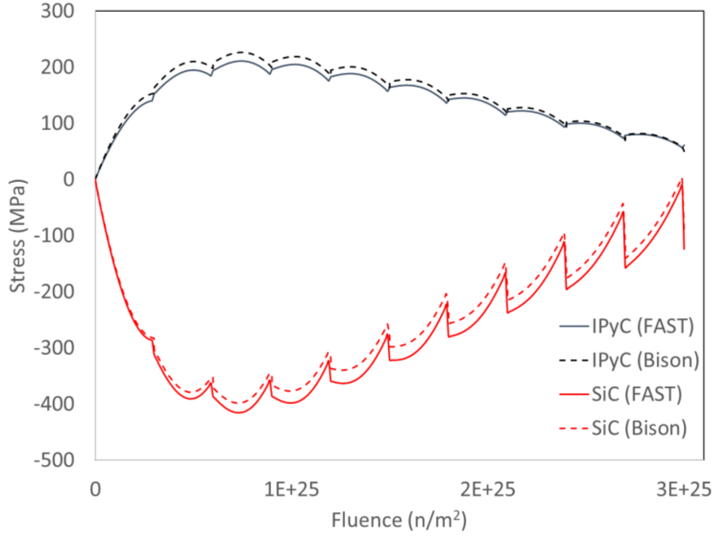
- Code development
  - Mechanical model recently extended to include PyC swelling and creep
  - Currently developing correlations for stress concentrations due to PyC cracking and debonding and aspherical particles (using Abaqus)
- Code assessment
  - Results in good agreement with CRP-6 fuel performance cases 1-8 in IAEA-TECDOC-1674
  - Work comparing to AGR fission product release and failure data ongoing



CRP-6 Case 4d



CRP-6 Case 6



CRP-6 Case 8

NRC staff is maintaining awareness as more irradiation and integral data becomes available for code development and validation purposes.

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LWR Spent Fuel Performance Analyses

- FAST has been used to support LWR spent fuel analyses, by determining
  - Initial conditions to support cask analyses (e.g., end-of-life fuel characteristics)
    - Cladding oxide thickness and hydrogen content
    - HBU mechanical properties
    - Rod internal pressure
  - Initial conditions for creep rupture analyses

- Recent updates in FAST enhanced LWR spent fuel analyses
  - Ability to change ex-reactor boundary conditions
  - New backend, ex-reactor spent fuel models
    - cladding creep models
    - helium production and release model
    - pellet swelling model
  - New ex-reactor cladding creep rupture criteria

While spent fuel, ex-reactor, modeling enhancements have focused on LWR fuels, these capabilities can be leveraged to support non-LWR spent fuel modeling.

Applying FAST to TRISO Spent Fuel Storage and Transportation

- FAST-TRISO was developed with in-reactor behavior in mind
  - However, the code addresses phenomena that are also important during storage and transportation conditions (e.g., PyC creep and SiC fracture, fission gas release and gas pressurization, fission product diffusion through particle layers)
  - The code also provides initial conditions at start of storage or transportation
- Using FAST-TRISO for storage and transportation has some challenges
  - Many models are only valid at higher temperatures (> 600 C) than what would be expected during normal storage conditions
    - Extrapolating to lower temperatures is possible, but it is hard to trust the results without some data for validation
    - Fortunately, many of the phenomena modeled by FAST-TRISO occur very slowly at low temperature (e.g., PyC creep, fission product diffusion)
  - Representative temperatures during storage or transportation conditions must be provided as input to the code
    - Can be provided from other codes that can calculate expected temperatures
    - Can also be taken from imposed limits

Effects of RE Doping and High-Energy Irradiation in Silicon Carbide for Advanced Nuclear Applications

Presenting Author :Umapathy R Ganjigatte

Indian Institute of Technology Delhi, New Delhi, India

& Inter University Accelerator Center, New Delhi, India

Sample Preparation

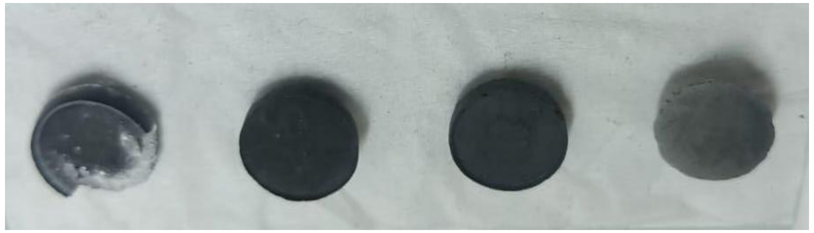
Synthesis

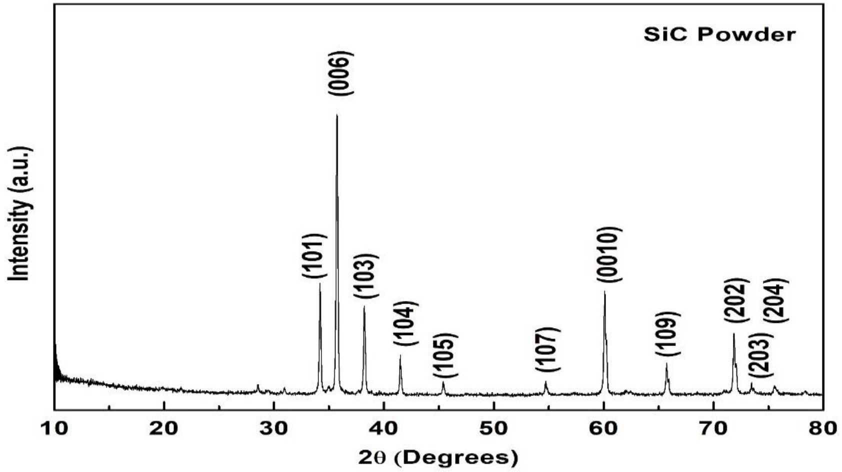
- ❑ Pellets preparation at 7 mbar
  - ❑ SiC
  - ❑ SiC + PVA
  - ❑ SiC + PVA + Stearic Acid
- ❑ Direct sintering in Argon
  - ❑ 1500º C for 3h
- ❑ Step sintering in Argon
  - ❑ 400º C for 1h
  - ❑ 1000º C for 1h
  - ❑ 1500º C for 3h

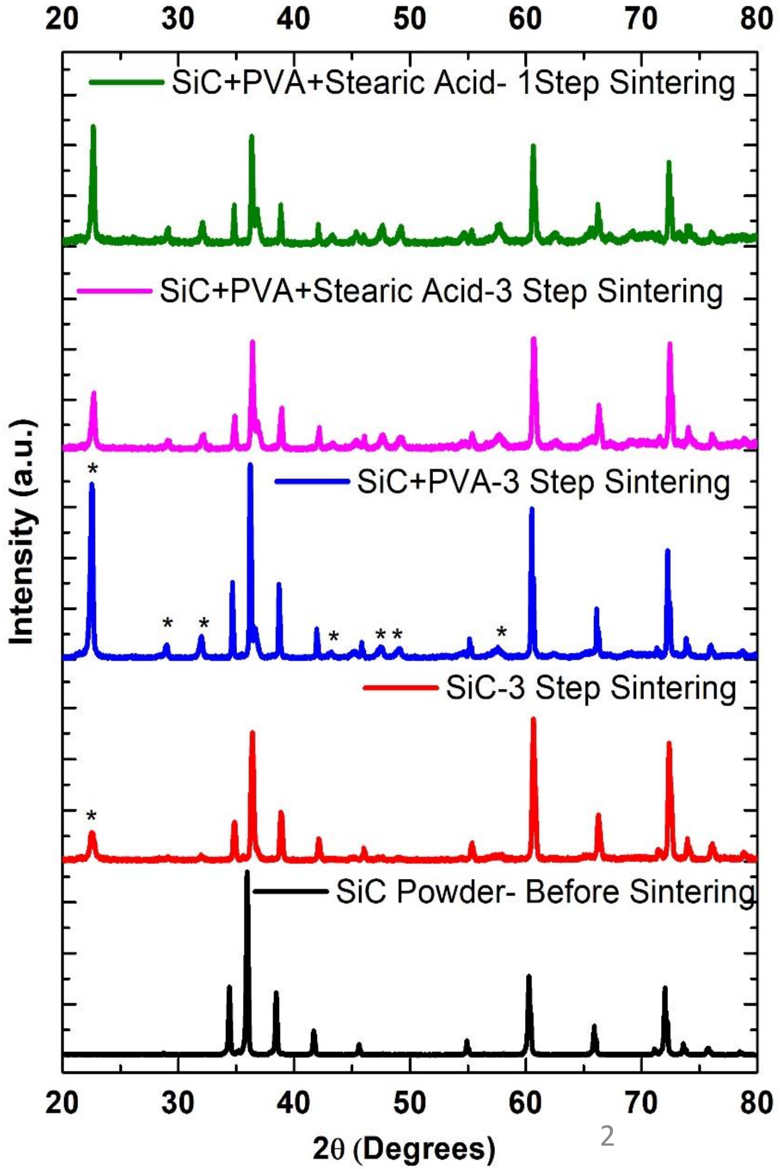
Dopant+Al2O3_SiC

- ❑ 1000º C for 1h
- ❑ 1400-1600º C for 10 h

XRD

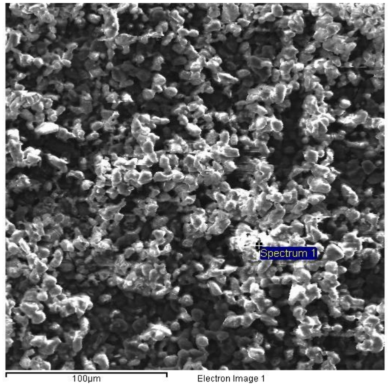




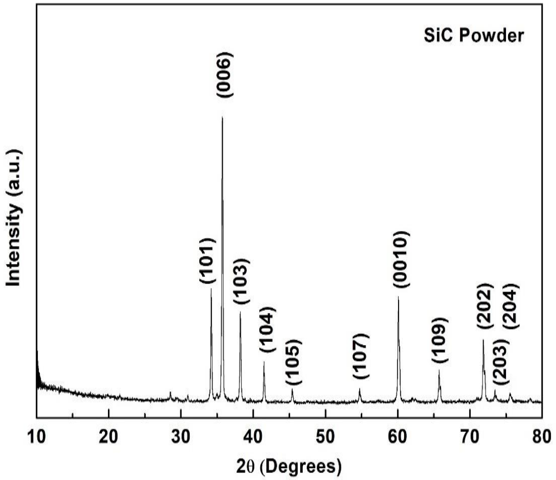


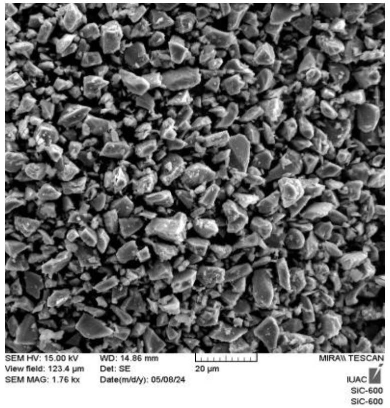
SiC Pellet Sintering

Element	Weight%	Atomic%
C K	34.15	54.81
Si K	65.85	45.19

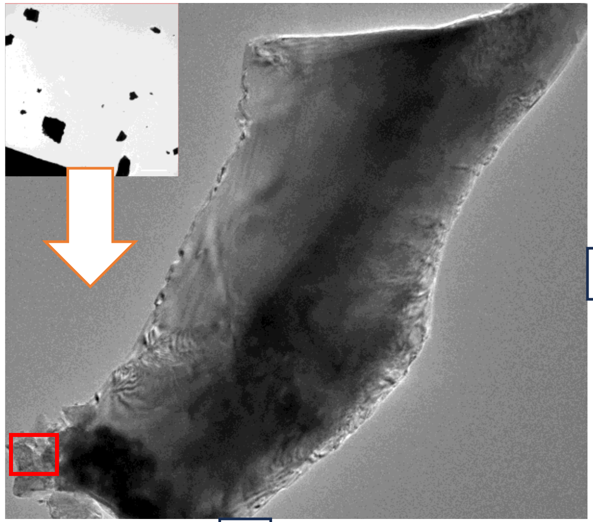


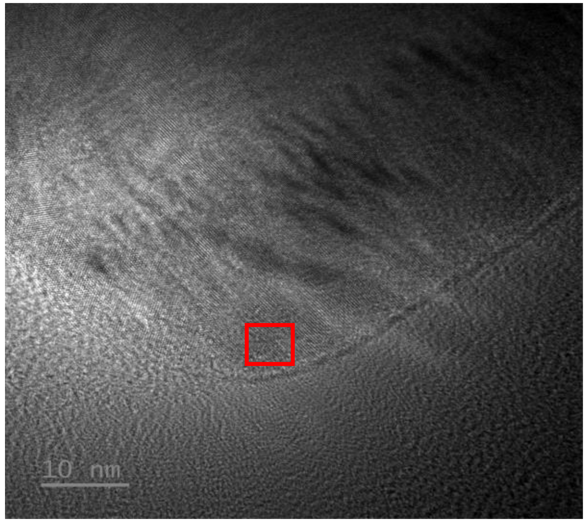
Element	Weight %	Atomic %
C K	0.24	0.40
O K	52.74	66.05
Si K	47.02	33.55

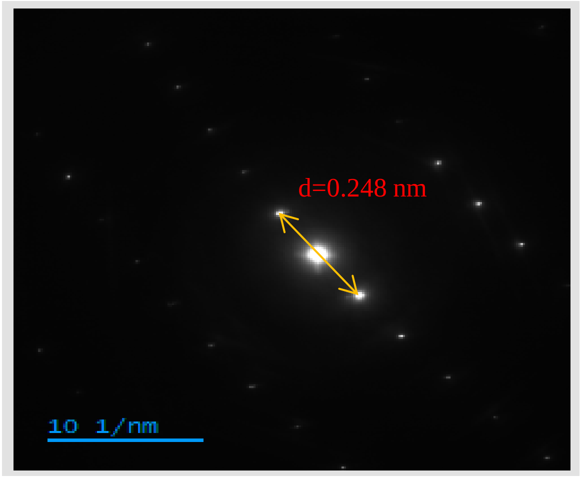


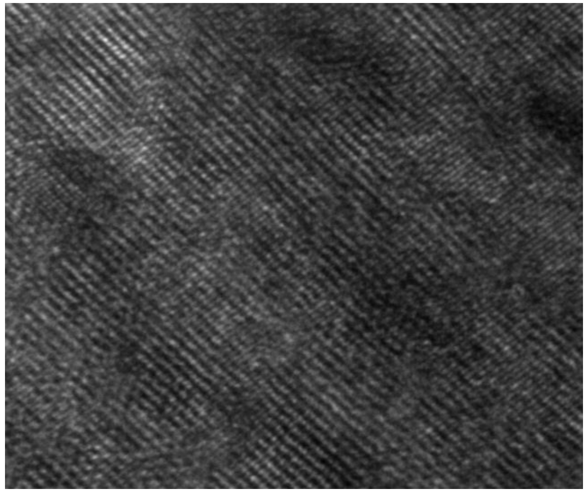


Element	Weight%	Atomic%
C K	34.47	55.16
Si K	65.53	44.84



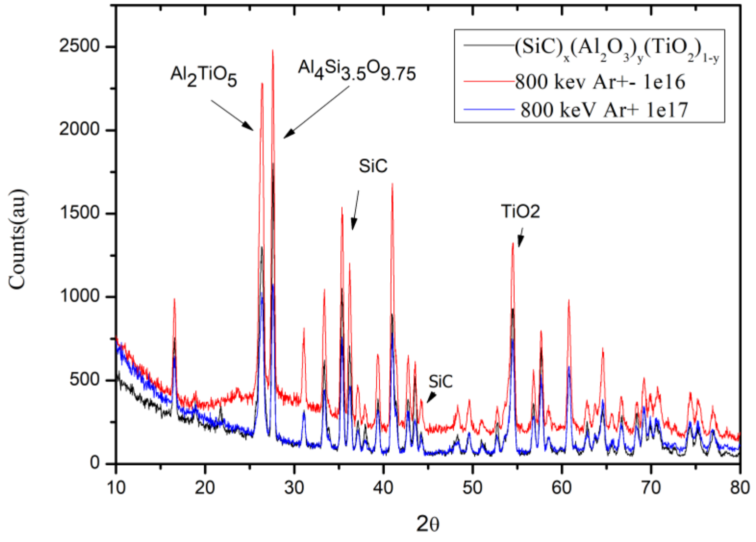


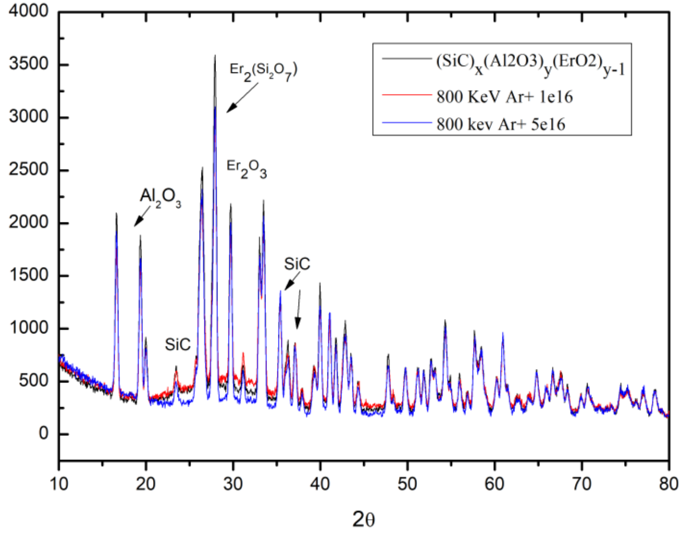


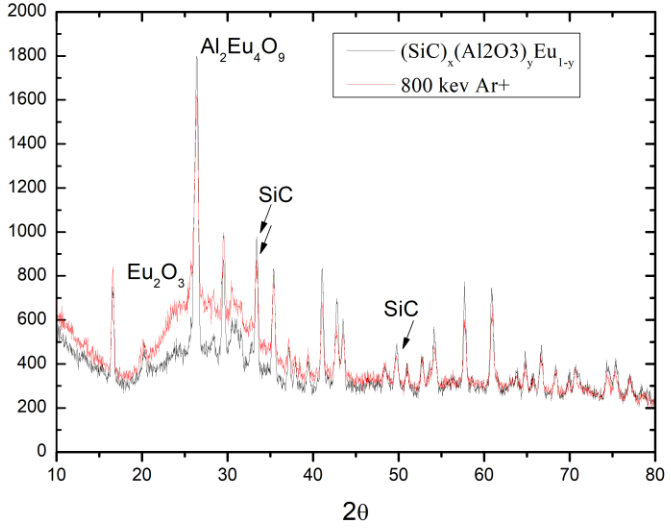


$(SiC)_x(RE/D^*)_y$ Pellet Sintering

Property	SiC	Al_2O_3	TiO_2	Er_2O_3	Nb	Eu
Melting Point (°C)	~2730	~2072	~1850	~2343	~2468	~822
Thermal Conductivity (W/m·K)	120–270	25–35	6–11	15–20	~54	~13.9
Thermal Expansion ($\times10^{-6}$ /°C)	~4	~8	~8–9	~7	~7	~35
Oxidation Resistance	Excellent	Excellent	Moderate	Excellent	Poor (coated)	Poor
High-Temp Strength	High	High	Moderate	Moderate	High	Low







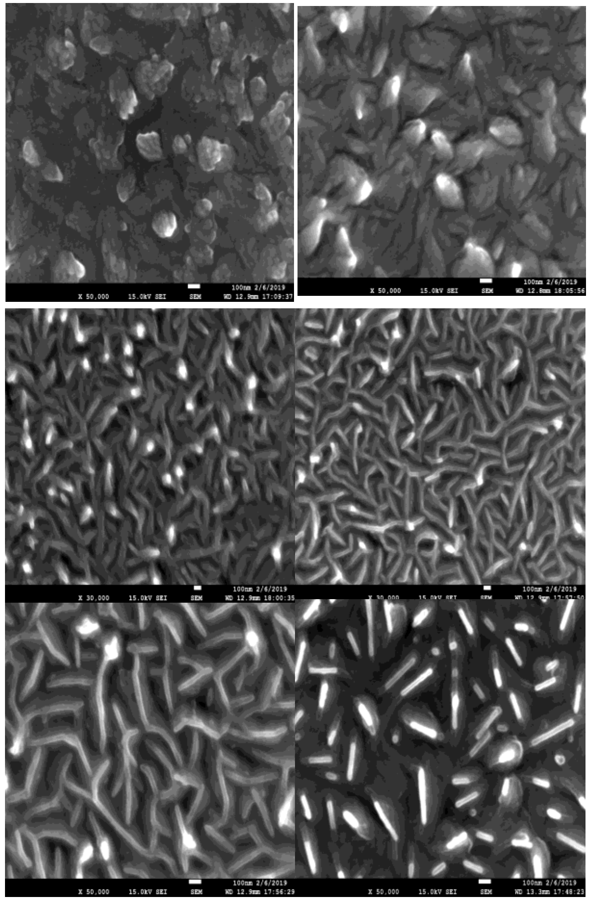


Fig 3C) The intra-granular and grain face bubbles grow as fission gas atoms and vacancies diffuse in them. The inter-granular bubbles are initially lenticular, but as they grow and coalesce, they become first elongated and then vermicular.

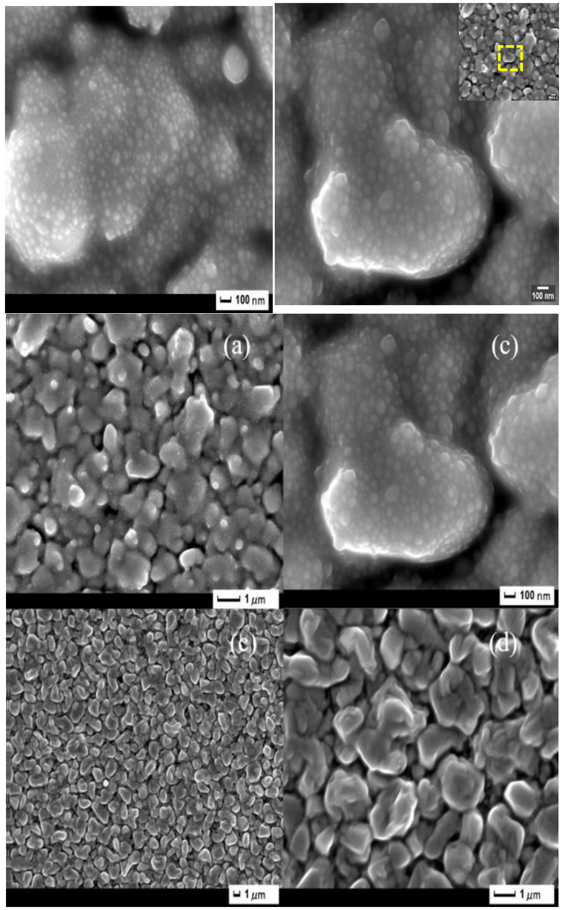


Fig 3A) circular and uniform in the range 20-50 nm in Sn target The bubbles are initially smaller in size and try to migrate to grain surface..

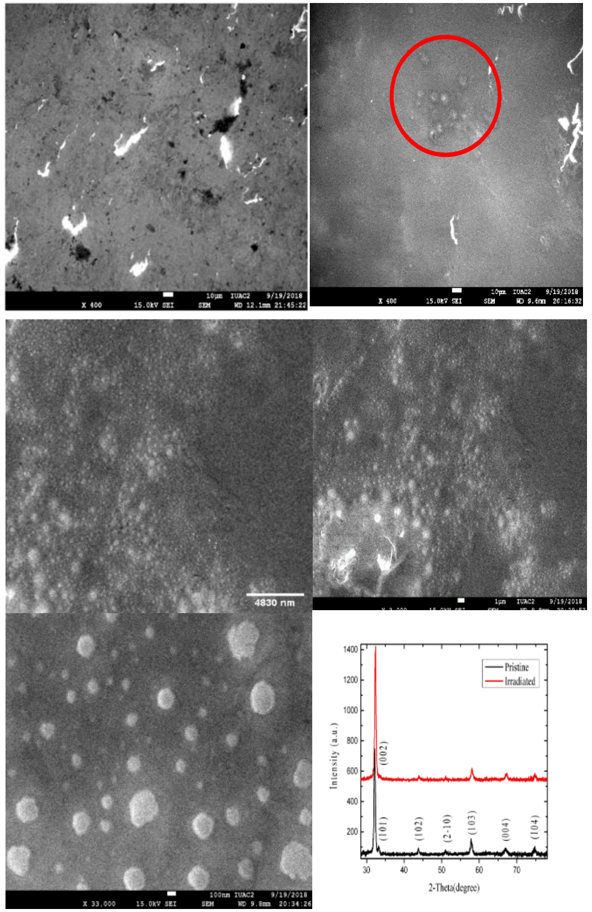


Fig 3B). inter-granular bubbles on migration to the adjacent grains. The smaller bubbles coalescence to became bigger bubbles. In these event gas traps in this give rise to the bubble burst leaving behind the crack's marks

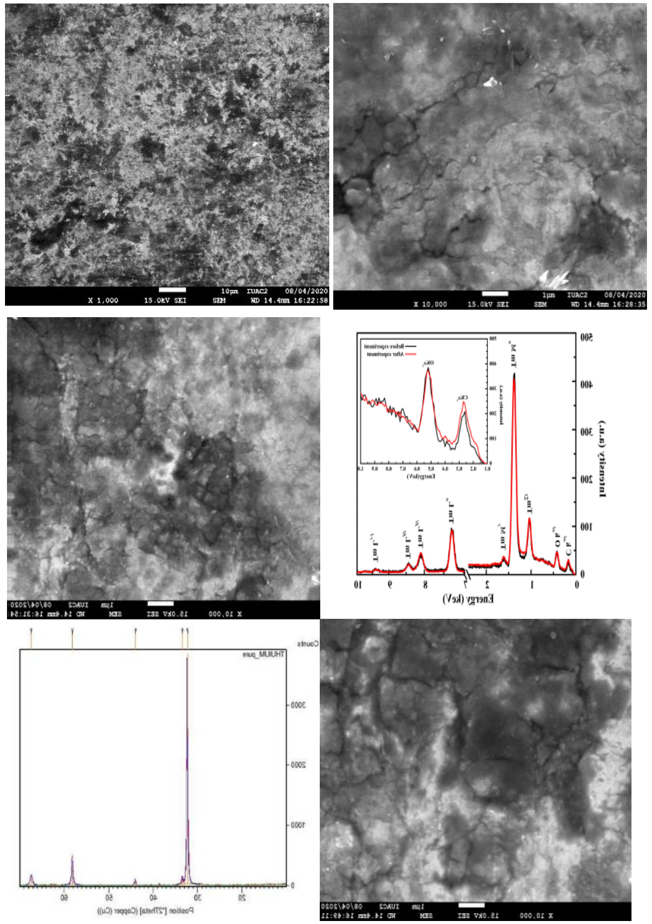


Fig3D). The black surface observed :on the beam spot is examined and this is observed to be carbon rich., The building of carbonaceous is due to locally developed heat and trapping carbon and other gases from the surrounding

TABLE 1. The target, preparation, type, thickness, beam, energy, and other details

	For **Sn** target	For **Er** target	For **Te** Target	For **Tm** target
Beam	^{7}Li	^{28}Si	^{35}Cl	^{18}O
Energy(MeV)	30	180	121-155	94
DC/Pulsed	DC	Pulsed	DC	DC
Enrichment (isotope)	^{118}Sn	^{166}Er	^{130}Te	^{169}Tm
Coulomb barrier energy (MeV)in CMF	19.55	111.85	105.16	65.91
Stable evaporation (%)	$^{122}I(84)$	$^{187}Tl(15),^{188}Tl(8.7)$	$^{161}Tm(46),\ ^{160}Tm(32)$	$^{182}Ir(65),^{183}Ir(11)$
Thickness	50-100 nm	1000 nm	40-50 nm	1-6 um
Preparation method	Thermal evaporation	Cold Rolling	Thermal evaporation	Cold Rolling
Target type	Freestanding	Free standing	C backed	Free Standing
Reference	[4]	[2]	[1]	[3]

1. A.Banerjee, et. al. Nucl. Instrum. Methods Phys. Res. A 887(2018) 34-39.
2. Rudra N.Sahoo, et al. ,Nucl. Instrum. Methods Phys. Res. A 935(2019) 103-109.
3. A Sharam et. al., Nucl. Instrum. Methods Phys. Res. B 511, (2022),1-5.
4. Arshiya Sood, et. al.,172 Vacuum (2020), 109107.
5. Giyn Rossiter and Mike Mignanelli, NNL(10) 10930, Issue 2, online pfd source, (2011)

TABLE 2. The Simulation details PACE4 and SRIM /TRIM* code

Simulation	For Sn target	For Er target	For Te Target	For Tm target
Projectile(10k)	^{7}Li	^{28}Si	^{35}Cl	^{169}Tm
H	106	5295	1279	821
He	77	3585	740	769
Neutrons	28295	44898	41548	47611
Vacancies /ion*	0.1	1.8	5.6	1

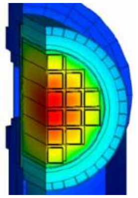
Thank you

10 CFR PART 71 - CERTIFICATION OF TRANSPORTATION PACKAGES FOR METAL FUEL

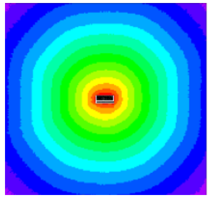
WORKSHOP ON STORAGE AND TRANSPORTATION OF TRISO AND METAL SPENT NUCLEAR FUELS DECEMBER 3-5, 2024

Drew Barto
U.S. NRC
NMSS Division of Fuel Management

Nuclear Physics Considerations for TRISO/SFR Spent Fuel Safety



Decay Heat



Shielding and Radiation Protection

Neutron Multiplication and Criticality

NRC Regulations limit radiation dose under all phases of the fuel cycle:

- Direct radiation dose
- Radioactive material releases
- Inadvertent criticality

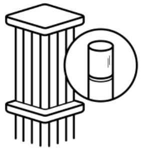
Computer codes used to determine:

- Irradiated fuel composition for nuclides that contribute to:
  - Direct radiation dose and dose from radioactive material releases
  - Decay heat
  - Determination of criticality safety (k_{eff})
- Radiation dose and k_{eff}

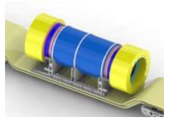
Codes must be validated against measured irradiated fuel data

10 CFR 20 – Radiation Protection

10 CFR 50/52 – Power Plants



10 CFR 70 – Fuel Cycle Facilities

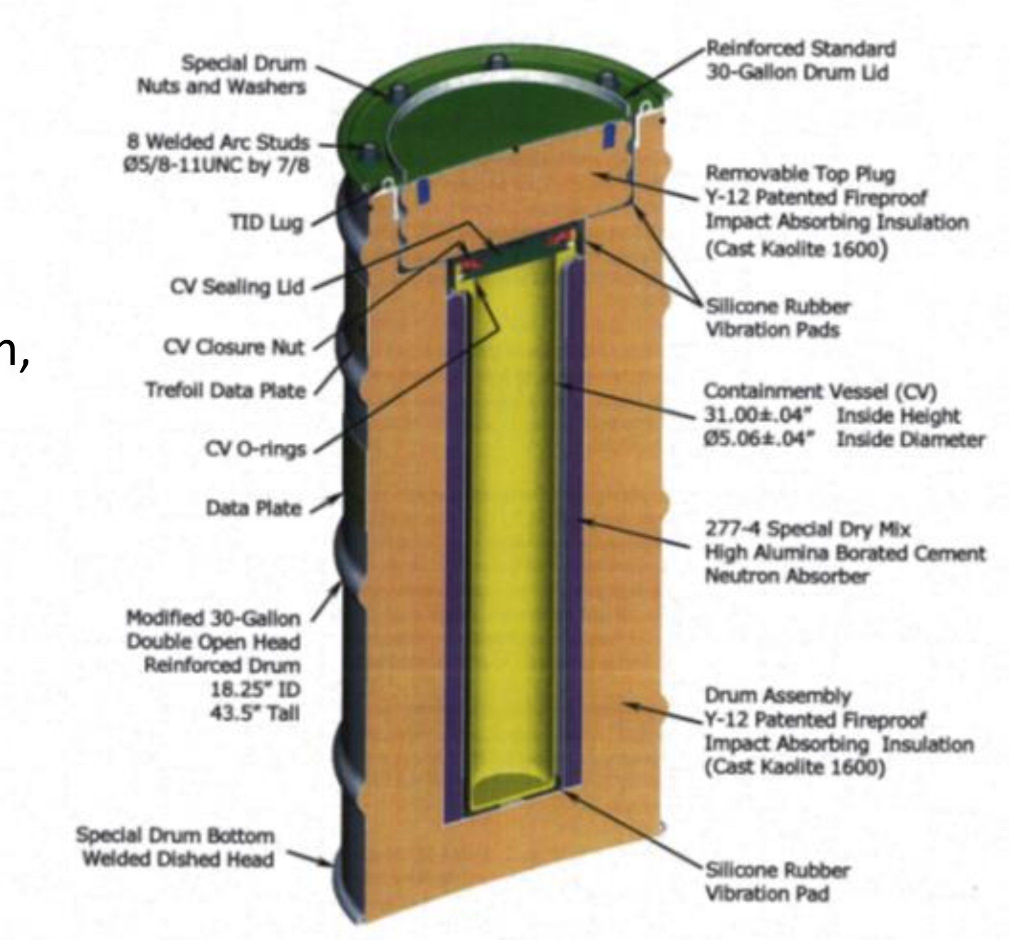


10 CFR 71 – Transportation

10 CFR 72 – SNF Storage

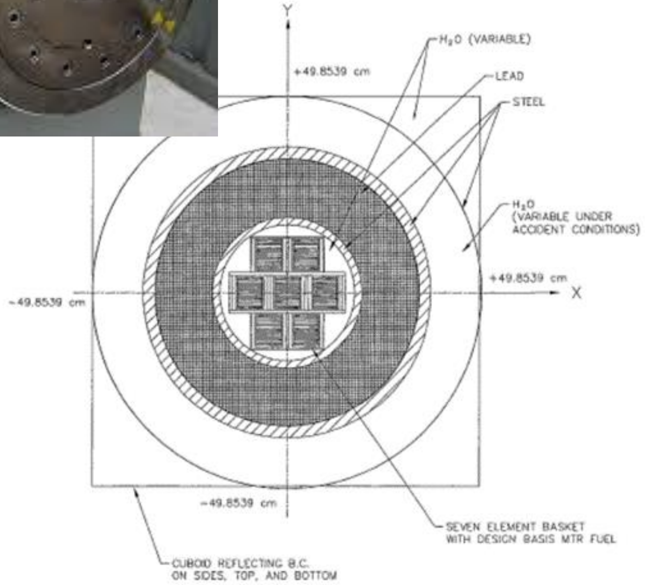
Metal Fuel Transportation Package Designs

- ES-3100
  - USA/9315/B(U)F-96: "Uranium as solid metal or alloy, packaged in stainless-steel or tin-plated carbon steel convenience cans. Alloys of uranium include uranium-aluminum, uranium-molybdenum, and uranium-zirconium.
  - Up to 100% enriched
  - Up to 35 kg 235U per package, depending on geometry and use of neutron absorbing spacers
  - CSI as low as 0.0 (no limit on number of packages per conveyance)
  - Unirradiated metal



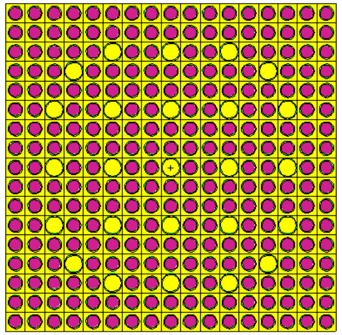
Metal Fuel Transportation Package Designs

- NAC-LWT
  - Truck cask for spent commercial/research reactor fuel
  - USA/9225/B(U)F-96: “Metallic fuel rods containing natural enrichment uranium pellets with aluminum cladding 0.080-inches thick.”
  - Low enrichment (natural) and low burnup (1,600 MWd/MTU)
  - 15 metallic fuel rods
  - CSI of 0.0; package is 52,000 lbs (one per truck)



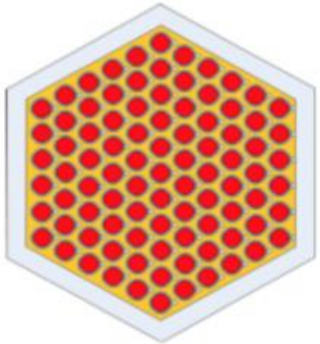
LWR vs. Metal

LWR



- UO_2 fuel pellets
- Zirconium alloy cladding
- Helium gap
- Up to 5% enrichment (8% LEU+)
- Water moderated/cooled
- Burnup up to 60 GWd/MTU (higher for LEU+)
- Static during irradiation

SFR



- Uranium metal alloy slugs or rods
- Stainless steel cladding
- Sodium bonded gap
- Up to 20% enrichment; may include TRU
- Sodium moderated/cooled
- Burnup up to 150 GWd/MTU
- Static during irradiation

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Spent SFR Fuel Storage and Transportation

Key Messages

- Although SFRs are not new, NRC does not have much experience certifying storage and transportation systems for spent SFR fuel
- Some similarities between LWR fuel storage and transportation systems and those likely to be used for SFR fuel
- Current neutronics codes well capable of evaluating SFR fuel to estimate decay heat, radiation dose, and criticality safety
- May need some additional margins due to lack of code validation data for isotopic depletion and criticality codes
- Margins can be reduced when code validation data becomes more available

Storage and Transportation of TRISO and Metal Spent Nuclear Fuels

NRC's simulation capabilities supporting criticality, reactor physics, decay heat, and shielding for metallic fueled non-LWRs

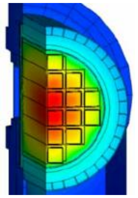
Andy Bielen, Ph.D.

Office of Nuclear Regulatory Research
Division of Systems Analysis
Fuel & Source Term Code Development Branch

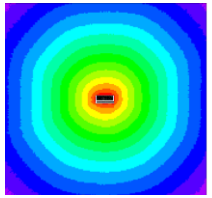
Objectives

- NRC’s simulation capabilities supporting nuclear fuel safety for metallic fuel designs
  - Decay Heat
  - Neutron Multiplication & Criticality
  - Shielding and Radiation Protection
- Overview of data availability, gaps, and where additional data would be beneficial

Nuclear Physics Considerations for TRISO/SFR Spent Fuel Safety



Decay Heat



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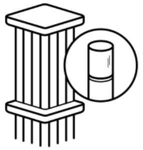
  - Direct radiation dose and dose from radioactive material releases
  - Decay heat
  - Determination of criticality safety (k_{eff})
- Radiation dose and k_{eff}

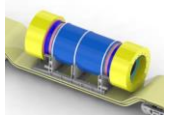
Codes must be validated against measured irradiated fuel data

10 CFR 20 – Radiation Protection

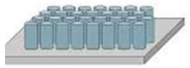
10 CFR 50/52 – Power Plants



10 CFR 70 – Fuel Cycle Facilities

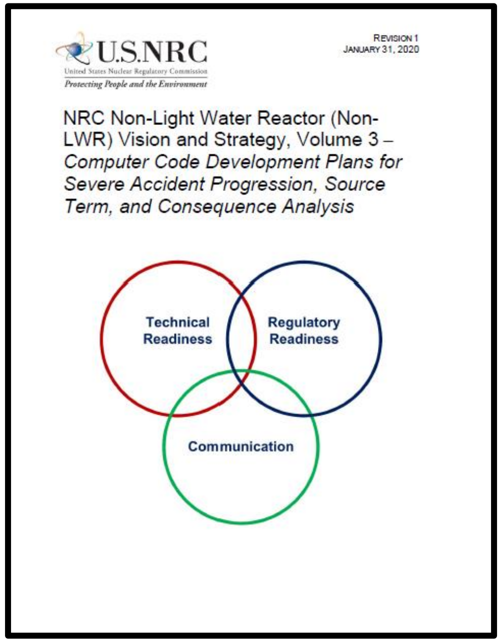


10 CFR 71 – Transportation



10 CFR 72 – SNF Storage

Non-LWR Source Term & Fuel Cycle Demonstration Projects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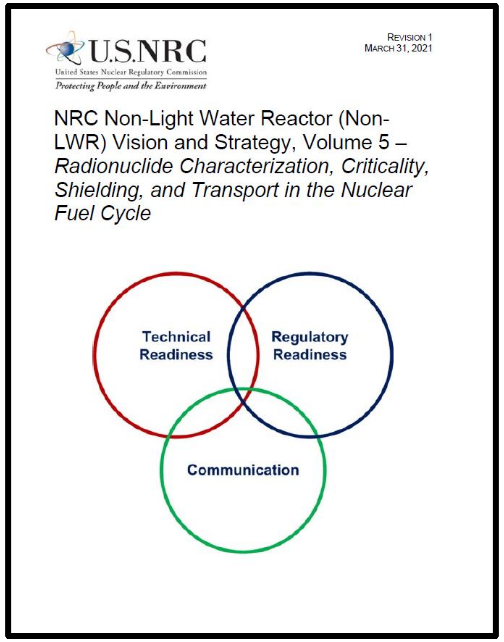
REVISION 1
JANUARY 31, 2020

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 3 – *Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis*

[Source Term]

NRC's comprehensive neutronics package
- Cross-section processing
- Decay heat analyses
- Criticality safety
- Radiation shielding
- Radionuclide inventory & depletion generation
- Reactor core physics

NRC's comprehensive severe accident progression and source term code
- Accident progression
- Thermal-hydraulic response
- Core heat-up, degradation, and relocation
- Fission product release and transport behavior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REVISION 1
MARCH 31, 2021

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 5 – *Radionuclide Characterization, Criticality, Shielding, and Transport in the Nuclear Fuel Cycle*

[Fuel Cycle]

Non-LWR demonstration projects improve and validate SCALE & MELCOR for simulating non-LWRs for severe accident progression and fuel cycle analyses.

Using SCALE to Calculate Non-LWR Neutronics Quantities of Interest

Decay Heat

- SCALE/TRITON is used to generate specific ORIGEN reactor libraries; functionally bounds fuel enrichment and burnup.
- SCALE/ORIGAMI is used to obtain the spent fuel inventories; uses ORIGEN to compute detailed irradiated and decayed isotopic compositions.

Criticality Safety

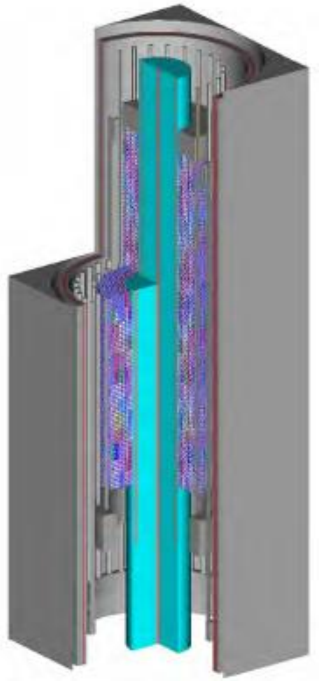
- SCALE/CSAS is used to perform criticality safety analyses. CSAS is a sequence that uses Monte Carlo transport codes KENO or Shift.
- Used to determine the multiplication factor of any system.

Shielding & Dose

- SCALE/MAVRIC is used to perform the shielding and dose analyses.
- Uses the radiation source term & radionuclide inventories generated from SCALE/TRITON or SCALE/ORIGAMI.

Non-LWR Reference Models

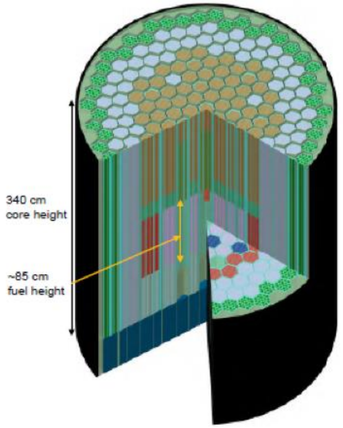
High-Temp. Gas Cooled Reactor



PBMR-400

- 400 MWth reactor, graphite moderated
- Helium-cooled & TRISO-particle pebble-fueled at 10 wt.% U-235
- Fuel discharged at high burnup (90 GWd/MTIHM)

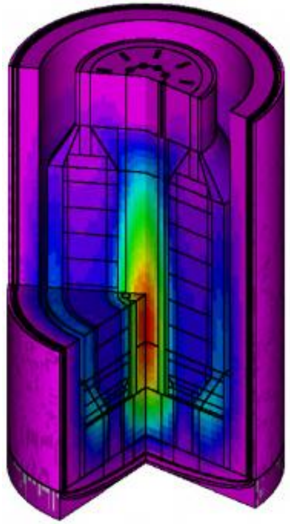
Sodium-Cooled Fast Reactor



ABTR

- 250 MWth pool-type reactor, utilizing metallic U / HT-9 fuel rods
- Reactor fueled with U-Pu-Zr fuel slugs
- Liquid sodium coolant

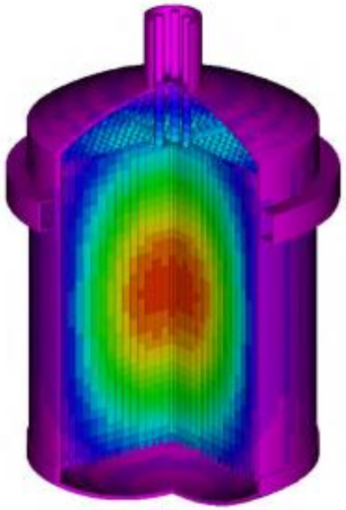
Molten Salt-Cooled Reactor



UCB Mk1 PB-FHR

- 236 MWth reactor at atmospheric pressures
- Flibe cooled & Pebble fueled (TRISO) at 19.9 wt.% U-235
- Online refueling

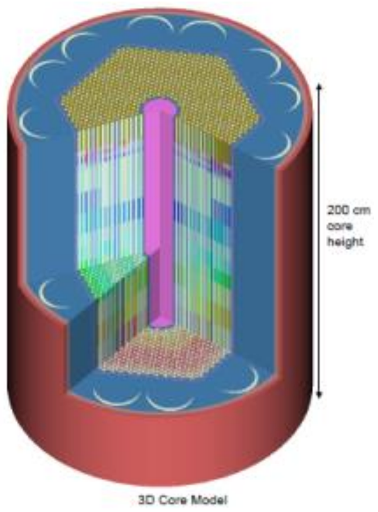
Molten Salt-Fueled Reactor



MSRE

- 10 MWth reactor, graphite moderated at near atmospheric pressures
- Reactor fueled with liquid dissolved fuel in molten salt (34.5 wt. % U-235)

Heat Pipe Reactor

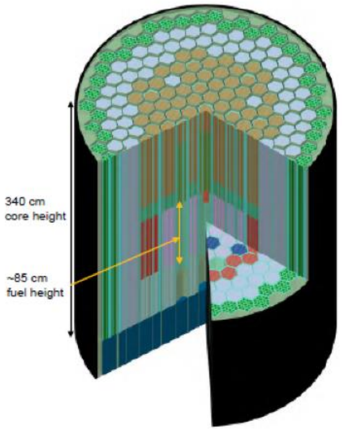


INL Design A

- 5 MWth with a 5-year operating lifetime
- 1,134 heat pipes fueled with UO_2 fuel (19.75 wt.% U-235)
- Reactivity controlled via control drums

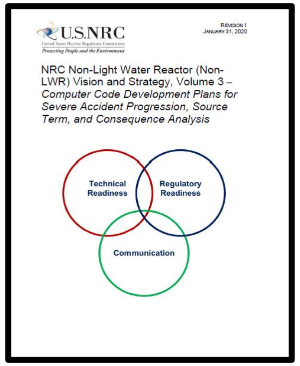
Sodium Fast Reactor Workflows

Sodium-Cooled Fast Reactor



ABTR

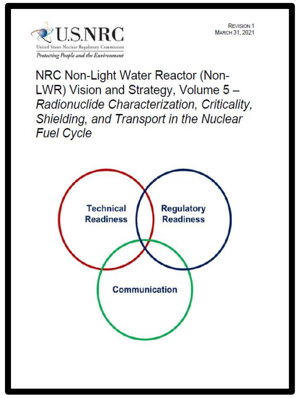
- 250 MWth pool-type reactor, utilizing metallic – U-fueled / HT-9 clad fuel rods
- Reactor fueled with U-Pu-Zr fuel slugs
- Liquid sodium coolant



U.S.NRC
Revision 1
January 31, 2020

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 3 – *Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis*

[Source Term]



U.S.NRC
Revision 1
March 31, 2021

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 5 – *Radionuclide Characterization, Criticality, Shielding, and Transport in the Nuclear Fuel Cycle*

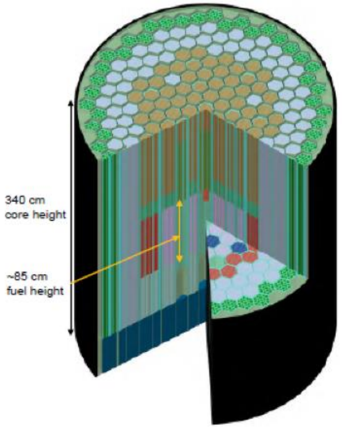
[Fuel Cycle]

Fuel Depletion, Decay Heat & Radionuclide Inventory Generation

- Decay Heat
- Criticality Safety
- Shielding & Dose

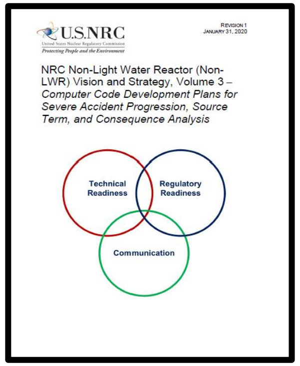
Fuel Depletion, Decay Heat, and Nuclide Inventory Generation

Sodium-Cooled Fast Reactor



ABTR

- 250 MWth pool-type reactor, utilizing metallic U / HT-9 fuel rods
- Reactor fueled with U-Pu-Zr fuel slugs
- Liquid sodium coolant

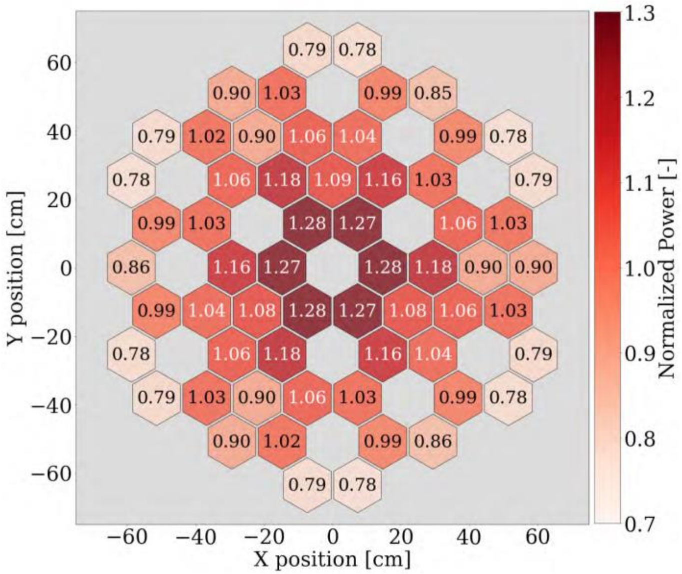

U.S.NRC
Revision 1
January 31, 2020

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 3 – *Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis*

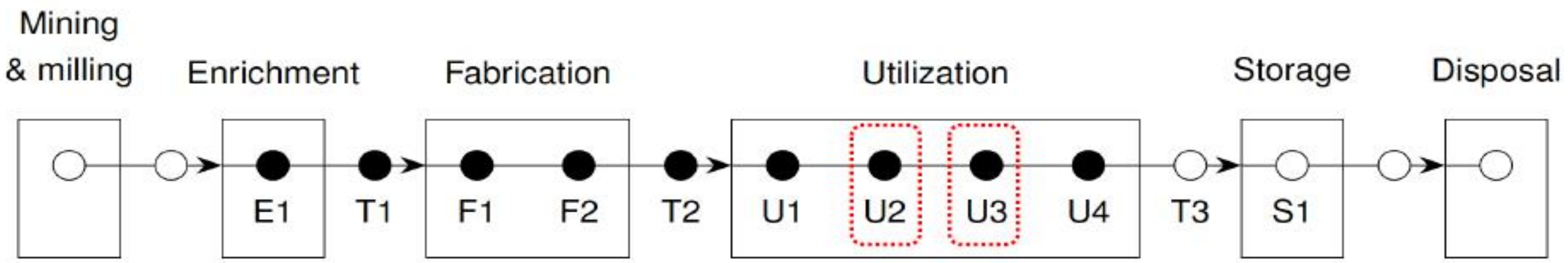
Fuel Depletion, Decay Heat & Radionuclide Inventory Generation

Source Term

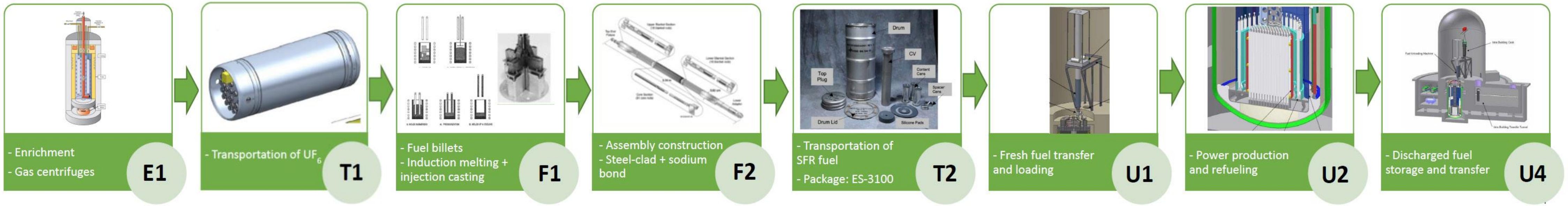
- SCALE/TRITON used for fuel depletion
  - Full core 3D continuous energy Monte Carlo physics
  - All fuel assemblies in the core depleted (Total of 60)
  - ORIGEN used to track >2,000 nuclides
- Radionuclide inventories used to support downstream analyses.
  - MELCOR for severe accident progression & radionuclide transport
  - MAVRIC for shielding & dose analyses



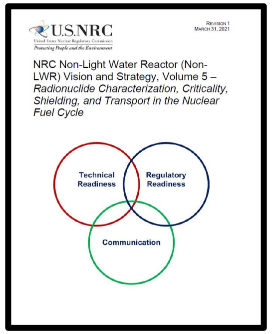
Sodium Fast Reactor Fuel Cycle



E1 – UF_6 enrichment
T1 – Transportation of UF_6 to fabrication facility
F1 – Fuel fabrication
F2 – Fuel assembly/pebble fabrication
T2 – Transportation of assemblies/pebbles/salt to plant
U1 – Fresh fuel staging/preparation/loading
U2 – Power production
U3 – Spent fuel pool/shuffle operations
U4 – On-site dry cask storage
T3 – Transportation of spent fuel to off-site storage
S1 – Off-site storage

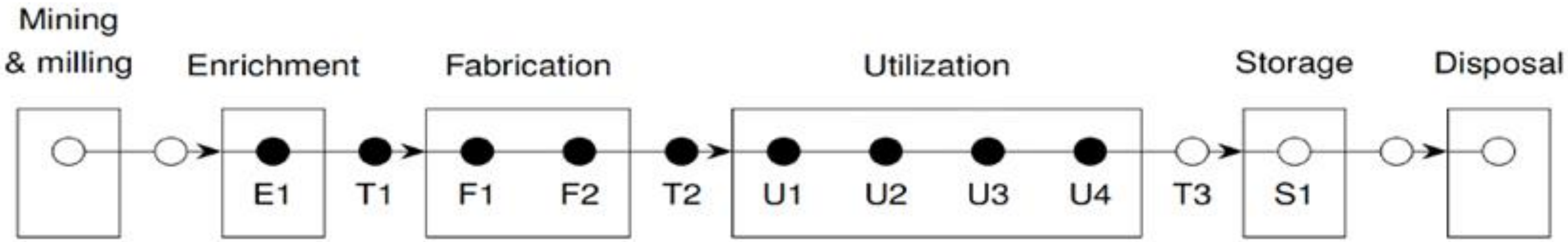


Decay Heat, Criticality Safety, and Radiation Shielding / Dose



Fuel Cycle

Decay Heat | Criticality Safety | Shielding & Dose



E1 – UF_6 enrichment
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U1 – Fresh fuel staging/preparation/loading
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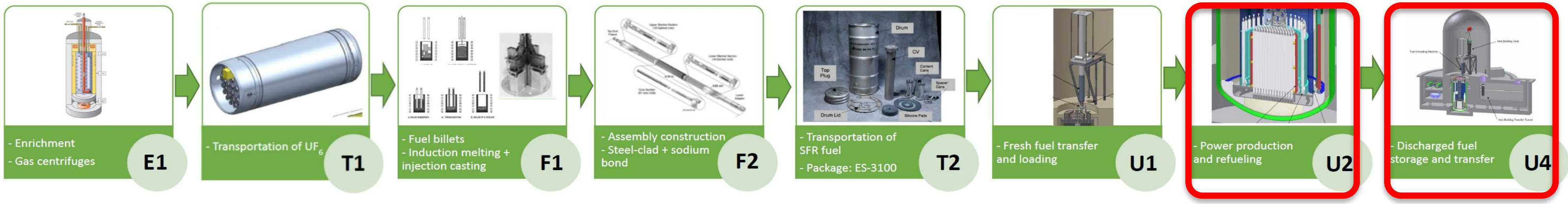
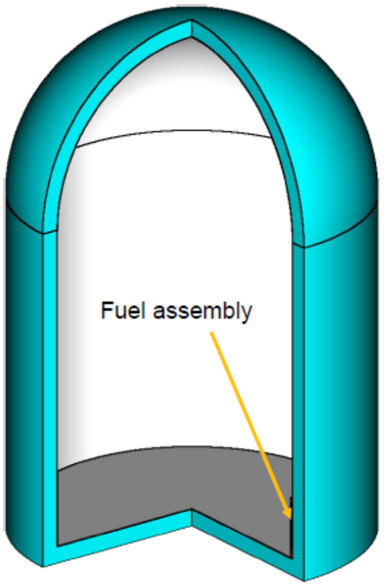


Shielding & Dose Analyses for Metallic Fuels / SFRs

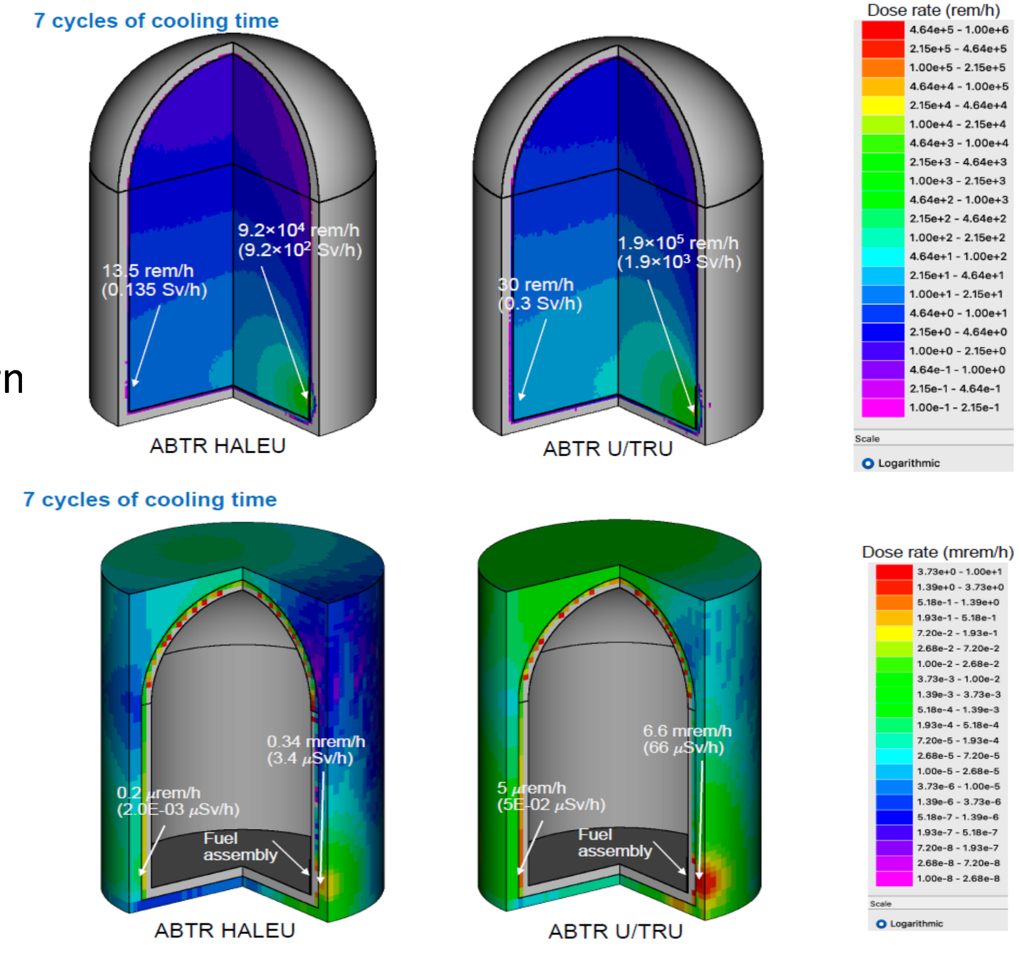
Scenario 1: Release of fission products during operation / refueling (U3)

- Accident: Seismic event causing the refueling machine to fall and release the fuel assembly.
- Analysis: Determine fuel inventory and perform SCALE radiation dose calculations.



3D view of the CB with front quarter segment removed

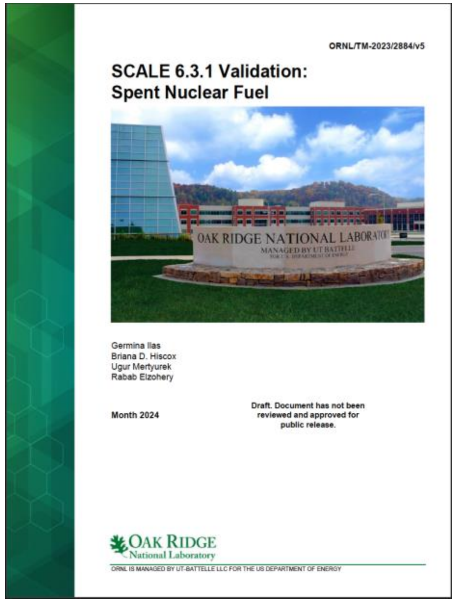
- Generated inventories used for radiative source tern
  - Leveraged from the non-LWR demonstration source term work
  - TRITON & ORIGAMI for inventories
  - MAVRIC for shielding & dose
- Radionuclide inventories used to support downstream analyses.
  - MELCOR for severe accident progression & radionuclide transport
  - MAVRIC for shielding & dose analyses

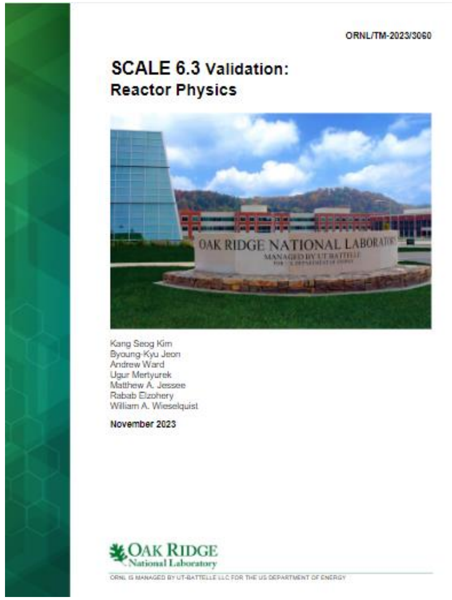


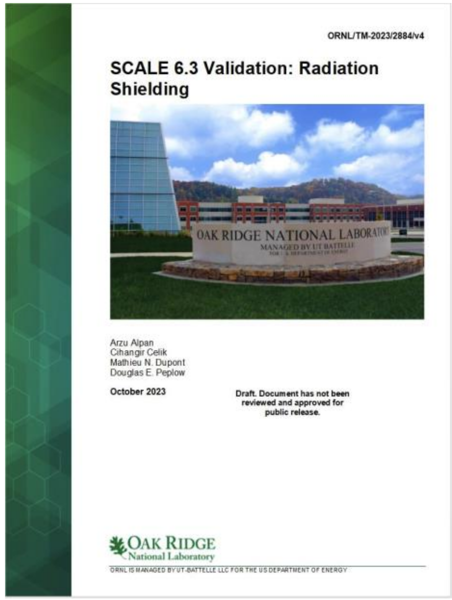
3D total dose rate maps of the containment building, generated from SCALE MAVRIC

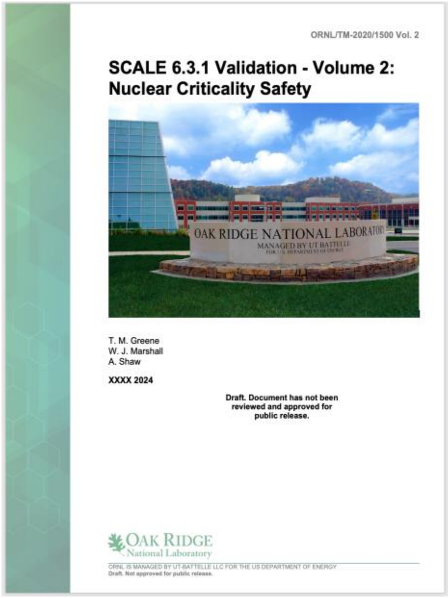
NRC’s Computer Codes and Validation

SCALE Validation in Four Major Areas (Criticality Safety, Radiation Shielding, Reactor Physics, and Spent Fuel Inventory)









SFRs

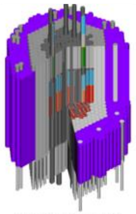


Figure 5.7. EBR-II SCALE model [62].

Table 5.4. Eigenvalue results for the high-fidelity EBR-II benchmark.

	k_{eff}	σ	Δk_{eff}^{a} (pcm)
Benchmark value [7]	1.00927	± 0.00618	reference
SCALE 6.3.1/KENO-VI CE ENDF/B-VII.1	1.00722	± 0.00010	-205 (± 618)
SCALE 6.3.1/KENO-VI CE ENDF/B-VIII.0	1.00691	± 0.00013	-236 (± 618)

[a] Calculated as 10^5($k\text{-}eff_{calculated} - k\text{-}eff_{benchmark}$).

HTGRs

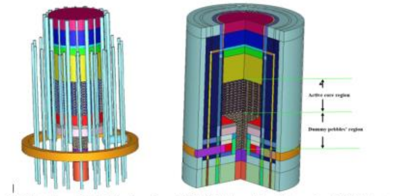


Figure 5.1. Illustration of HTR-10 benchmark model details.¶ (Channels in reflector regions [left], full reactor model [right]; images not to scale) [35].

Table 5.1. Eigenvalue results for high-fidelity HTR-10 benchmark.

	k_{eff}	σ	Δk_{eff}^{a} (pcm)
Benchmark value [6]	1.00000	0.00370	reference
SCALE/KENO-VI CE ENDF/B-VII.1	1.00303 ± 0.00041	0.99661 ± 0.00031	303 ± 370
SCALE/KENO-VI CE ENDF/B-VIII.0	1.00604 ± 0.00027	0.99919 ± 0.00026	604 ± 370
SCALE/KENO-VI 252-group ENDF/B-VII.1	1.00265 ± 0.00031	0.99595 ± 0.00025	265 ± 370
SCALE/KENO-VI 252-group ENDF/B-VIII.0	1.00376 ± 0.00027	0.99746 ± 0.00025	376 ± 370

[a] Calculated as 10^5($k\text{-}eff_{calculated} - k\text{-}eff_{benchmark}$).

MSRs

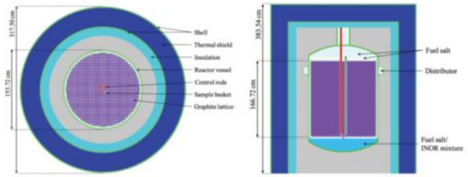


Figure 5.5. Cross-sectional illustrations of MSRE benchmark models [60].

Table 5.3. Eigenvalue results for the high-fidelity MSRE benchmark.

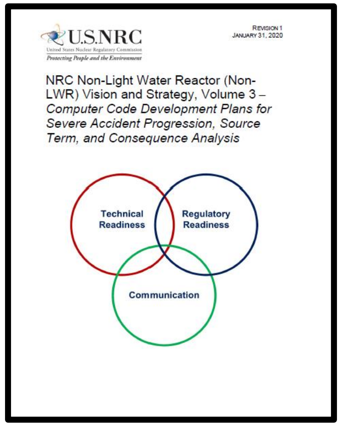
	k_{eff}	σ	Δk_{eff}^{a} (pcm)
Benchmark value	0.99978	± 0.00420	reference
SCALE 6.3.1/Shift CE ENDF/B-VII.1	1.019016	± 0.00010	1924 (± 420)
SCALE 6.3.1/Shift CE ENDF/B-VIII.0	1.021833	± 0.00010	2205 (± 420)

[a] Calculated as 10^5($k\text{-}eff_{calculated} - k\text{-}eff_{benchmark}$).

SCALE has been heavily validated for standard fuel designs in LWRs. SCALE 6.3 validation efforts are underway to validate SCALE for several advanced non-LWR systems.

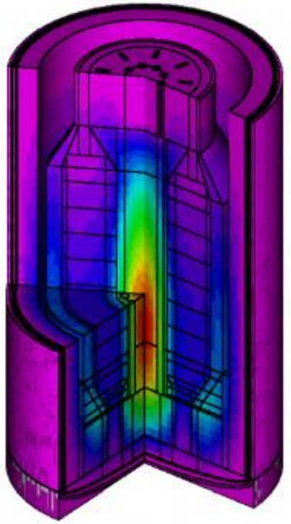
U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Applications of non-LWR Demonstration Project - Kairos Hermes Construction Permit Application Support



Source Term

Molten Salt-Cooled Reactor



UCB Mk1 PB-FHR

- 236 MWth reactor at atmospheric pressures
- Flibe cooled & Pebble fueled (TRISO) at 19.9 wt.% U-235
- Online refueling

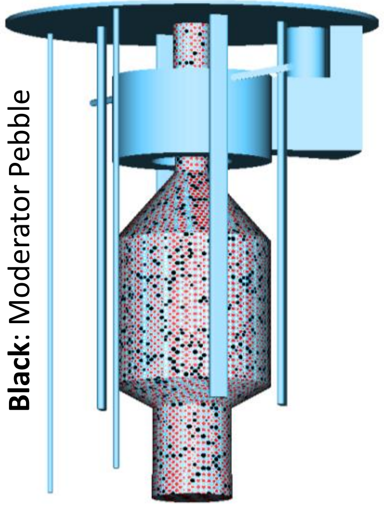
- *Generated a library of well-tested & demonstrated non-LWR reference plant models in SCALE & MELCOR.*
- *Models can be heavily leveraged to support licensing reviews.*

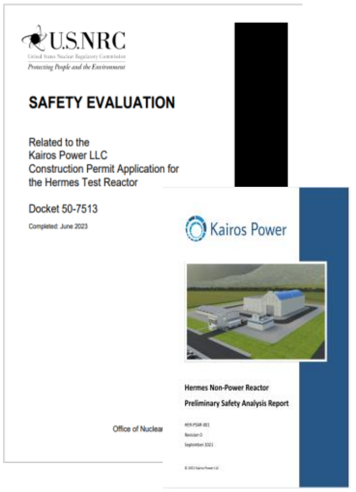
- *Leveraged the FHR model to supportrt the licensing review of Hermes I*
  - *Similarities between the UCB Mk1 & Hermes I noted*
  - *Leveraged existing models & insights from non-LWR demonstration project*
- *SCALE and MELCOR used for analyzing various scenarios (e.g., loss of forced circulation, accidental control rod withdrawal)*

Kairos Hermes I

- 35 MWth reactor at atmospheric pressures
- Flibe cooled & Pebble fueled (TRISO) at 19.75 wt.% U-235
- Online refueling

Blue: FLiBe
Red: Fuel Pebble
Black: Moderator Pebble





Non-LWR demonstration project was instrumental in an effective and efficient review of a first of a kind non-LWR.

For More Information

SCALE/MELCOR non-LWR source term demonstration project	

• Heat-pipe reactor workshop
  ◦ Slides
  ◦ Video Recording EXIT
• SCALE report
• MELCOR report
	June 29, 2021

• High-temperature gas-cooled reactor workshop
  ◦ Slides
  ◦ Video Recording EXIT
• SCALE report
• MELCOR report
	July 20, 2021

• Fluoride-salt-cooled high-temperature reactor workshop
  ◦ Slides
  ◦ Video Recording EXIT
• SCALE report
• MELCOR report
	September 14, 2021

• Molten-salt-fueled reactor workshop
  ◦ Slides
  ◦ Video Recording EXIT
• SCALE report
• MELCOR report
	September 13, 2022

• Sodium-cooled fast reactor workshop
  ◦ Slides
  ◦ Video Recording EXIT
• SCALE report
• MELCOR report
	September 20, 2022

SCALE/MELCOR non-LWR fuel cycle demonstration project	

• High-temperature gas-cooled reactor fuel cycle workshop
  ◦ Slides
  ◦ Video Recording EXIT
  ◦ SCALE Report
  ◦ MELCOR Report
	February 28, 2023

• Sodium-cooled fast reactor fuel cycle workshop
  ◦ Slides
  ◦ Video Recording
  ◦ SCALE Report
  ◦ MELCOR Report
	September 20, 2023

• Molten salt reactor fuel cycle workshop
  ◦ Slides
  ◦ Video Recording
  ◦ SCALE Report
  ◦ MELCOR Report
	July 11, 2024

• Microreactor fuel cycle workshop	Coming in 2025
• Non-LWR Fuel Cycle Scenarios for SCALE and MELCOR Modeling Capability Demonstration ◦ Report	December 15, 2023

Public workshop videos, slides, reports at advanced reactor source term webpage

MODELING CAPABILITIES FOR TRISO AND METALLIC SPENT NUCLEAR FUEL

GORDON PETERSEN
SPENT FUEL ANALYST

2024 Workshop on Storage and Transportation of TRISO and Metal Spent Nuclear Fuels

U.S. DEPARTMENT OF ENERGY | Office of NUCLEAR ENERGY | SPENT FUEL & HIGH-LEVEL WASTE DISPOSITION

December 3-5, 2024

DISCLAIMER

This is a technical presentation that does not take into account contractual limitations or obligations under the Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste (Standard Contract) (10 CFR Part 961).

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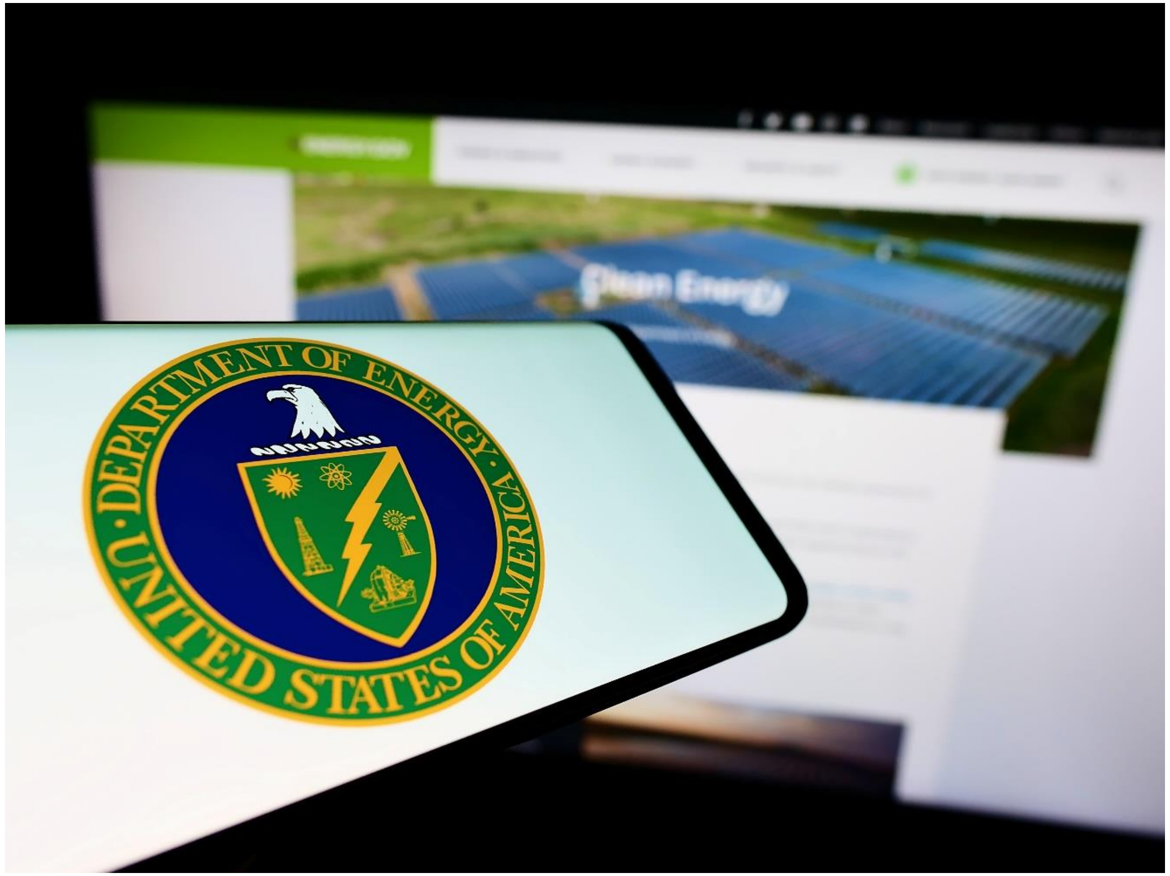


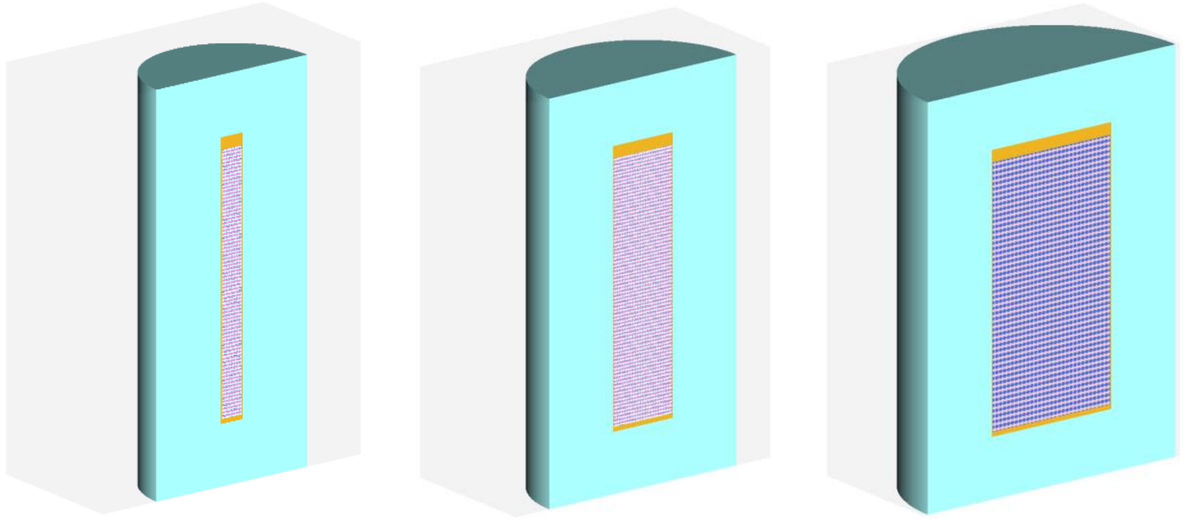
MODELING LIGHT-WATER REACTOR SPENT NUCLEAR FUEL

- Light-water reactor (LWR) spent nuclear fuel (SNF) has been packaged and modeled for decades
- Criticality Evaluations
  - Boiling-water reactor SNF models peak reactivity
  - Pressurized-water reactor utilizes burnup credit
  - Typical uncertainty between 500–800 per cent mille (pcm) for LWR [1]
- Dose/Decay Heat Evaluations
  - Decay heat uncertainty ~2% [2]

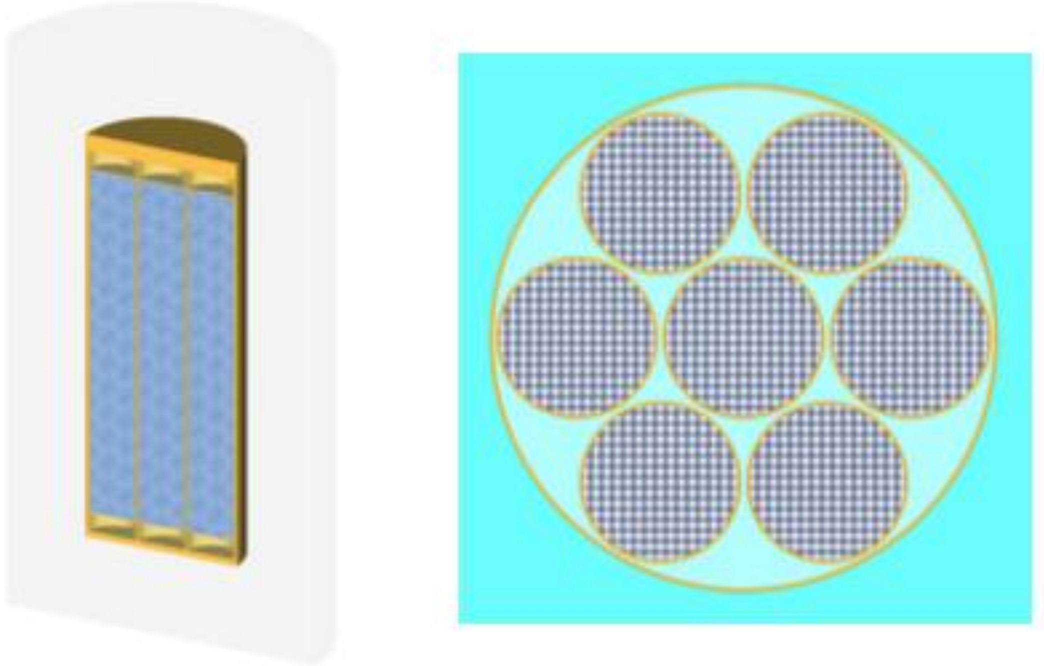
Dry storage systems in the U.S.

MODELING TRISO SNF

- Performed potential packaging analyses for TRi-structural ISOtropic (TRISO) SNF
- Criticality evaluations
  - Top contributors to uncertainty [1,2]
    - ^{235}U, ^{238}U, and graphite*****
  - Uncertainty expected to be less than 1,000 pcm [1,2] for SNF packages
- Dose/decay heat evaluations
  - TRISO SNF is expected to have lower dose and decay heat compared to LWR SNF.
- No significant modeling challenges expected in packaging TRISO SNF for extended storage and transportation.



12 in. Canister | 36 in. Canister | 72 in. Canister

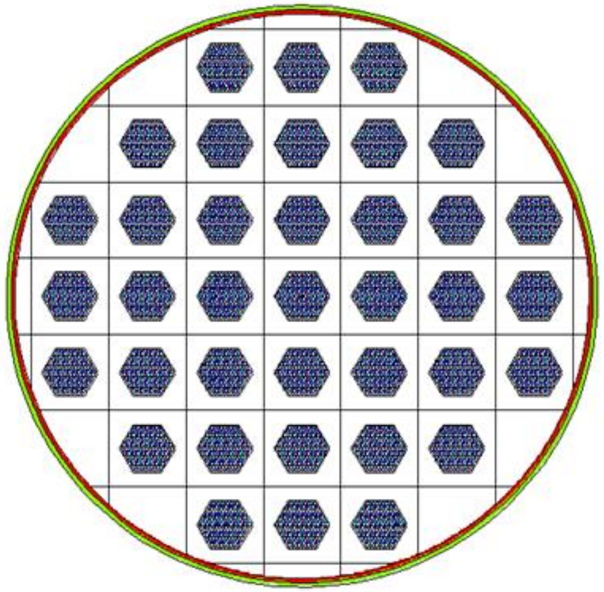


Multiple DOE Standard Canisters in a large diameter canister

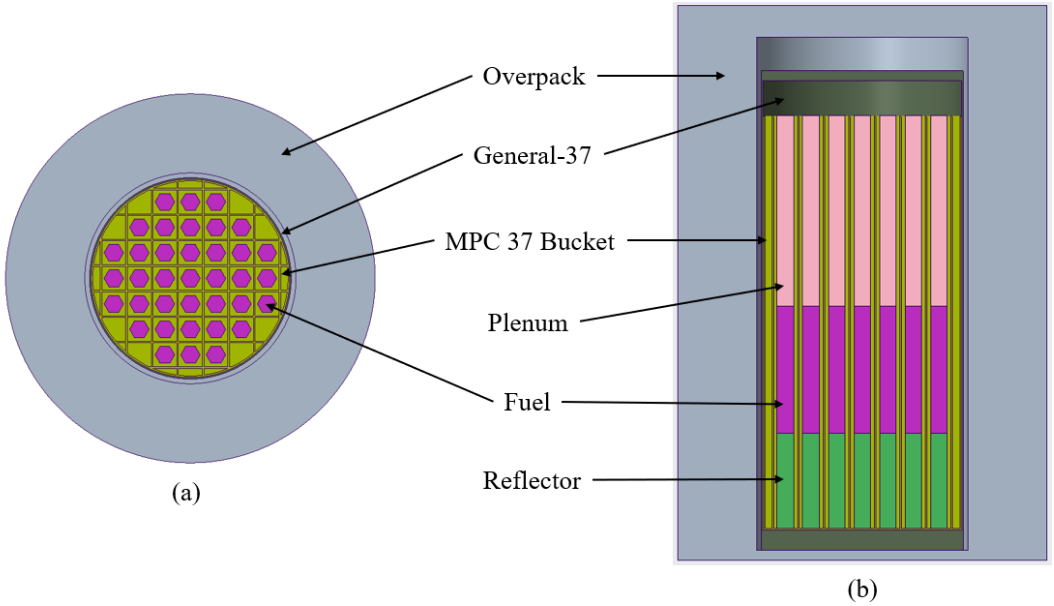
* Due to the large amount of graphite and other carbon non-fuel layers

MODELING METALLIC SNF

- Performed potential packaging analyses for metallic SNF
- Criticality evaluations
  - Top contributors to uncertainty [1]
    - ^{235}U, ^{238}U, and ^{56}Fe*****
    - Uncertainty of sodium fast reactor expected to be between 1000-1500 pcm.
- Dose/decay heat evaluations
  - Decay heat and dose largely depends on configuration.
- No significant modeling challenges expected in packaging metallic SNF for extended storage and transportation.



Radial and axial representations for metallic SNF from a sodium fast reactor.



* Due to the structural materials being composed of iron-based alloys

U.S. DEPARTMENT OF ENERGY | Office of NUCLEAR ENERGY | SPENT FUEL & HIGH-LEVEL WASTE DISPOSITION

REFERENCES

1. Bostelmann, F., Illas, G., Wieselquist, W. "Key Nuclear Data Impacting Reactivity in Advanced Reactors." ORNL/TM-2020/1557, June 2020.

2. Ebiwonjumi, B., Kong, C., Zhang, P., Cherezov, A., Lee, D. "Uncertainty Quantification of PWR Spent Fuel Due to Nuclear Data and Modeling Parameters." Nuclear Engineering and Technology, Volume 53, Issue 3, 2021.

3. Wing, J., Maldonado, G. I., Petersen, G., Joseph, R. "Uncertainty Quantification for Pebble Bed Reactor Fuels Burnup Credit." Transactions, Volume 130, Number 1, June 2024, Pages 168-171.

TRISO AND METAL SPENT NUCLEAR FUEL (SNF) CANISTER DECAY HEAT

LAURA PRICE

Principal Member of Technical Staff
Sandia National Laboratories

U.S. DEPARTMENT OF ENERGY | Office of NUCLEAR ENERGY | SPENT FUEL & HIGH-LEVEL WASTE DISPOSITION

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December 3-5, 2024

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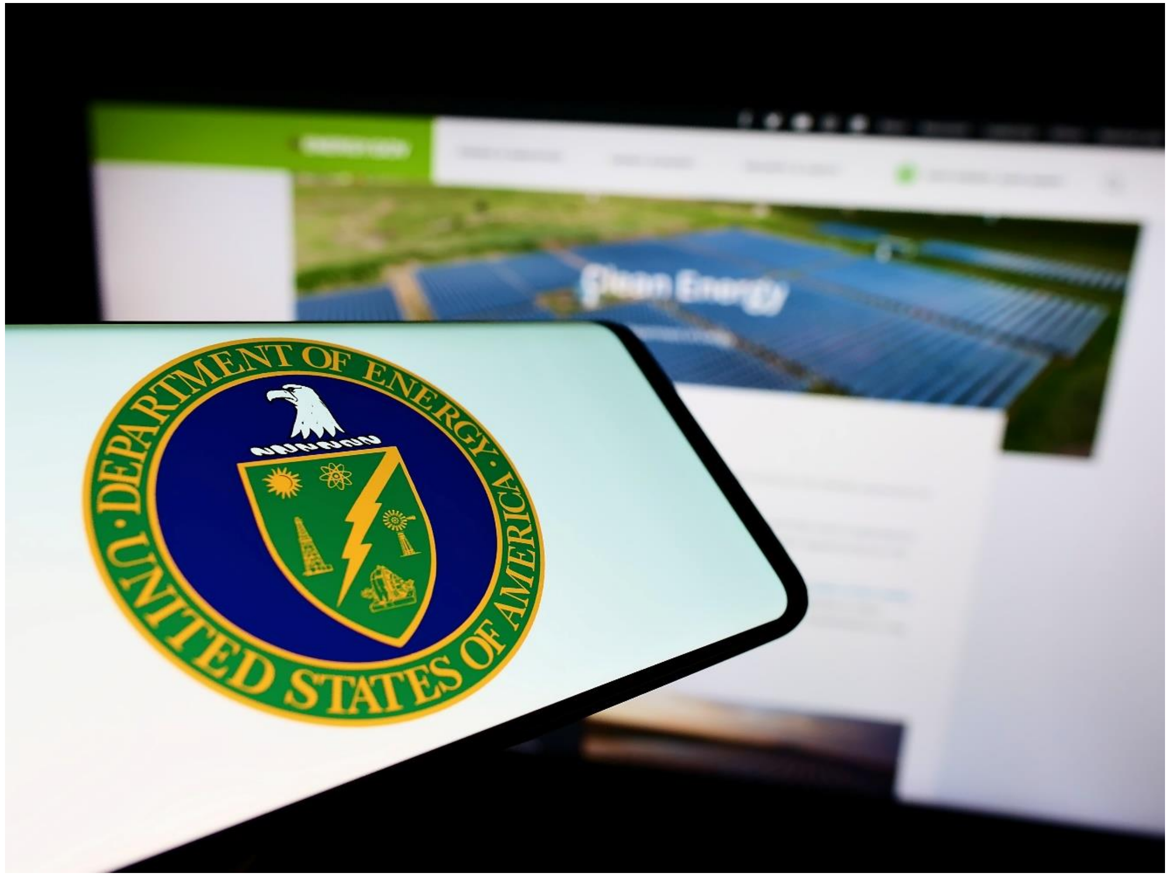
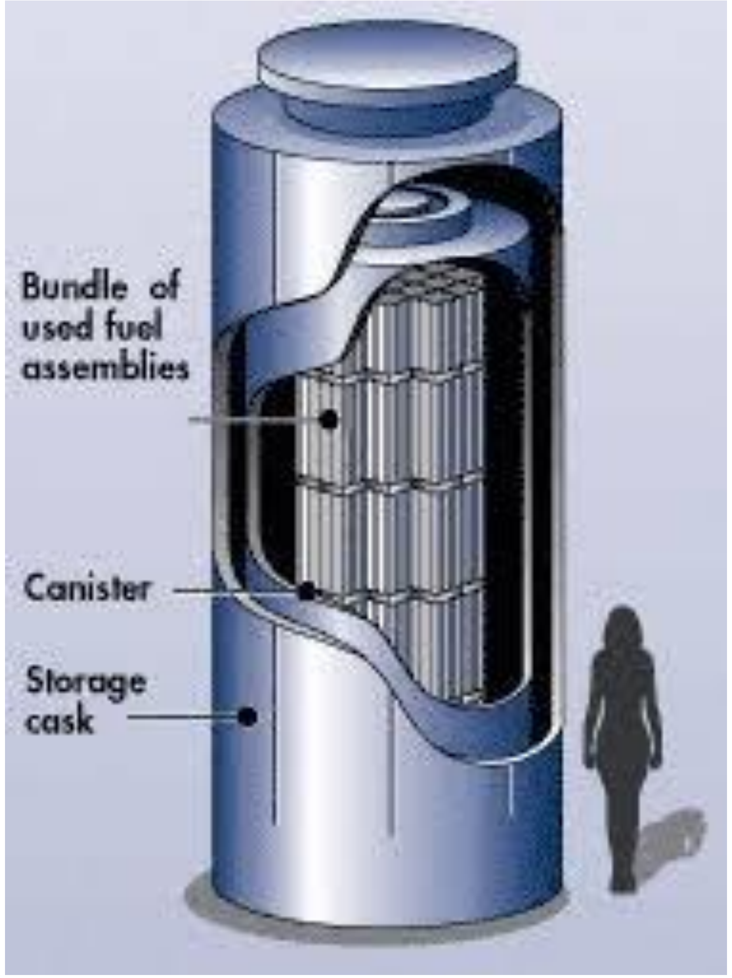


COMPARISON OF TYPICAL LIGHT WATER REACTOR SPENT NUCLEAR FUEL (SNF) WITH TRISO AND METALLIC SNF

- Hoffman et al. (2024) compared the characteristics of three different types of proposed irradiated TRISO fuels and one type of proposed irradiated metallic fuel (sodium free) to those of “typical” LWR SNF.
- Characteristics considered
  - Physical dimensions
  - Isotopic composition and fissile inventory
  - **Decay heat of SNF in Canisters – drives loading limits for storage, transportation, and disposal**
  - Radiation source strength
  - A2 value
  - Energy generation
- Comparison made on a basis of what could fit in a 37-PWR canister; done for comparison purposes only

TYPICAL 37-PWR CANISTER SYSTEM



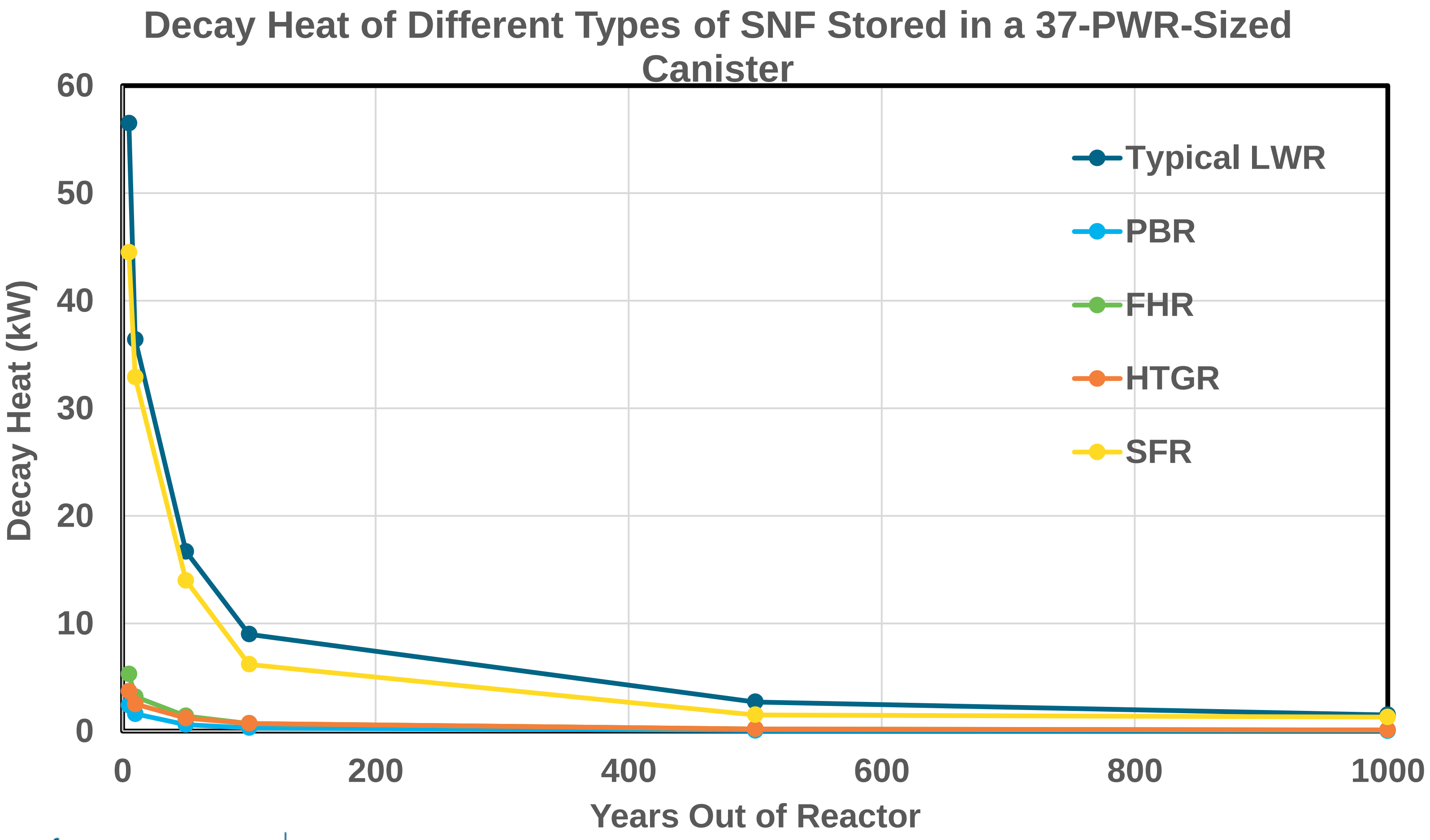
Picture on left: Courtesy of the NRC
Above picture: Courtesy of Holtec International

SELECTED SNF PARAMETERS

	Typical LWR	Pebble Bed Reactor (PBR)	Fluoride-salt-cooled high-temperature reactor (FHR)	Prismatic block high-temperature gas reactor (HTGR)	Metallic sodium-cooled fast reactor (SFR)
Heavy metal loading in 37-PWR-size canister (initial MTU)	19.9	0.3	0.5	0.6	8.3
Average proposed discharge burnup (GWd/MTU)	50	165	180	120	147.3

Source: Hoffman et al. (2023)

SNF CANISTER DECAY HEAT



REFERENCES

Hoffman, E., Kim, T.K., and Price, L., 2024. “Characteristics of Potential Significance in Waste Management from HALEU Spent Fuel – 24226,” Waste Management Conference, March, 2024,Phoenix, AZ, USA.

LEARN MORE

Office of Spent Fuel & High-Level Waste Disposition

energy.gov/ne/office-spent-fuel-and-high-level-waste-disposition

U.S. DEPARTMENT OF ENERGY
Office of NUCLEAR ENERGY
SPENT FUEL & HIGH-LEVEL WASTE DISPOSITION
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Storage and Transportation of TRISO and Metal Spent Nuclear Fuels

NRC's simulation capabilities supporting fuel & cladding performance modeling for metallic fueled non-LWRs

James Corson, Ph.D.

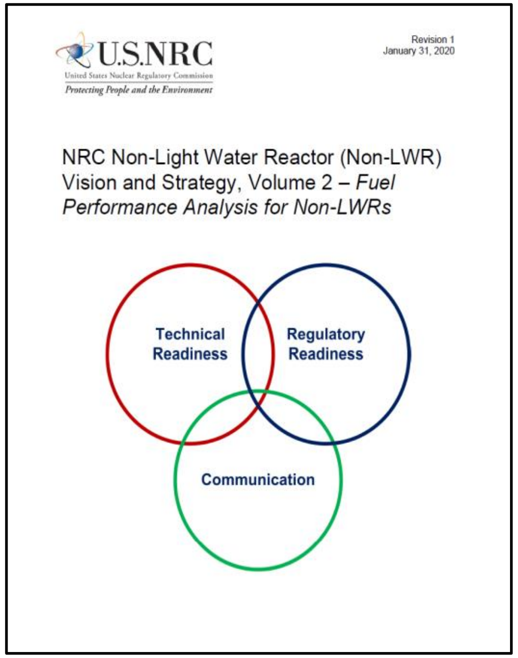
Office of Nuclear Regulatory Research
Division of Systems Analysis
Fuel & Source Term Code Development Branch

Objectives

- *NRC's simulation capabilities for modeling metallic fuels*
  - *Fission Gas Release*
  - *Fuel Swelling*
  - *Cladding Deformation*

- *Overview of data availability, gaps, and where additional data would be beneficial*

Non-LWR Fuel Performance Analyses



ML20030A177

NRC's comprehensive fuel performance code

- *Models the thermal-mechanical response of nuclear fuel*
- *Is used for normal operations, anticipated operational occurrences, accident conditions, and spent fuel storage*
- *Is used for LWR & non-LWR fuel types*

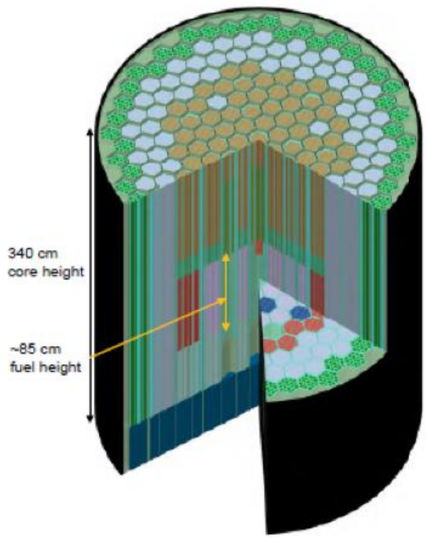
Non-LWR demonstration project for fuel performance

- *Developed new models for TRISO and metallic fuel designs*
- *Performed assessments & validation activities with available experimental data*

Non-LWR demonstration project for fuel performance has improved and validated FAST for simulating non-LWR fuel designs, including metallic fuel designs for use in SFRs.

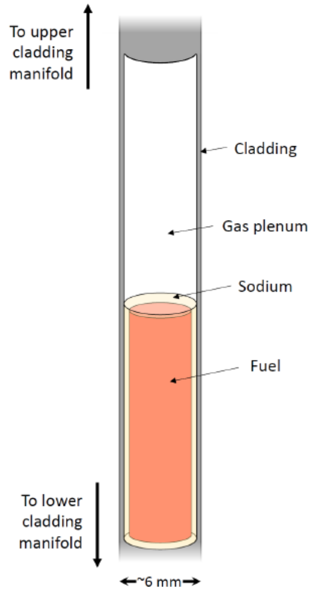
Non-LWR Fuel Designs & Phenomena Relevant to Safety

Sodium-Cooled Fast Reactor



- Pool-type reactors, utilizing metallic fuel designs.
- Fueled with metallic slugs of U-Zr or U-Pu-Zr
- High-temperature steel claddings (e.g., HT-9)
- Liquid sodium bond & coolant

Uranium Metal Fuel Rods



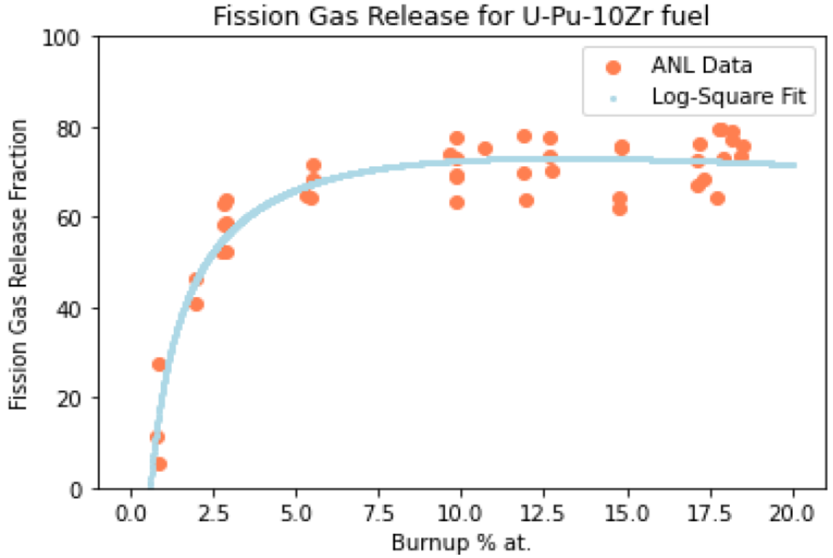
- Fuel pins designed with adequate smear density to accommodate U-Pu-Zr fuel swelling
- Fuel also designed with large plenum to accommodate fission gas release
- Metallic fuel rods operated successfully in Experimental Breeder Reactor II (EBR-II)

Phenomena Relevant to Safety

- Impact of temperature and irradiation on material properties
- Radial redistribution of fuel constituents and impacts on local melting / eutectic temperatures, power distribution, fuel swelling, and fuel-cladding chemical interaction
- Fission product migration, diffusion, and fission gas release

Metallic Fuel Models in FAST

- Existing U-10Zr fuel, HT-9 cladding models are empirical, based primarily on EBR-II experience
  - Anisotropic fuel swelling fitted to experimental data
  - Fission gas release fitted to experimental data
  - Material properties for HT-9 cladding
    - Thermal conductivity, specific heat capacity, melting temperature, thermal expansion, emissivity, density, Young's modulus, creep, etc.



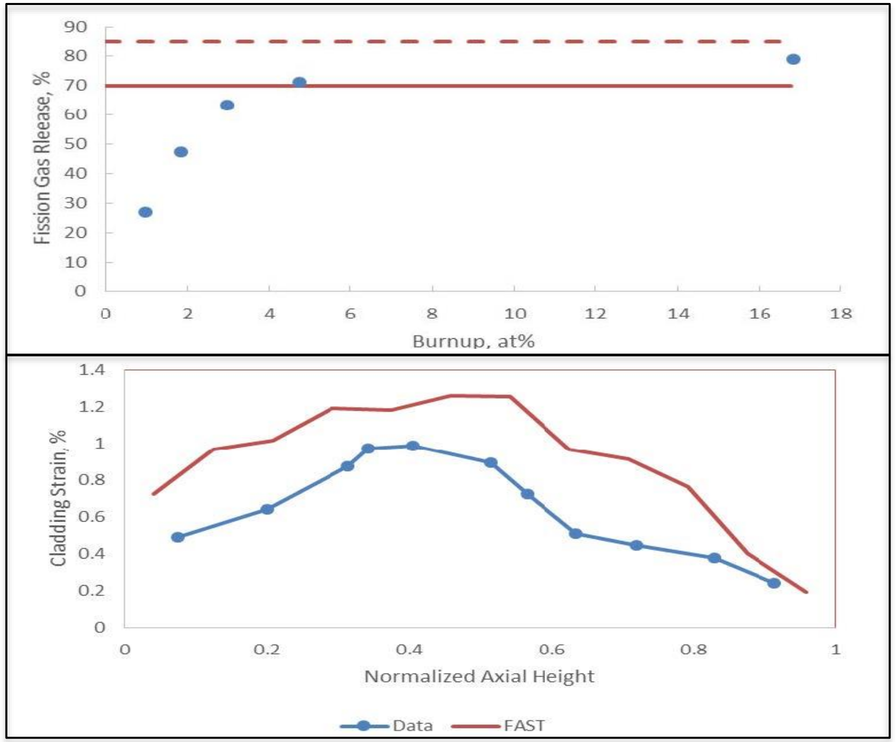
FGR data from Pahl et al., *JNM* 188 (1992) 3

- Future work needed for fuel failure models and to extend beyond the existing database
  - Fuel clad chemical interaction (FCCI) model
  - Cladding overpressure failure models
  - Release of fission products other than noble gases (e.g., cesium, iodine) from the fuel
    - Already covered by MELCOR for accident conditions

The existing framework of FAST has been leveraged for modeling metallic fuel forms. New material property and phenomenological models have been implemented for modeling in-reactor metal fuels.

Preliminary FAST Assessment on FGR – Metal Fuels

- FAST Initial Assessment on Metallic Fuel - 2018
  - Included constant swelling and FGR rates
- Reassessment using new FGR model in progress
  - Improved models can reduce uncertainties



Geelhood & Porter, Top Fuel 2018

Validation and assessment activities are leveraging historic SFR fuel performance data which includes historic EBR-II.

LWR Spent Fuel Performance Analyses

- FAST has been used to support LWR spent fuel analyses, by determining
  - Initial conditions to support cask analyses (e.g., end-of-life fuel characteristics)
    - Cladding oxide thickness and hydrogen content
    - HBU mechanical properties
    - Rod internal pressure
  - Initial conditions for creep rupture analyses

- Recent updates in FAST enhanced LWR spent fuel analyses
  - Ability to change ex-reactor boundary conditions
  - New backend, ex-reactor spent fuel models
    - cladding creep models
    - helium production and release model
    - pellet swelling model
  - New ex-reactor cladding creep rupture criteria

While ex-reactor modeling enhancements have focused on LWR fuels, these capabilities can be leveraged to support non-LWR ex-reactor spent fuel modeling.

Applying FAST to Metallic Fuel Storage and Transportation

- To date, code development and assessment has focused on in-reactor behavior
  - However, the code addresses phenomena that are also important during storage and transportation conditions (e.g., fission gas release, cladding mechanical deformation and integrity)
  - The code also provides initial conditions (e.g., rod internal pressure, moles of fission gas available for release) at start of storage or transportation

- Using FAST for metallic fuel storage and transportation has some challenges
  - Very little data available for metallic fuel behavior under storage conditions
    - Such data would be useful for validating models in FAST
  - Representative temperatures during storage or transportation conditions must be provided as input to the code
    - Can be provided from other codes that can calculate expected temperatures
    - Can also be taken from imposed limits

Advanced Fuels Campaign

Metal SNFs: Fission product diffusion

Presenter: Tiankai Yao

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Decay heat and its effect on FFTF sized U-10Zr fuel pin during storage and accidents

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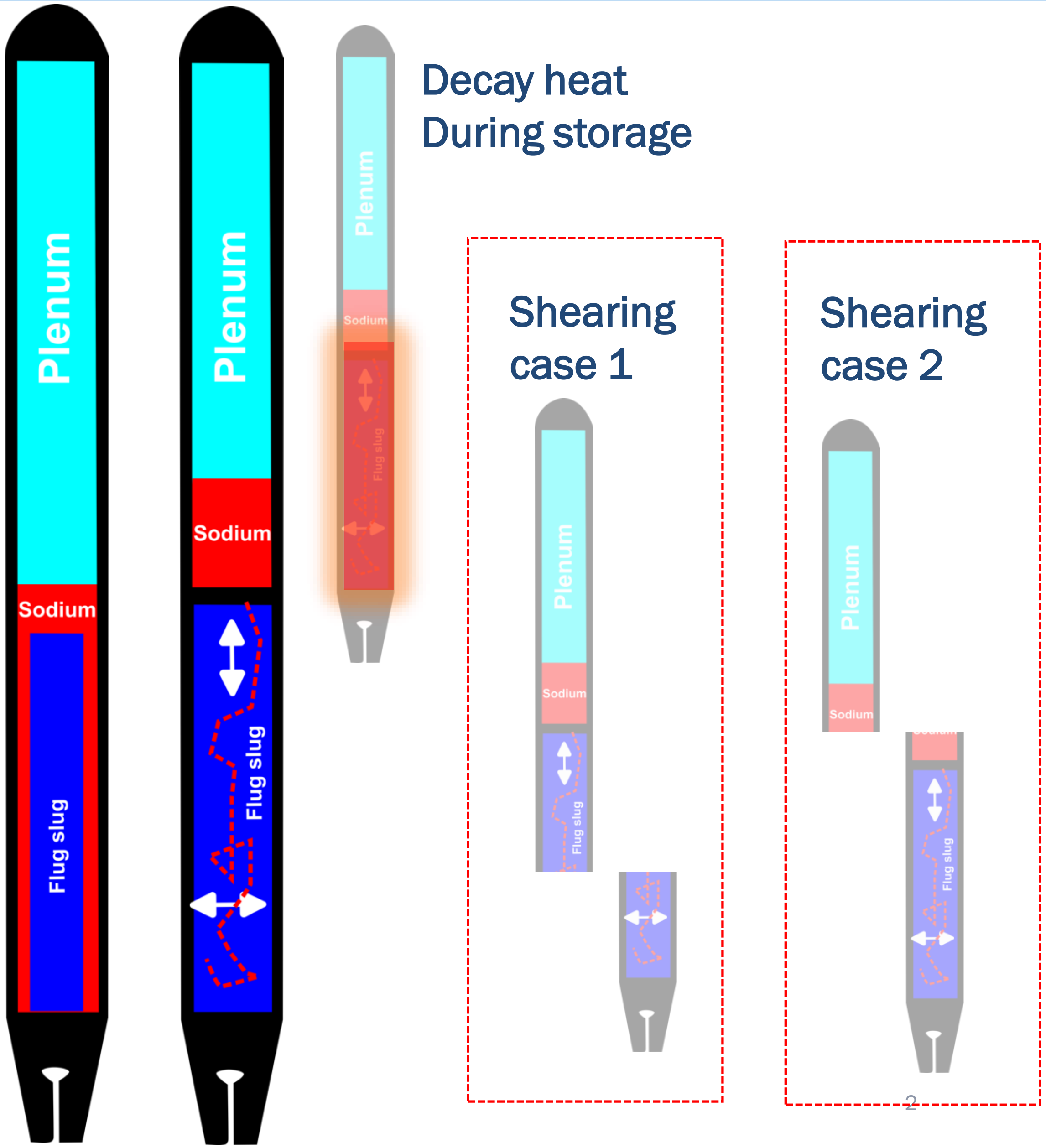

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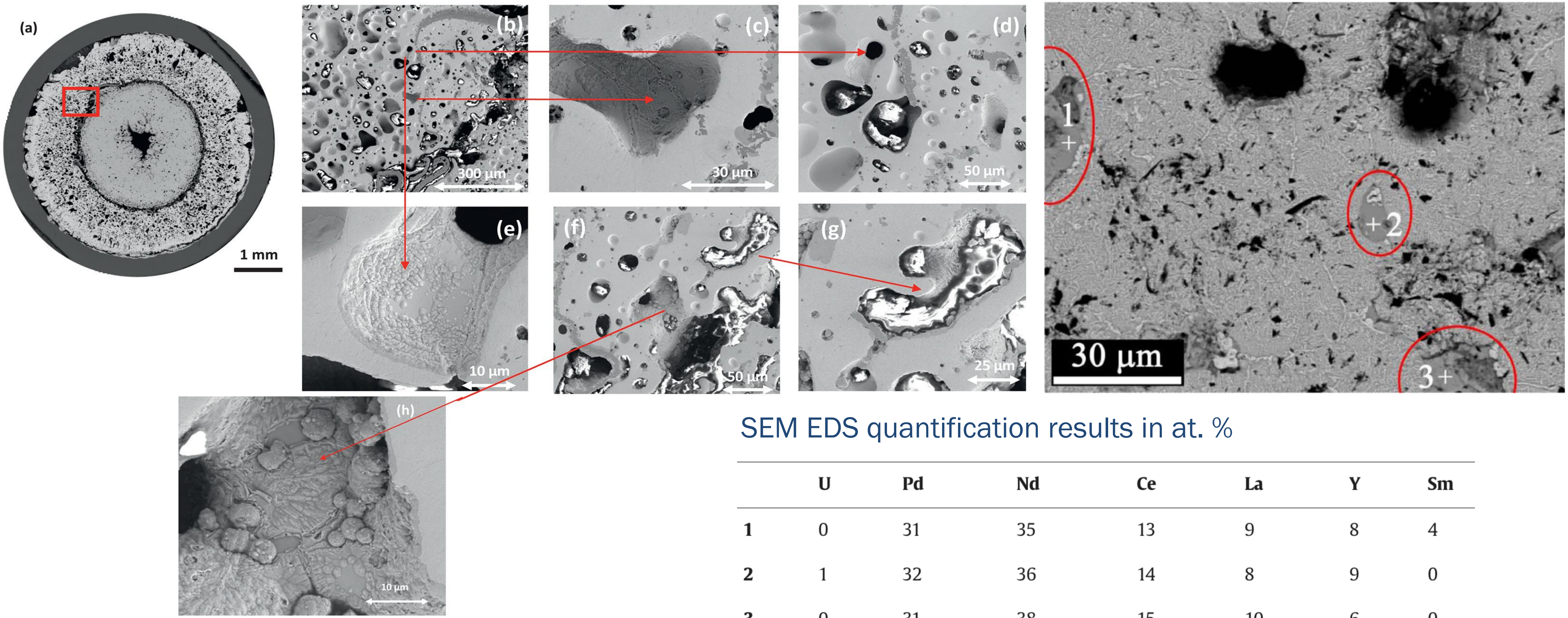
Review of FFTF fuel experiments for storage in Interim storage casks (ISC), 2005, by S. Chastain

Fission product in metal SNF



SEM EDS quantification results in at. %

	U	Pd	Nd	Ce	La	Y	Sm
1	0	31	35	13	9	8	4
2	1	32	36	14	8	9	0
3	0	31	38	15	10	6	0

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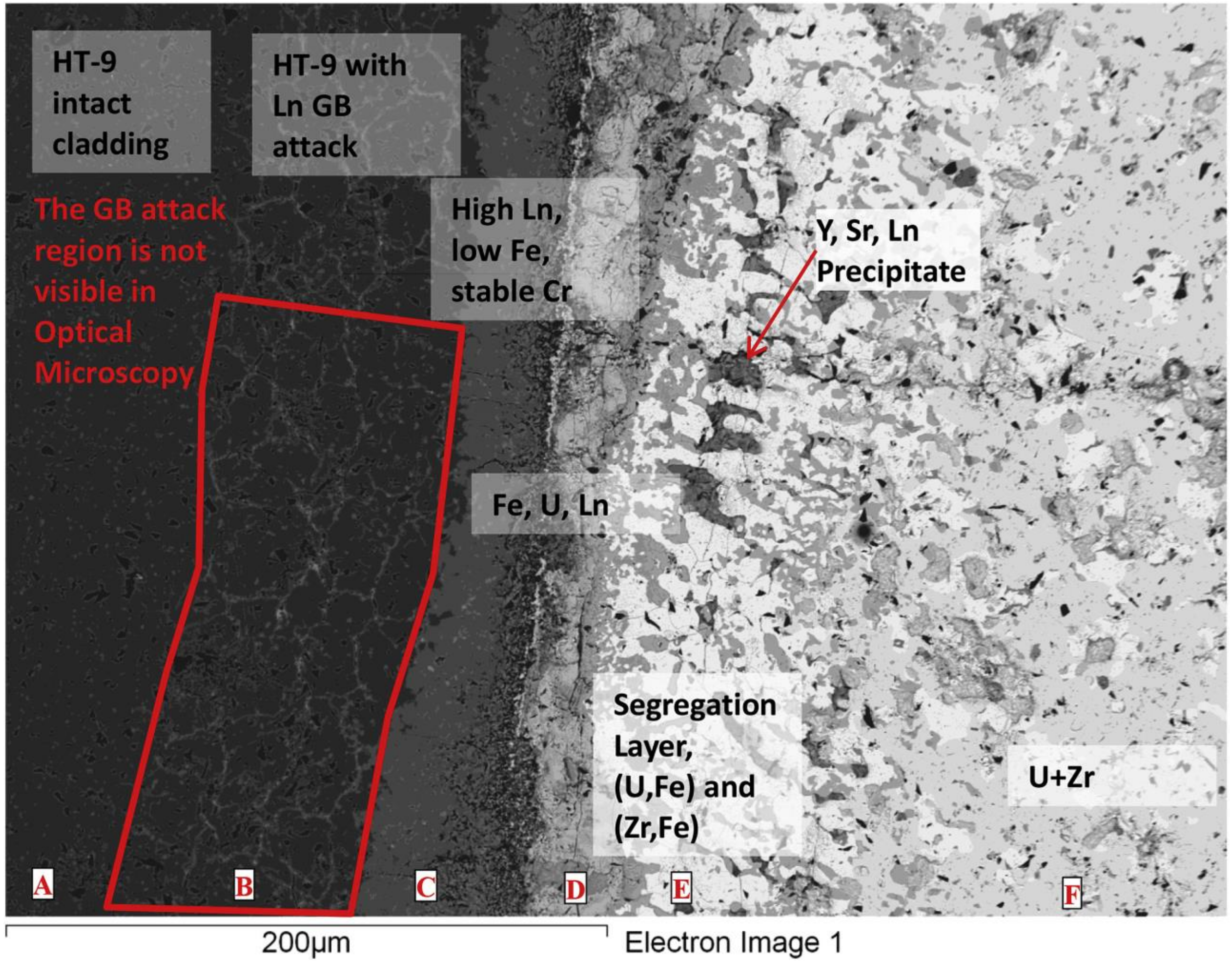
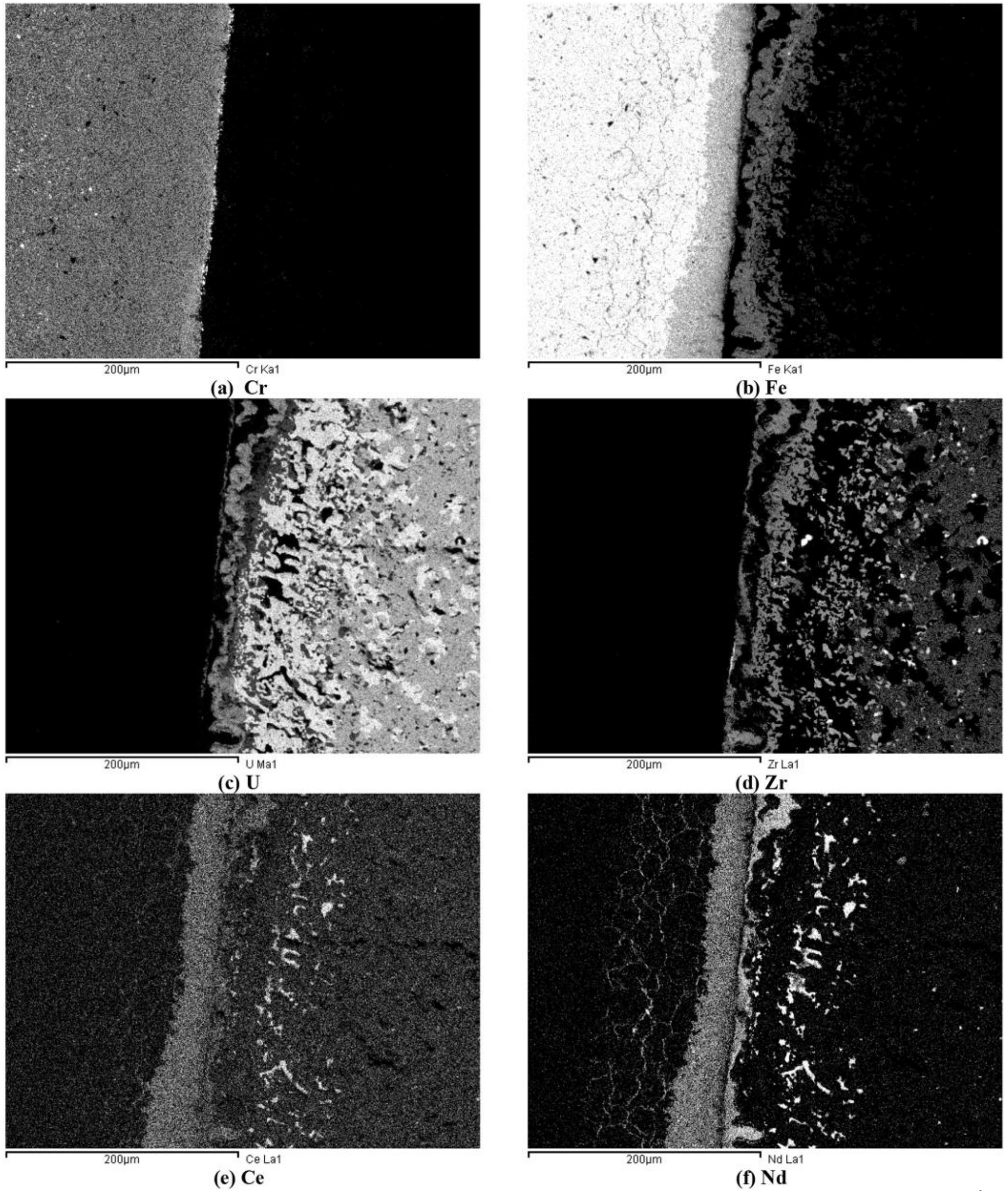
https://doi.org/10.1016/j.matchar.2021.111657
https://doi.org/10.1016/j.jnucmat.2017.07.040

Lanthanide accumulated at fuel cladding interface

U-10Zr irradiated in FFTF, 5.7% burnup
Local inner cladding temperature of 615 ℃





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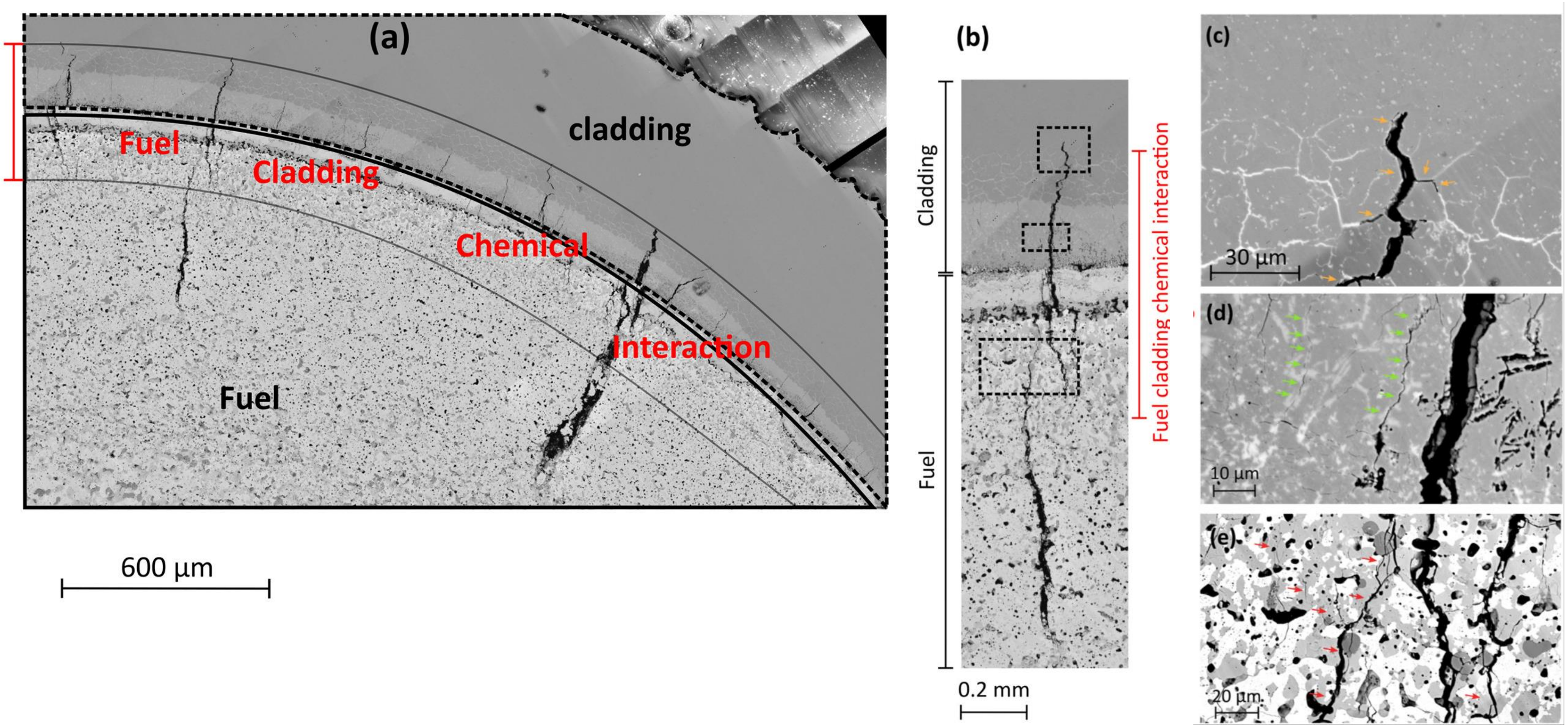
https://doi.org/10.1016/j.jnucmat.2017.07.040

Lanthanide move though the cladding



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https://doi.org/10.1016/j.jnucmat.2022.153990

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Metal SNFs: corrosion

Authors and contributors

Presenter: Tiankai Yao

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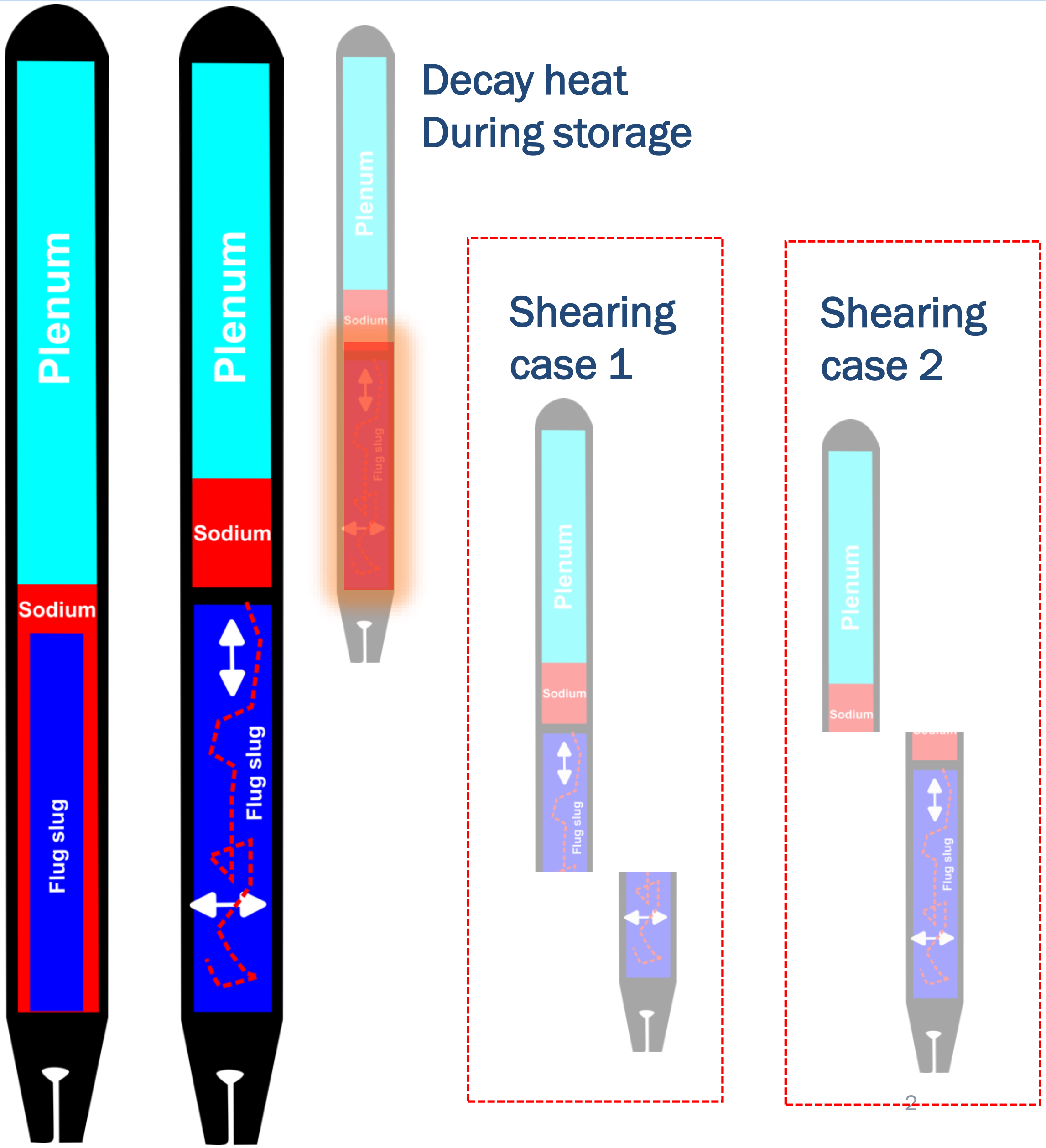

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Review of FFTF fuel experiments for storage in Interim storage casks (ISC), 2005, by S. Chastain

High temperature steam oxidation

$$4H_2O + 3\,Fe \rightarrow Fe_3O_4 + 4\,H_2$$

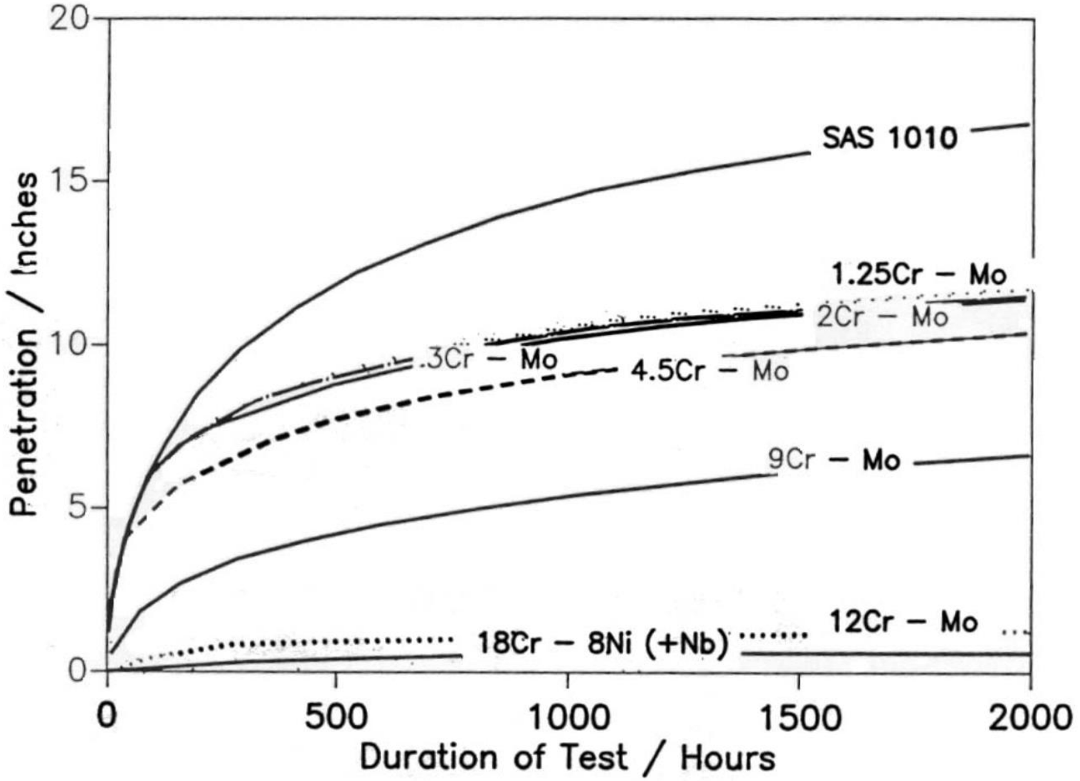

Figure 1 Corrosion of steel bars in contact with steam at 593 °C for 200, 500, 1000 and 2000 h (after Solberg *et al* [1])

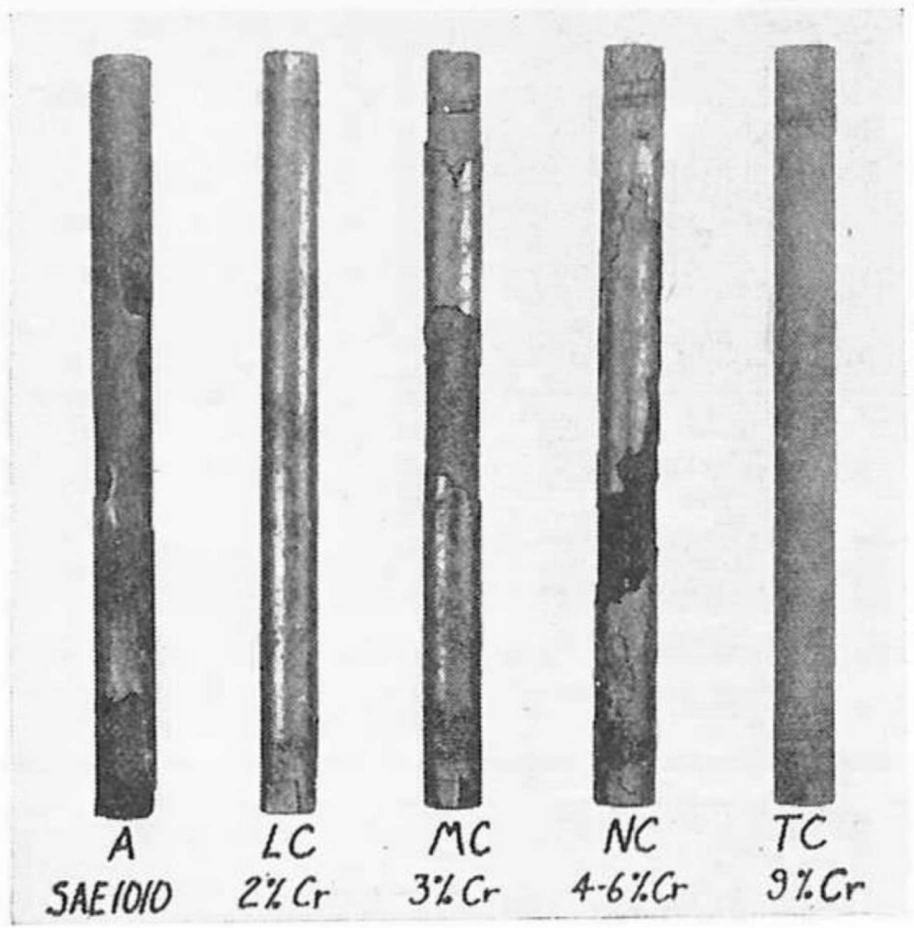

FIG. 12 SPECIMENS EXPOSED TO STEAM AT 1200 F FOR 1200 HR
(Specimens cooled to room temperature every 50 hr, after a continuous exposure to steam at 1200 F for 500 hr.)

Long term corrosion of Cr-Mo Steels in steam at 482 and 538 ℃ up to 28, 000 hrs

- 3.2 years experiments

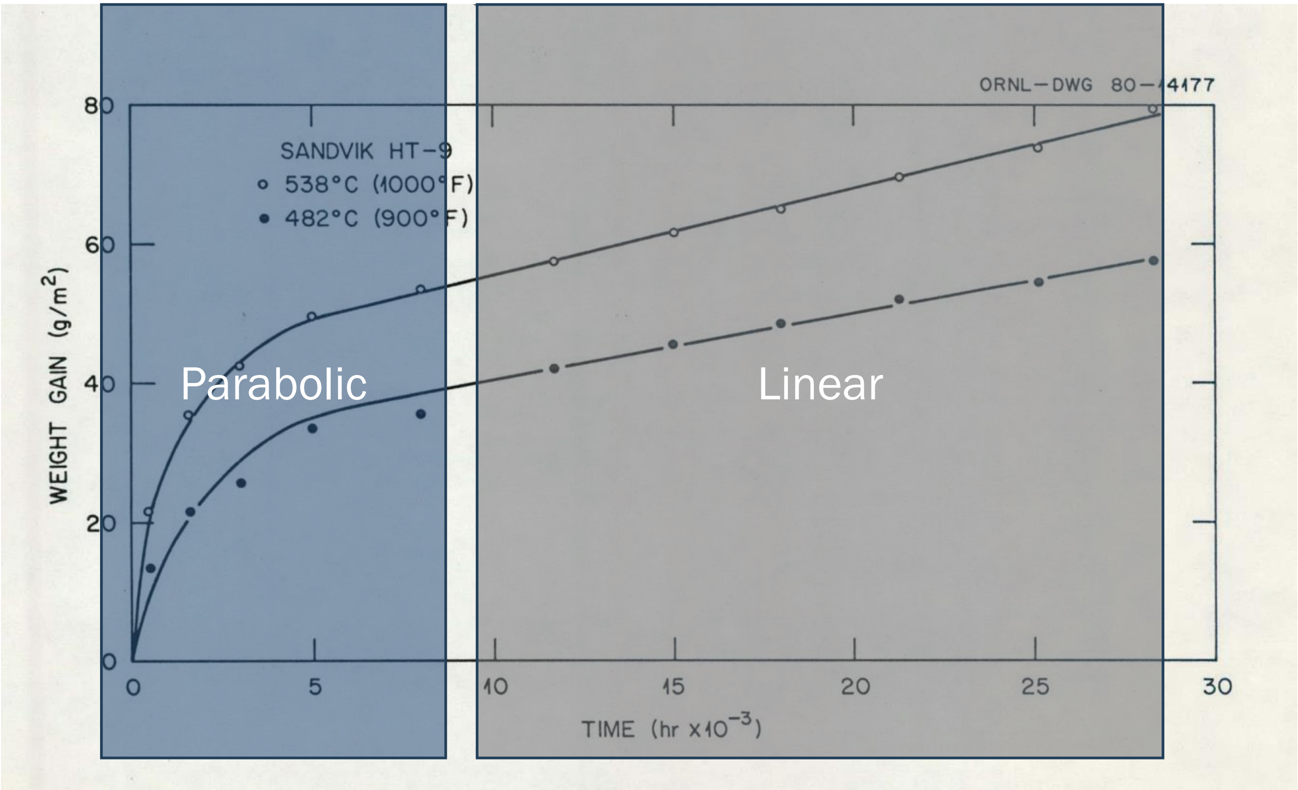

Table 3. Calculated Penetrations and Corrosion Rates for 9 Cr Steel Specimens with Ground Surfaces in Superheated Steam

Total Test in Time (h)	Heat Number of Test (alloy)	Penetration Calculated from Extrapolation of Weight Gain–Time Curves (μm)		Corrosion Rate After First Year (μm/year)	
		482°C	538°C	482°C	538°C
28,339	316381A[a]	11.9	18.2	3.2	5.1
27,835	316381A[a]	11.3	19.6	2.9	5.0
26,719	316381A[a]	10.6	18.2	3.3	5.1
23,267	316381A[a]	9.0	16.6	3.5	6.5
23,267	XA-3177[b]	9.6	21.6	4.0	6.7
23,267	XA-3178[b]	9.6	21.9	4.0	7.1
23,267	91887[b]	10.0	24.5	4.4	8.7

[a]Individual specimens.
[b]Average of three specimens.

3-4 μm/year at 482 ℃
5-8 μm/year at 538 ℃

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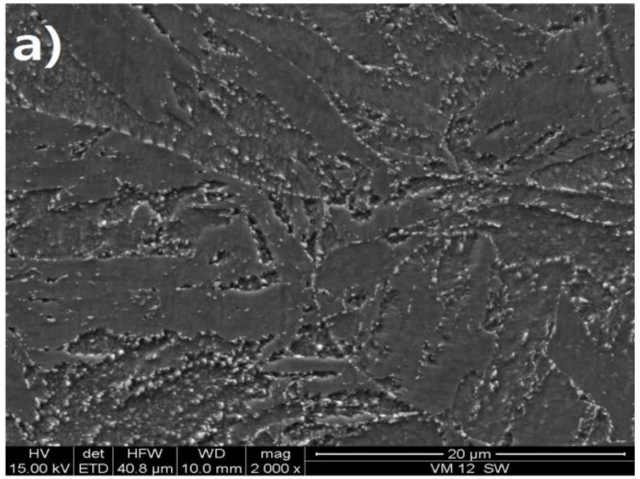
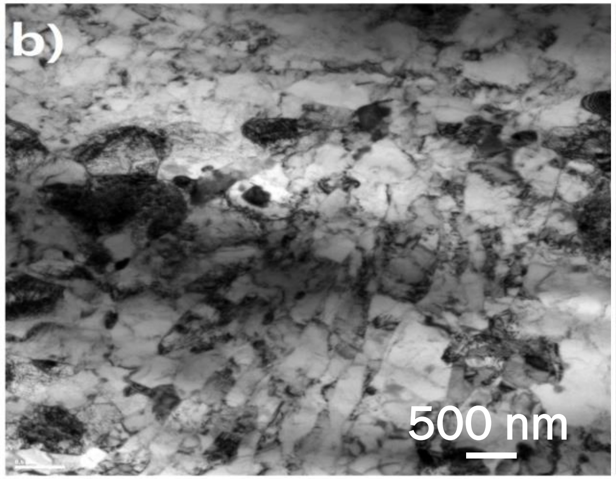
Pure thermal annealing effects in inert environment for a 12Cr steel

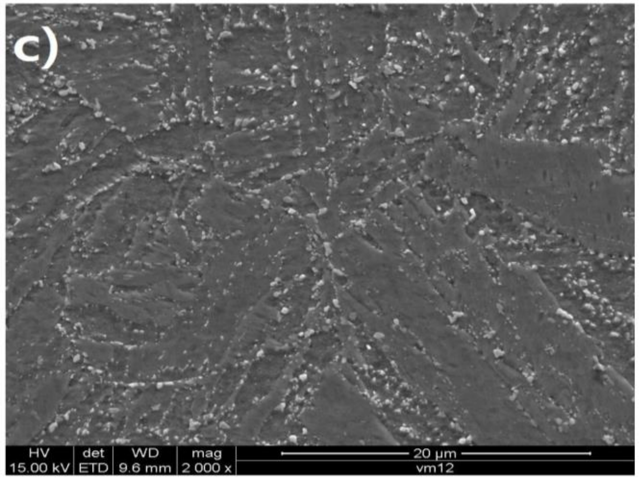
Steel grade	Content of elements [%]												
	–	C	Mn	Si	P	S	Cr	Mo	V	W	Nb	B	N
VM12	Mat. testing	0.13	0.22	0.48	0.01	0.01	11.40	0.27	0.22	1.30	0.05	0.003	0.05
	according to [9]	0.11-0.14	0.15-0.45	0.40-0.60	max 0.02	max 0.01	11.0-12.0	0.20-0.40	0.20-0.30	1.30-1.70	0.03-0.08	0.003-0.006	0.03-0.07



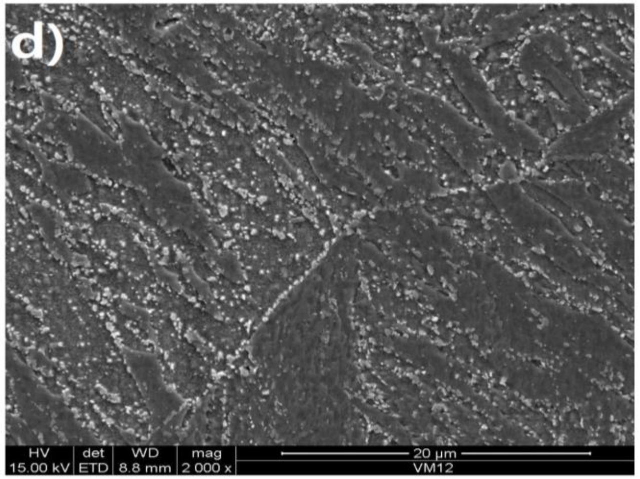
VM 12 As Received



VM 12 As Received (TEM)



After 30, 000 hours @ 600 °C



After 30, 000 hours @ 650 °C

as-received condition VM12	$M_{23}C_6$ – main phase MN	—	—
annealing 10,000h/600°C VM12	Main phase $M_{23}C_6$ (Fe_2Mo) – small amount, NbCrN – medium	annealing 10,000h/650°C VM12	Main phase $M_{23}C_6$ medium – Laves phase Fe_2Mo NbCrN
annealing 30,000h/600°C VM12	Main phase $M_{23}C_6$ (Fe_2Mo) – small amount, NbCrN – medium	annealing 30,000h/650°C VM12	Main phase $M_{23}C_6$ medium – Laves phase Fe_2Mo NbCrN

Comparison of the results of mechanical properties investigations of VM12 steel after long-term annealing

	As-received condition	Annealing 1000h/600°C	Annealing 5000h/600°C	Annealing 10000h/600°C	Annealing 30000h/600°C
TS, [MPa]	750	706	702	724	723
YS, [MPa]	571	514	512	540	533
El., [%]	23	25	25	23	23
YS^{550}, [MPa]	338	332	338	329	315
HV10	259	244	235	239	218
Impact energy [KV], J	78	49	43	41	38

DOI: 10.1515/amm-2016-0163

CHANGES IN PROPERTIES AND MICROSTRUCTURE OF HIGH-CHROMIUM 9-12%CR STEELS DUE TO LONG-TERM EXPOSURE AT ELEVATED TEMPERATURE

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Pure thermal annealing effects in inert environment for HT9 steel

Table 1
Examples of precipitation and phases observed in high-chromium F/M steels [8].

Precipitate or phase	Crystal structure	Typical composition	Temperature range (°C)
$M_{23}C_6$	FCC	$(Cr_{16}Fe_6Mo)C_6$ $(Cr_4Fe_{12}Mo_4Si_2WV)C_6$	400–900
MX	FCC	NbC, NbN, VN, (CrV)N, Nb (CN) and (NbV)C	<1200
Laves	HCP	Fe_2Mo, Fe_2W and $Fe_2(Mo,W)$	450–700
α’	BCC	Fe(Cr, Mo, Si, W) (Cr-rich ferrite)	400–550
σ-Phase	Tetragonal	FeCr, FeCrW	600–900

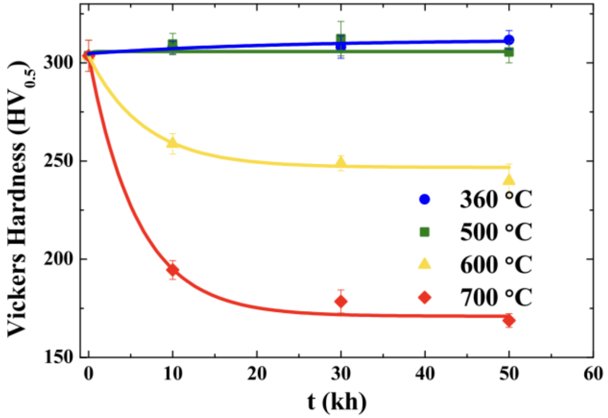

Fig. 10. Average Vickers microhardness of HT9 as a function of aging time and temperature.

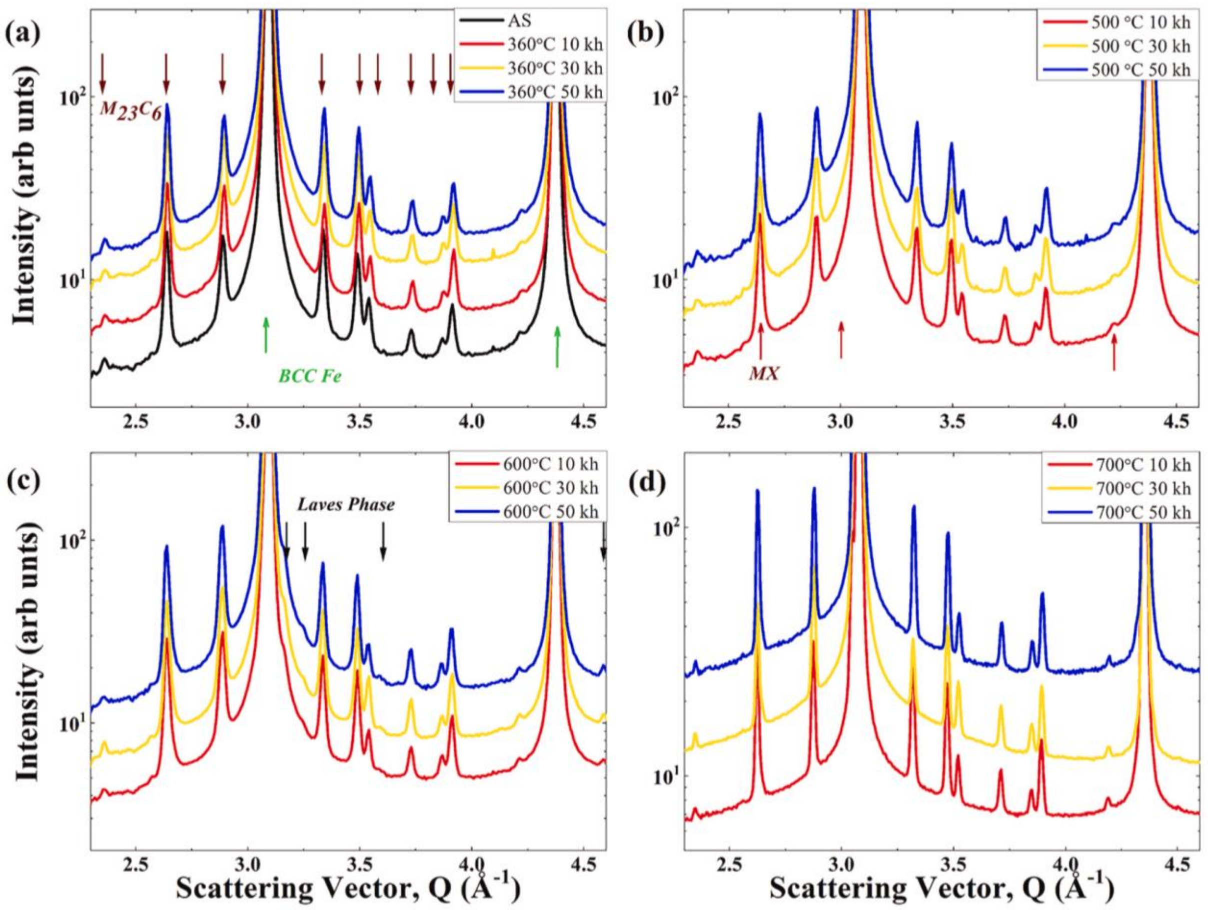

Fig. 7. XRD patterns for HT9 specimens after 10 kh, 30 kh and 50 kh of aging at (a) 360 °C, (b) 500 °C, (c) 600 °C and (d) 700 °C. Arrows in panels (a)-(c) indicate the different crystal phases – BCC Fe, FCC $M_{23}C_6$, FCC MX and HCP Fe_2Mo.

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https://doi.org/10.1016/j.matchar.2024.114418

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Metal SNFs: Reactions with Water

Authors and contributors

Presenter: Tiankai Yao

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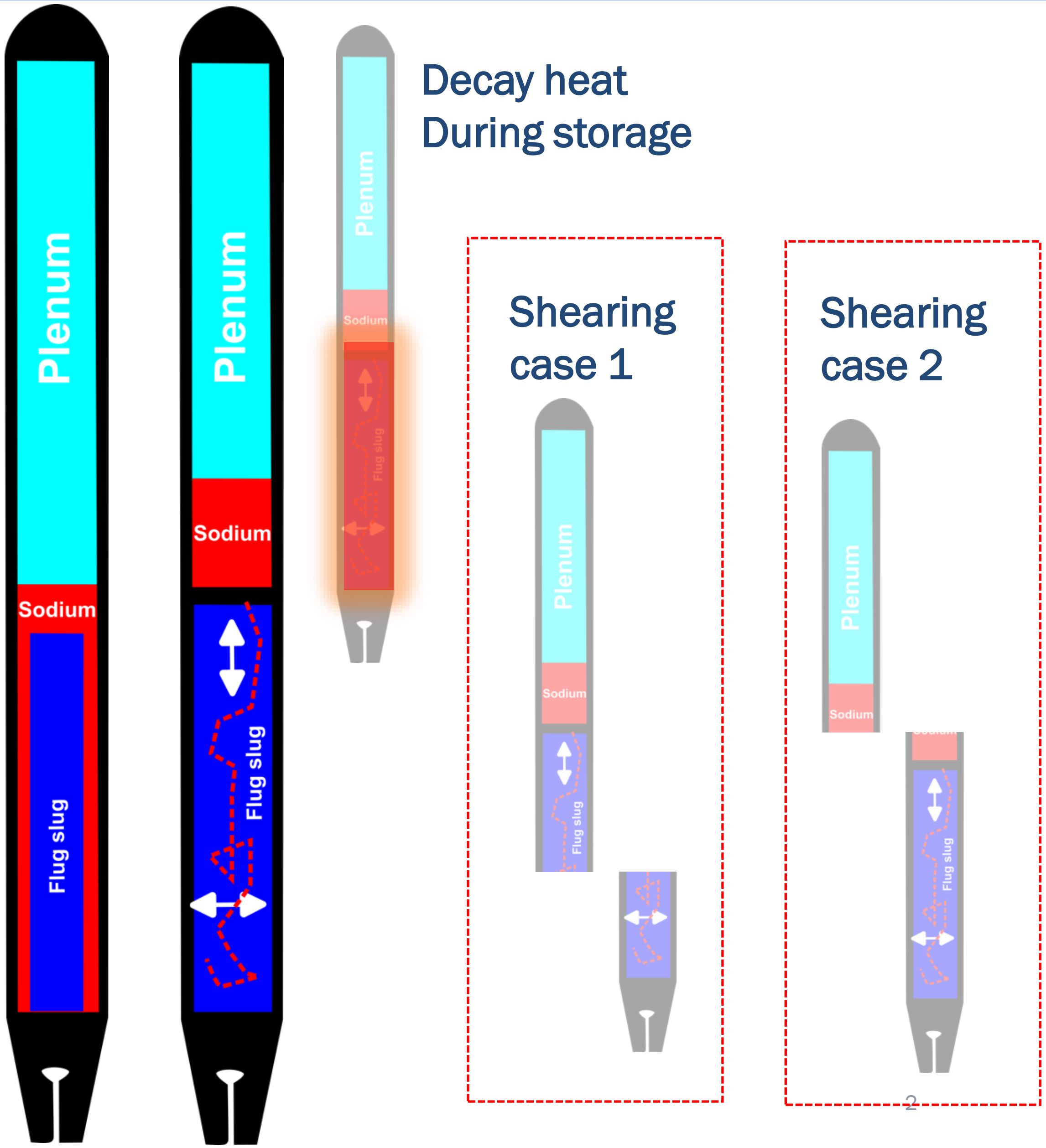

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Review of FFTF fuel experiments for storage in Interim storage casks (ISC), 2005, by S. Chastain

Past experiences for Uranium metal reaction with water

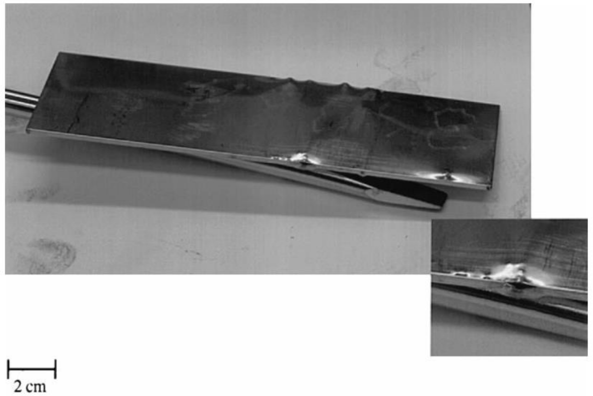


Fig. 2. Bulged ZPPR plate (3606); inset shows close-up of cladding breach.

$$U + (2+x)H_2O \rightarrow UO_{2+x} + (2+x)H_2$$

$$2U + 3H_2 \rightarrow 2UH_3$$

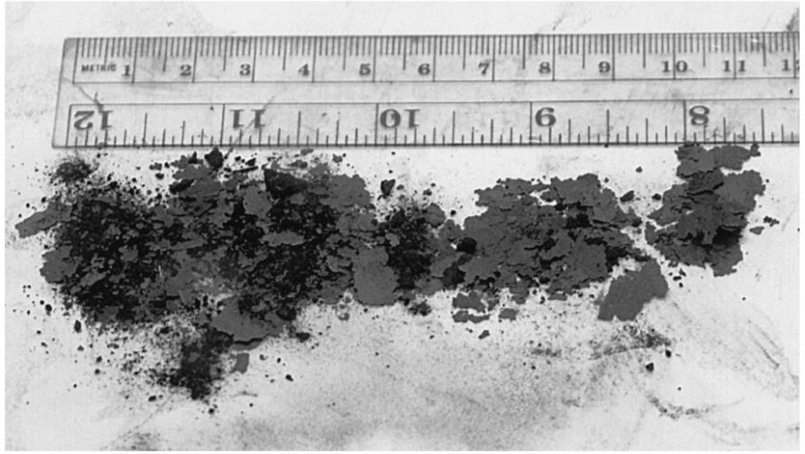
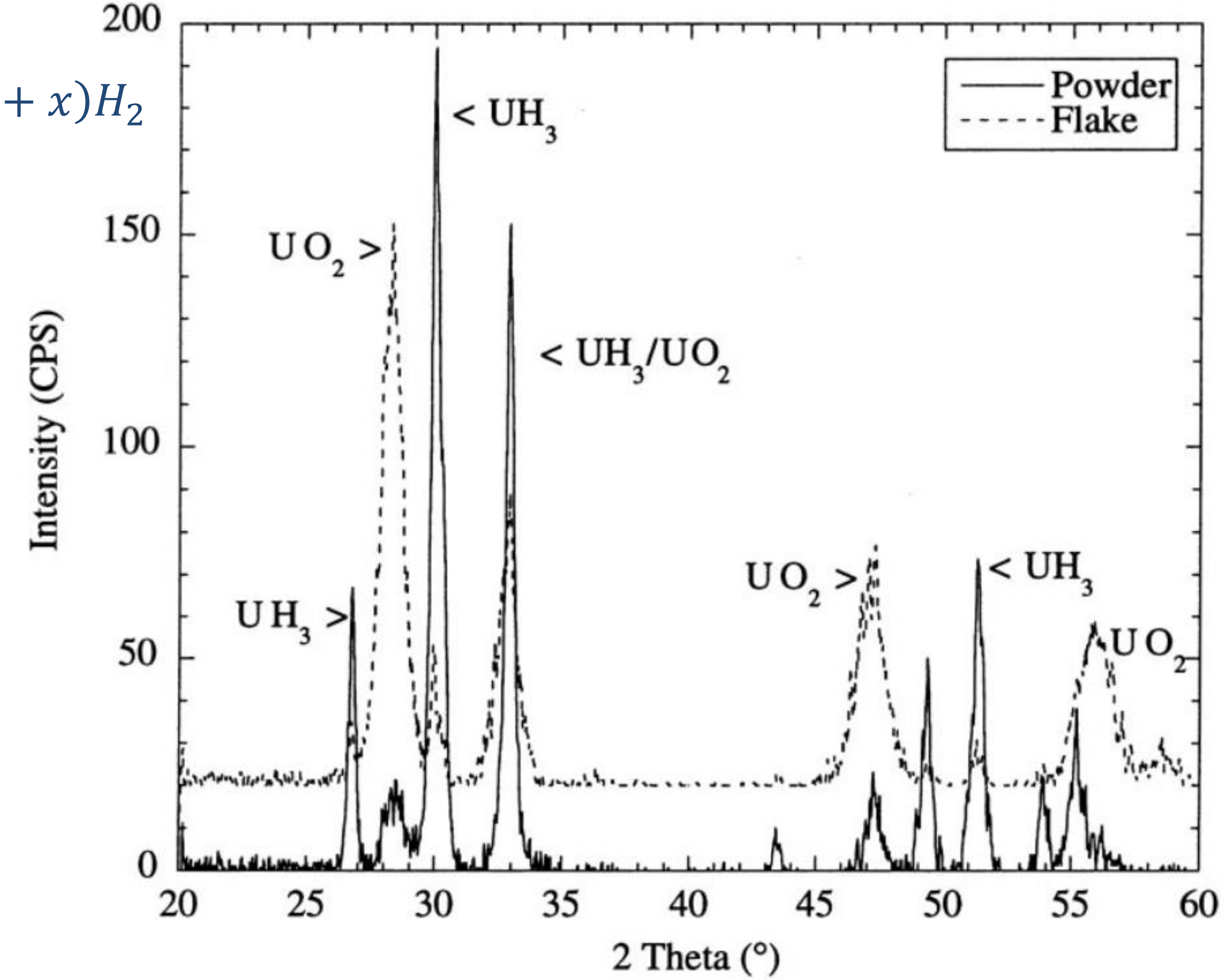


Fig. 6. Loose flake and powder corrosion product from plate shown in Fig. 2.



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https://doi.org/10.1016/S0022-3115(98)00448-6

Insights into the uranium-H_2O corrosion mechanism

$$U + (2 + x)H_2O \rightarrow UO_{2+x} + (2 + x)H_2$$

$$2U + 3H_2 \rightarrow 2UH_3$$

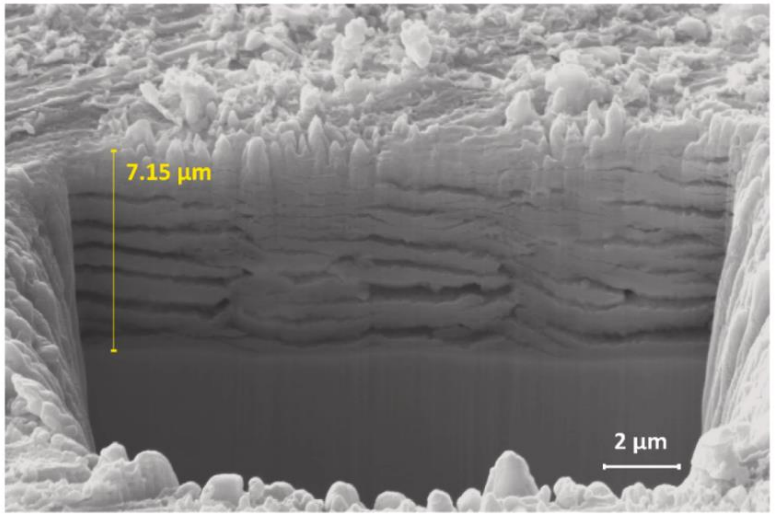
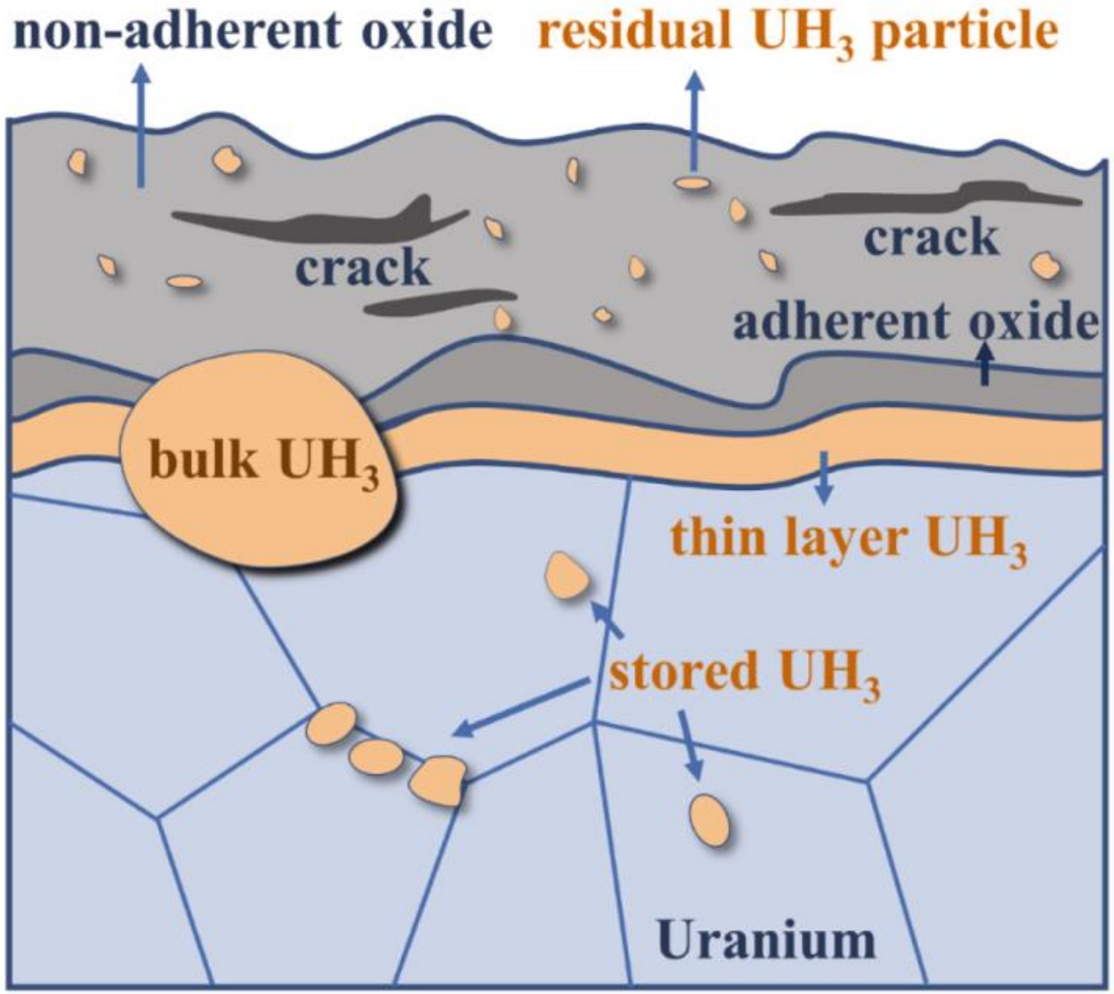


Fig. 2. FIB milling image of cross-sectional view for the corroded uranium by D_2O. The sample was tilted at an angle of 52°.

https://doi.org/10.1016/j.corsci.2023.111524



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It is hard to see the UH_3 phase

XRD can not detect UH3 due to low conc.

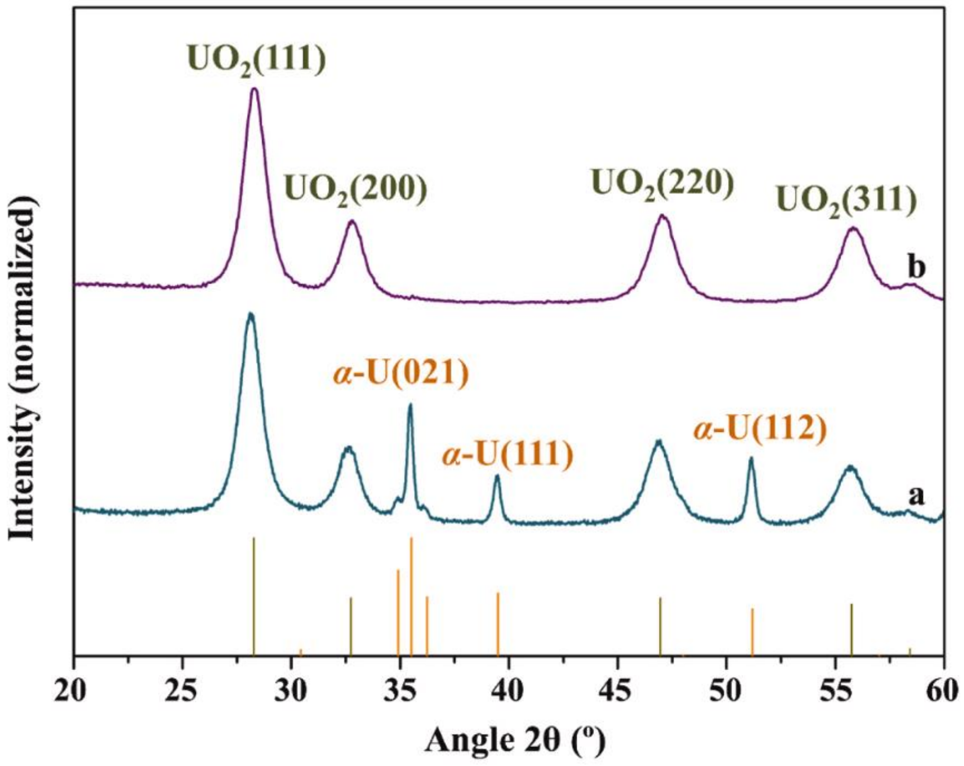
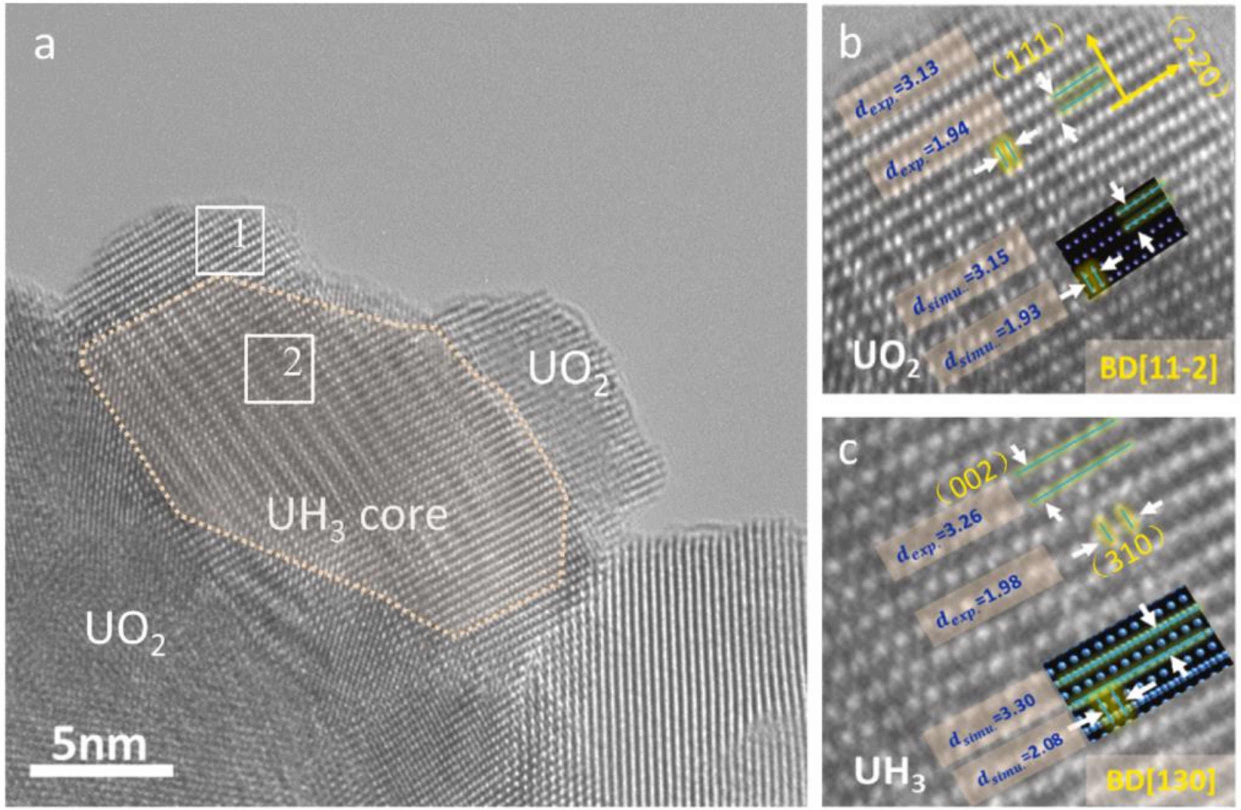


Fig. 3. (a) Conventional and (b) grazing-incidence (ω = 2°) XRD patterns of the corrosion layer on uranium surface.

TEM is revealing the microstructure in high resolution



https://doi.org/10.1016/j.corsci.2023.111524

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Different Uranium phase have different hydride formation energy

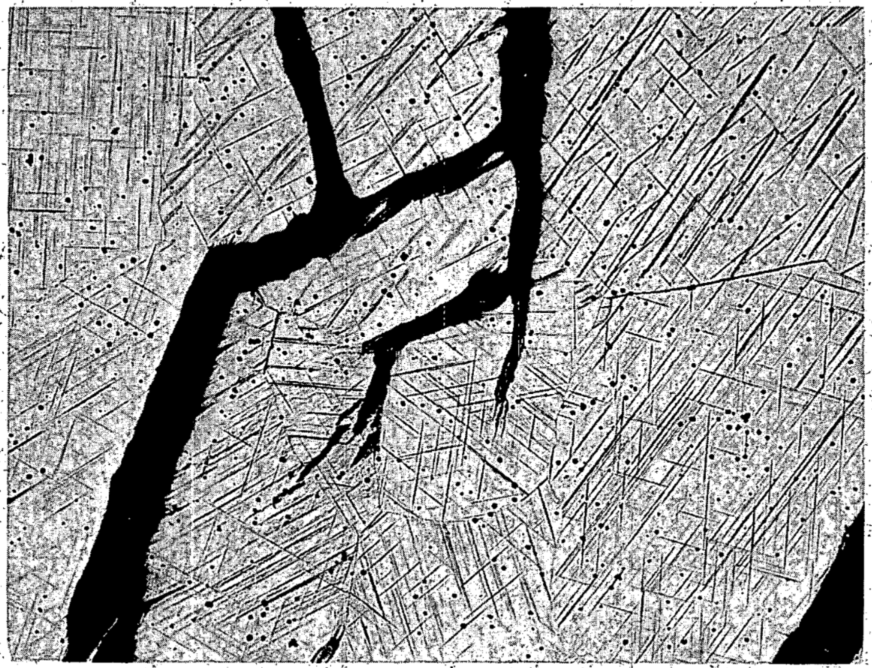
$$\Delta F_{\alpha U \rightarrow UH_3} = -4110\ cal/mol$$

$$\Delta F_{\gamma U \rightarrow UH_3} = -8132\ cal/mol$$

$$\Delta F_{12\ wt\%\ Mo-\gamma U \rightarrow UH_3} = -8406\ cal/mol$$

- While there is no study for U-10Zr interaction with water, the Zr could act as gutter for the hydrogen

- The formation of ZrH1.66 is well studied for Zircoloy claddings, how the formation of ZrH1.66 in U matrix, either in alpha or gamma, is an open question.



Neg 17420-10 100X

Fig. 4. Gamma-phase U-9.8 w/o Mo alloy after 28 days exposure to 650°F water

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U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Storage and Transportation of TRISO and Metal Spent Nuclear Fuels

Fission Product Induced Metal Fuel Swelling

Walter Williams, Ph.D.

Office of Nuclear Reactor Regulation
Division of Advanced Reactors and Non-power Production and Utilization Facilities
Advanced Reactor Technical Branch 2

Outline

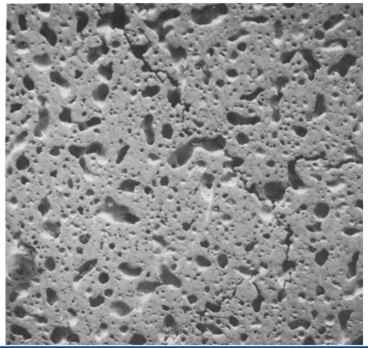
- *Operating experience with metallic U-Zr based fuels*
- *Phenomenon present in U-Zr system*
- *Fuel swelling*
  - *Solid, gaseous, and thermal expansion*
  - *Swelling correlations*
- *Potential data gaps*

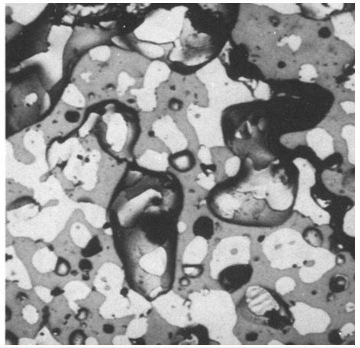
Operating experience with metallic U-Zr based fuels

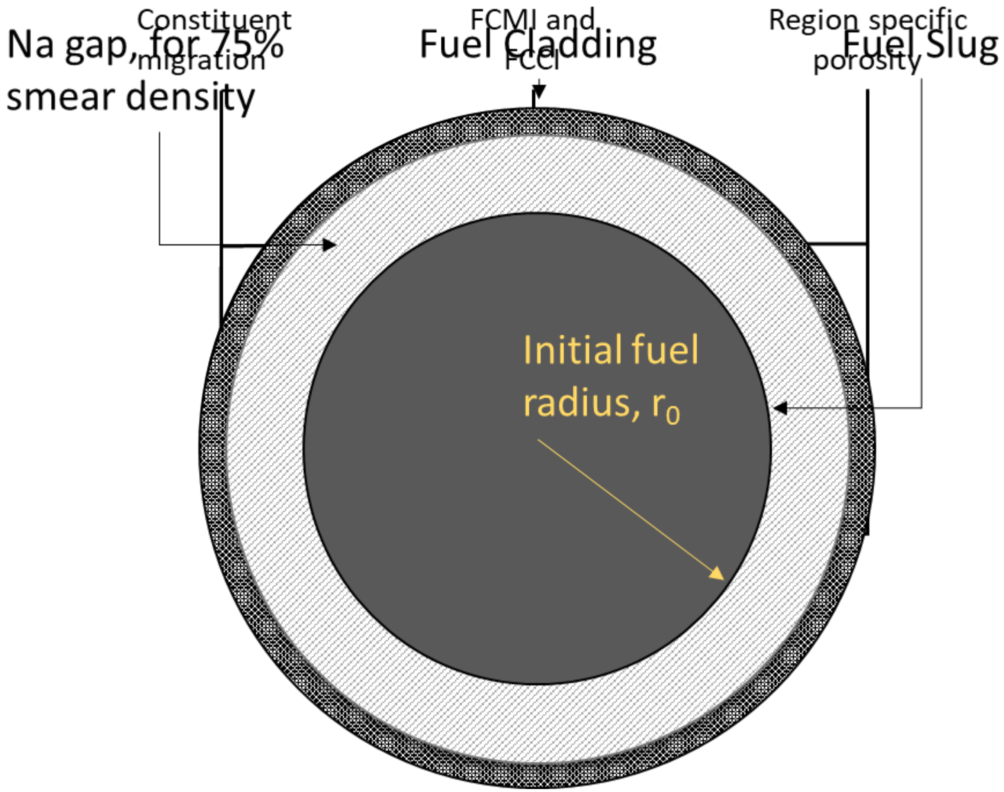
- *EBR-1 Mk 1, 2, 3, 5*
  - *1951 -- U, and U-2 (wt.%) Zr*
  - *Asymmetric swelling, embrittlement*
  - *Sodium compatibility, increased thermal conductivity, ease of fabrication and reprocessing*
- *EBR-II Mark I, II, III, IV, V*
  - *1964 -- U-Fs, U-10Zr, U-Pu-Zr*
  - *Smear density established to accommodate swelling*
  - *Plenum increase to accommodate gasses*
  - *Cladding thickness increased to constrain radial swelling*
  - *BU increase to 10 at%*
- *FFTF*
  - *1972 -- U-10Zr, U-Pu-Zr*
  - *Various cladding have been explored (316 SS, HT9)*
  - *Low radial swelling observed*
  - *15-20 at.% BU*

Phenomenon present in U-Zr system

- *Phenomenon:*
  - *Fuel swelling, constituent redistribution, fuel cladding chemical interaction, fuel cladding mechanical interaction*
- *Convolution of evolving driving forces (temperature, composition, power, burnup) hinders the derivation of mechanistic models outside of operating envelope*







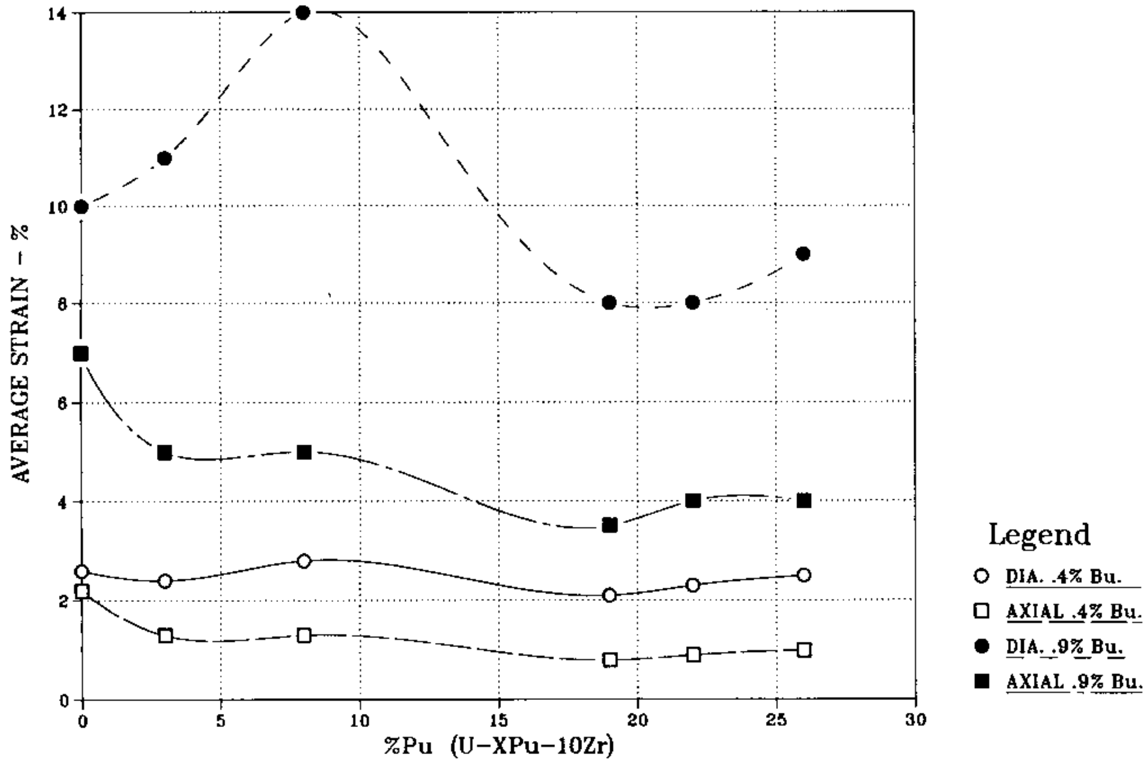
Artistic rendering of phenomena manifestation in the U-Zr system.

Williams, Dissertation, Purdue University

Hofman, G. L., Pahl, R. G., Lahm, C. E., & Porter, D. L. (1990). Swelling behavior of U-Pu-Zr fuel. *Metallurgical Transactions A*, *21*, 517-528.

Fuel Swelling

- *Asymmetric fuel swelling, primarily due to α-U*
  - *Implies thermal and composition dependency on swelling behavior*
- *Decrease in swelling with increase in Pu due to phase transition temperatures and subsequent properties*
  - *Implies operational temperature and composition dependency*
- *Minimal refined data on geometric dependency*
  - *Potential to suggest scalability of U-Zr diameters*



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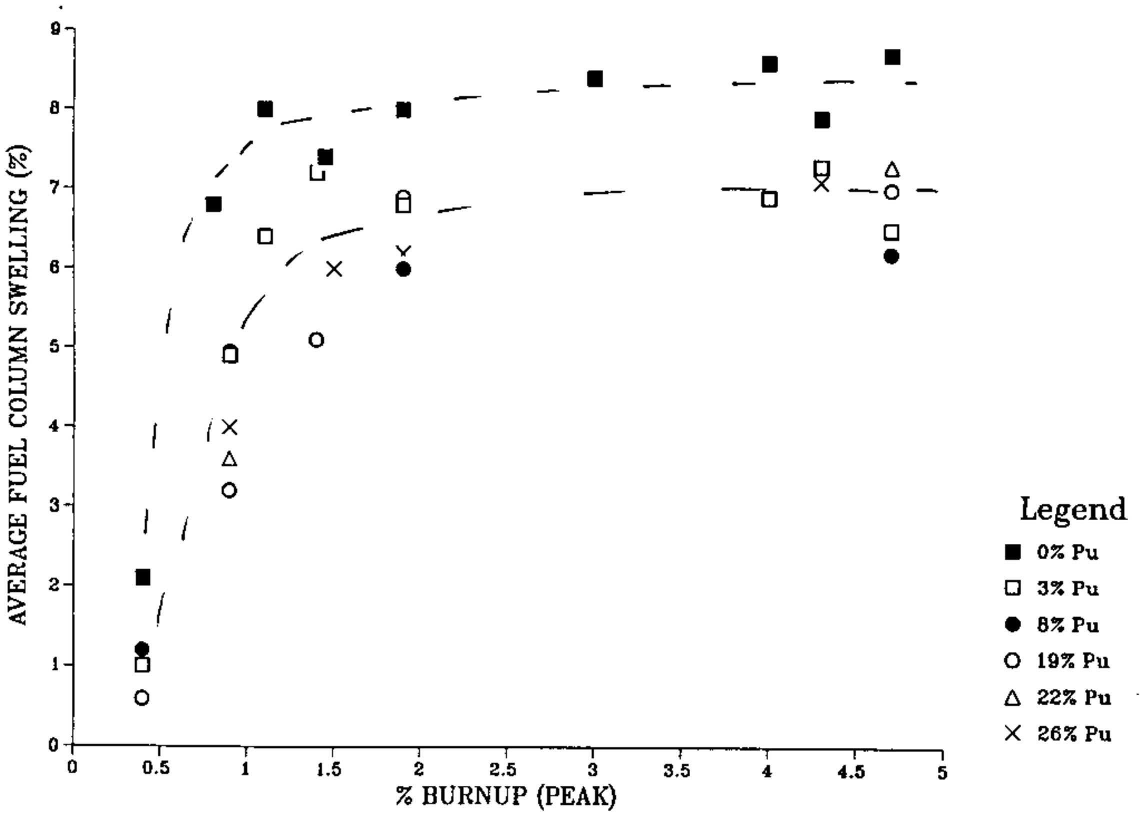

Fig. 1—Axial fuel swelling of 0.290-in. elements.

Hofman, G. L., Pahl, R. G., Lahm, C. E., & Porter, D. L. (1990). Swelling behavior of U-Pu-Zr fuel. *Metallurgical Transactions A*, *21*, 517-528.

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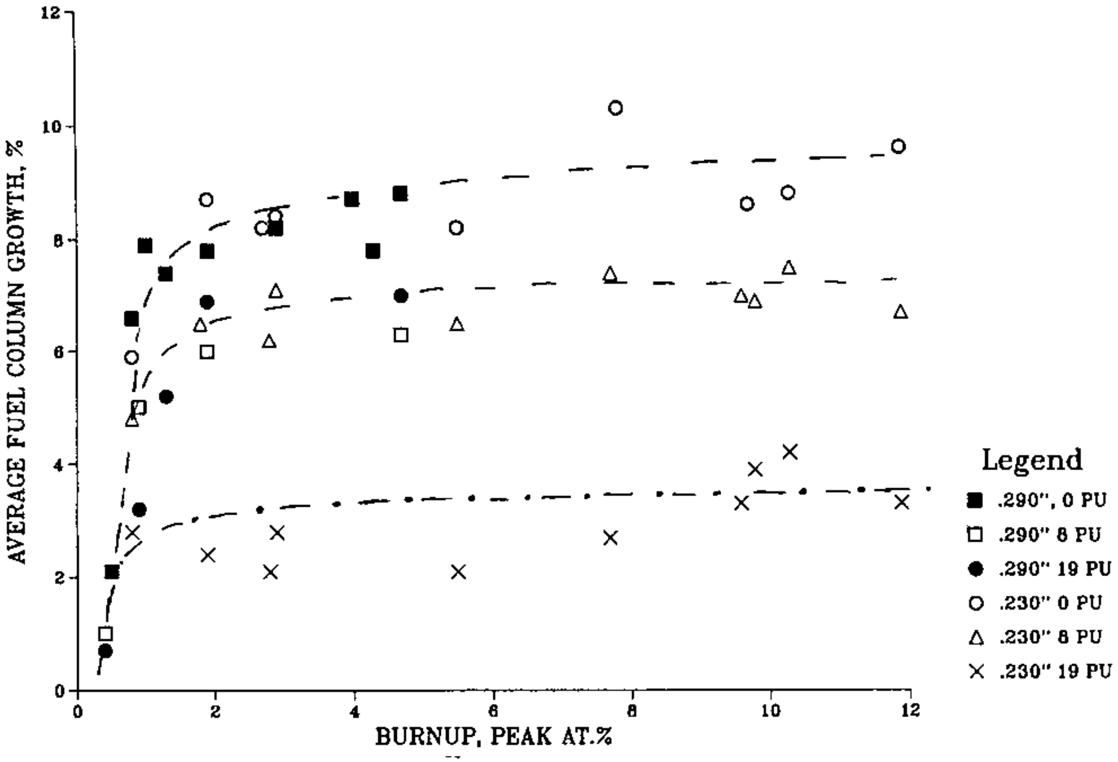
- *Minimal refined data on geometric dependency*
  - *Potential to suggest scalability of U-Zr diameters*

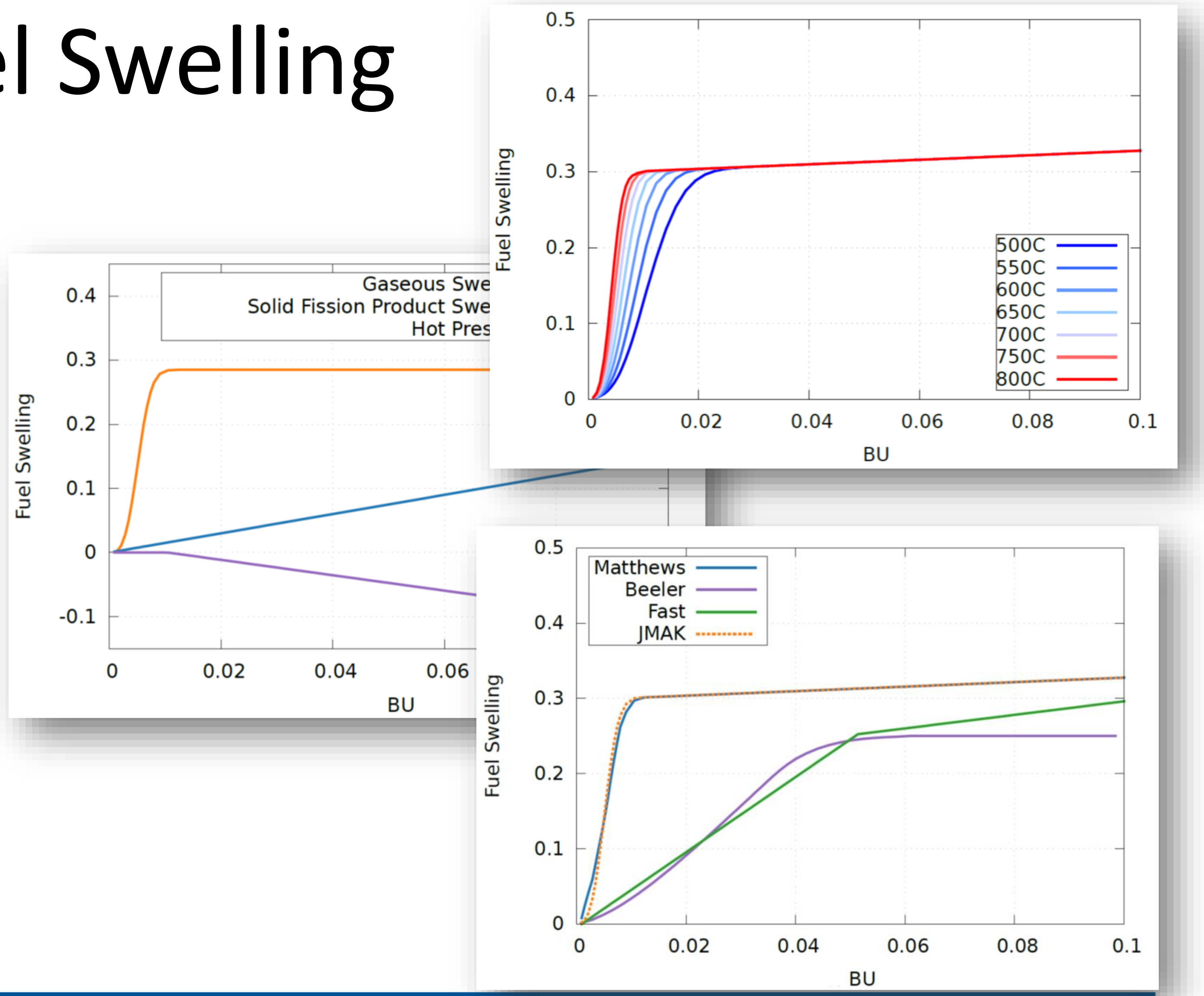


Hofman, G. L., Pahl, R. G., Lahm, C. E., & Porter, D. L. (1990). Swelling behavior of U-Pu-Zr fuel. *Metallurgical Transactions A*, *21*, 517-528.

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Fuel Swelling

- *Fission product accumulate within the fuel*
- *Swelling occurs as pores form*
- *Cladding contact occurs*
- *Interconnected porosity creates a pathway to plenum, slowing swelling*
- *Hot pressing leads to extrusion into plenum or radial strain*



Williams, TBD, A Concurrent Nucleation and Growth Model of Porosity and Subsequent Fuel Swelling in the U-Zr System

Data Gaps

- Largely empirical models
  - Current semi-mechanistic models require some level of fitting
  - Prevents deviation from historical operating envelope with certainty
- Initial geometry
  - Difficult to ascertain and defend geometric changes without concentrated effort and potential fuel testing
- Increasing burnup may allow for hot pressing to collapse pore network and increase swelling beyond 20at.% BU
- Fabrication methods may influence early-stage asymmetric swelling

Conclusion

- Swelling of historic metallic fuel systems is largely understood from an empirical standpoint.
- Mechanistic prediction of fuel swelling is an ongoing effort
- Not considered to be a life-limiting or safety-limiting phenomena when properly accounted for in manufacturing decisions such as smear density.
- Does not tend to evolve post-irradiation

Storage and Transportation of TRISO and Metal Spent Nuclear Fuels

Assessment on Metal Spent Nuclear Fuel Swelling Effects on Structural Integrity

Walter Williams, Ph.D.

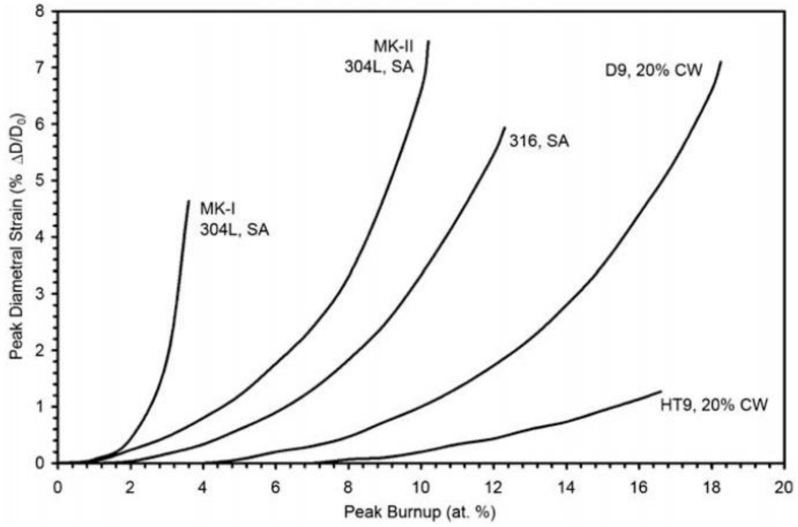
Office of Nuclear Reactor Regulation
Division of Advanced Reactors and Non-power Production and Utilization Facilities
Advanced Reactor Technical Branch 2

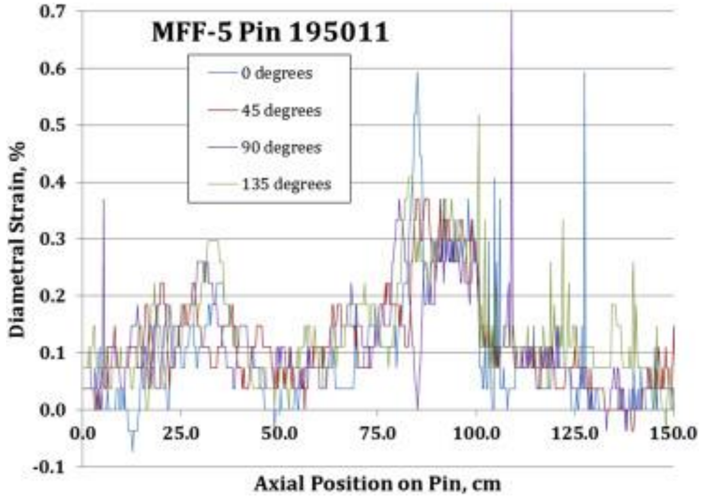
Outline

- *Fuel-Cladding mechanical interaction*
- *Fuel-cladding chemical interaction*
- *Storage experience*

Fuel Cladding Mechanical Interaction (FCMI)

- *FCMI was one of the first historic problems to be addressed in the U-Zr system.*
  - *Largely resolved with inclusion and tailoring of a smear density (as fabricated void space within the cladding).*
  - *HT9-clad U-10Zr fuel pins irradiated with a ≤75% smear density have never breached due to cladding strain between 370 and 615◦C below 17 at.% BU.*
- *Ceases after irradiation and fuel retracts slightly due to thermal expansion*
- *FCMI primarily affects in-reactor behavior but can leave residual stresses or microstructural changes in the cladding post-irradiation.*

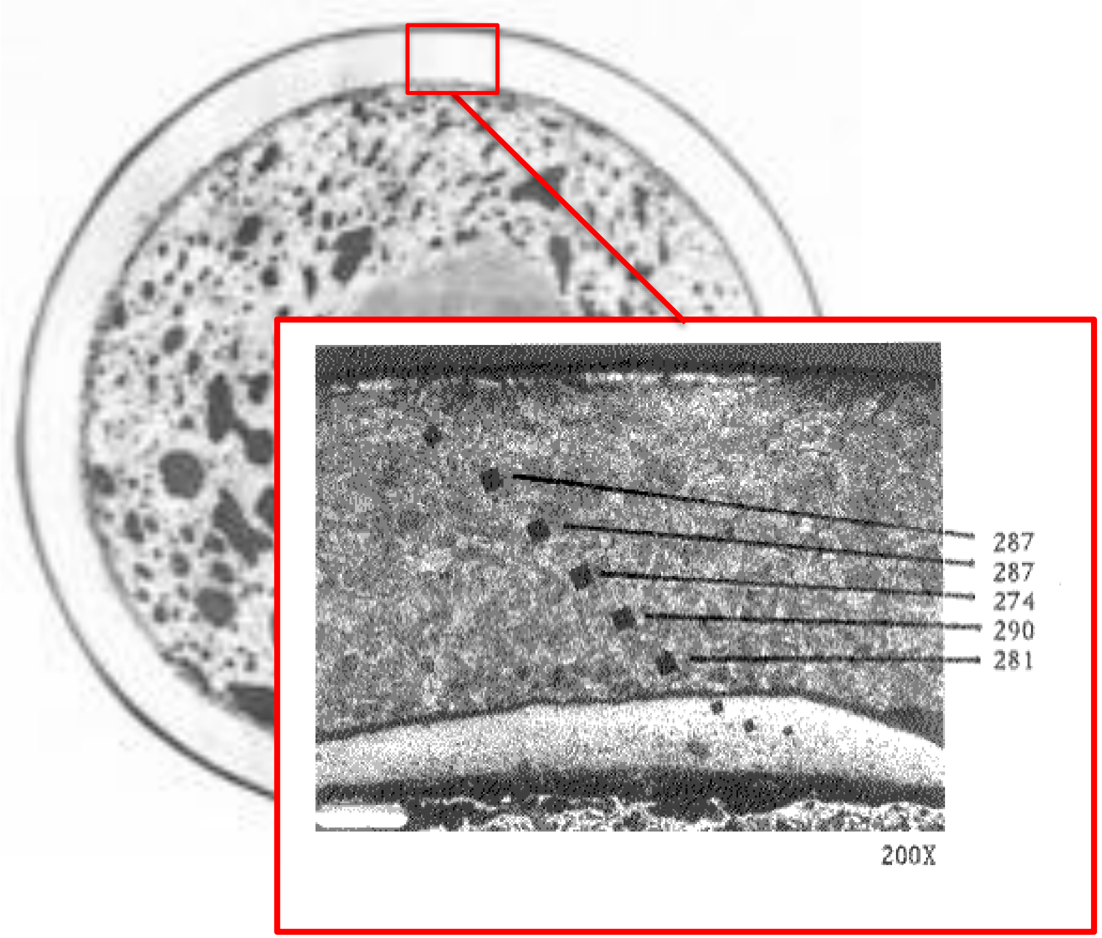




Burkes et al., "A US Perspective on Fast Reactor Fuel Fabrication Technology and Experience Part I."
NUREG/CR-7305 "Metal Fuel Qualification - Fuel Assessment Using NRC NUREG-2246, 'Fuel Qualification for Advanced Reactors'"
Carmack, W. J., et al. (2016). Metallography and FCCI in fast flux test facility irradiated metallic U-10Zr MFF-3 and MFF-5 fuel pins. JNM, 473, 167-177.

Fuel Cladding Chemical Interaction (FCCI)

- Lanthanide fission products react with cladding
- Formation of brittle interaction layer and cladding wastage
- Fuel-cladding interaction lowers solidus temperature
- While FCCI effectively ceases after irradiation, the damage can exacerbate cladding brittleness and complicate post-irradiation handling.



Steven L. Hayes, "METALLIC FUELS FOR FAST REACTORS," GENIV international Forum, 2017

Storage Experience

- Five fast reactors [EBR-I, EBR-II, FFTF, Fermi-1, and Dounreay] have had fuel stored and pulled for post-irradiation testing
- More than 20 metric tons of heavy metal (MTHM) untreated fuel remained in storage in 2007.
- Both wet and dry storage has been observed. Recovered fuel for post-irradiation testing shows no marked change in the fuel system under proper storage conditions.
- Potential issues regarding sodium
  - Moisture/oxygen ingress may cause a sodium reaction that pressurizes the storage container.
  - Interaction on cladding may exacerbate cladding wastage
  - Removal of sodium form fuel pin is the major step of reprocessing
- Long-Term Creep and Stress Relaxation
  - Not suspected to be an issue, but truly long-term storage has not been observed

Hypothetical Issues

- Cladding wastage continues at elevated temperature and results in a rupture
  - Should be detectable and predicted with standard post-irradiation examinations
  - Should not occur at lower temperature
- Spikes in storage temperature result in thermal expansion
  - Unlikely to naturally reach thermal conditions beyond in-pile conditions
- Oxygen/moisture intrusion into storage container reacts and degrades cladding further and pressurizes with H.
  - Limited stress is applied to cladding via the fuel, however the plenum may be at higher pressure
  - Can condition fuel elements to neutralize sodium to drastically reduce this
- Over time, residual stress relaxation due to creep could weaken the cladding or lead to permanent deformation.
  - Unexpected, but unknown

Conclusion

- FCCI and FCMI conclude post-irradiation, but their effects on cladding integrity may have implications for storage
- Temperature should be maintained sufficiently low to prevent continued FCCI and FCMI
- Ingress of moisture likely the primary concern
- Data is lacking on long term storage of metallic fuel as it is primarily meant for reprocessing.

PNNL-SA-205973

Potential Treatment Options for Sodium-Bonded Metal Spent Nuclear Fuel (SNF)

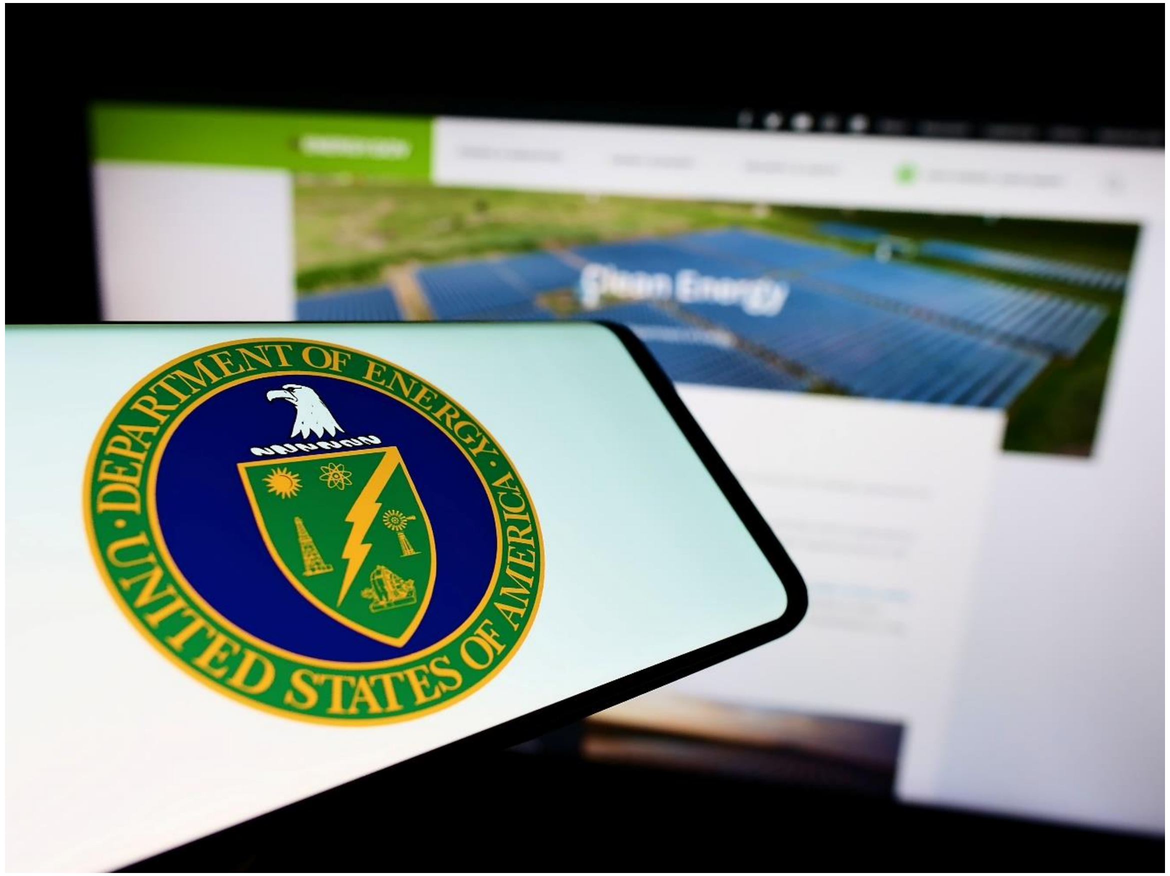
Stuart Arm
Senior Technical Advisor,
Radiochemical Flowsheets
Pacific Northwest National Laboratory

DISCLAIMER

This is a technical presentation that does not take into account contractual limitations or obligations under the Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste (Standard Contract) (10 CFR Part 961).

To the extent discussions or recommendations in this presentation conflict with the provisions of the Standard Contract, the Standard Contract governs the obligations of the parties, and this presentation in no manner supersedes, overrides, or amends the Standard Contract.

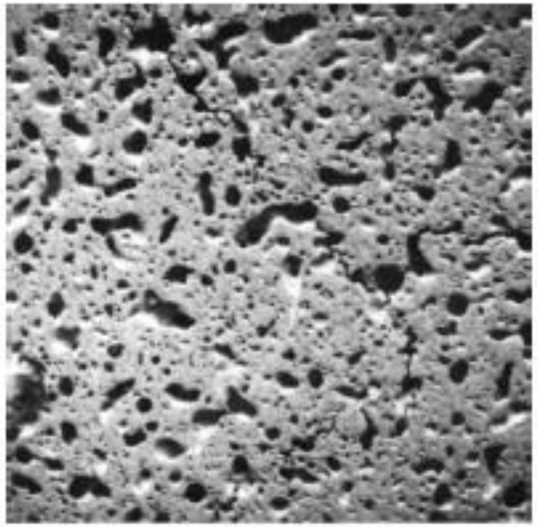
This presentation reflects technical work which could support future decision making by the U.S. Department of Energy (DOE or Department). No inferences should be drawn from this presentation regarding future actions by DOE, which are limited both by the terms of the Standard Contract and Congressional appropriations for the Department to fulfill its obligations under the Nuclear Waste Policy Act including licensing and construction of a spent nuclear fuel repository.



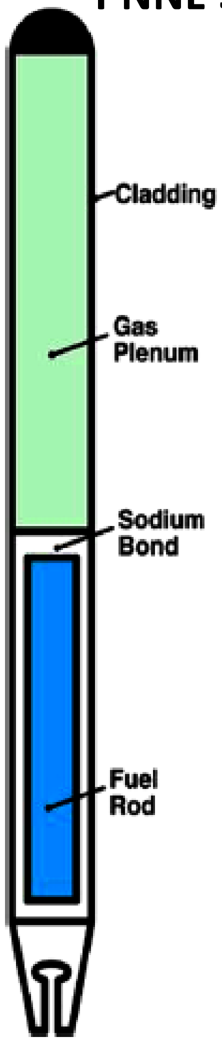
OBJECTIVE OF TREATMENT

Sodium-bonded metal spent nuclear fuel may require treatment before disposal to mitigate the hazard from reactivity of sodium metal

The sodium infiltrates the fuel as it becomes porous during irradiation



100 µm



PNNL-SA-205973

TREATMENT OPTIONS

Technology	Advantages	Disadvantages	Overall Feasibility
Sodium separation (MEDEC or alcohol wash) only	Not applicable to the impregnated sodium. However, the technology could be applied as a first step to any of the below processes. The technology has been demonstrated for separating non-impregnated sodium.		
Melt-dilute	Analogously developed for aluminum-bonded fuel, simple, flexible for waste form production.	Some process design development (off-gas system).	Feasible, some moderate uncertainties associated with adaptation for sodium-bonded, steel-clad fuel.
Electrometallurgical Treatment (electrochemical or pyrochemical)	Currently demonstrated on a small scale in US by treating the driver fuel from the Experimental Breeder Reactor-II (EBR-II).	Complexity, produces metallic and ceramic waste forms. Not demonstrated at the scale likely needed.	Feasible and demonstrated. Complex but with potential for enriched uranium recovery.

MEDEC - Melt Drain Evaporate Carbonate

TREATMENT OPTIONS

PNNL-SA-205973

Technology	Advantages	Disadvantages	Overall Feasibility
Direct conversion to glass (e.g., GMODS, GeoMelt®)	No separations	Significant RD&D needed. Not demonstrated at the scale likely needed.	Feasible but expensive with relatively significant RD&D needed.

GMODS - Glass Material Oxidation and Dissolution System
RD&D - Research, Development and Demonstration
GeoMelt® - Veolia Nuclear Solutions, https://www.nuclearsolutions.veolia.com/en/our-expertise/case-studies/geomelt-proven-technology

Information source: *Final Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel*. DOE/EIS-0306. U.S. Department of Energy Office of Nuclear Energy, Science and Technology, Washington, DC 20585.

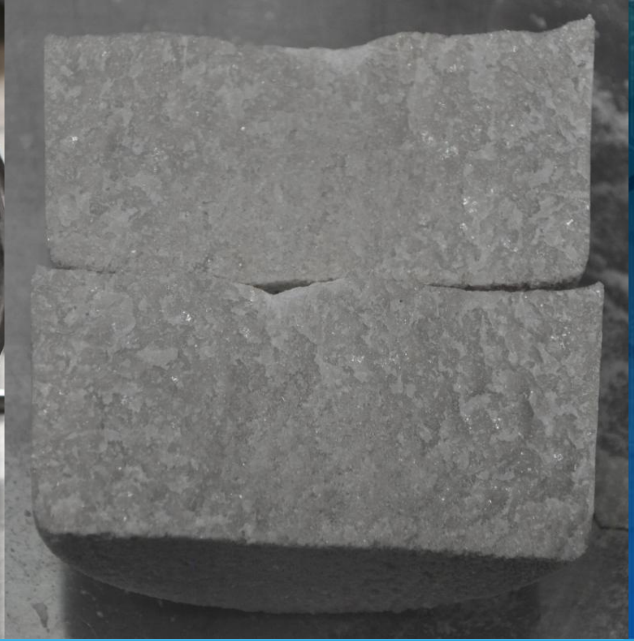
CONCLUDING REMARKS

- Sodium infiltration of the fuel complicates treatment before disposal

- Treatment options are available, but none have been demonstrated on a commercial scale

THANK YOU





December 4, 2024

Steven D. Herrmann

NRC Workshop on Storage and Transportation of TRISO and Metal Spent Nuclear Fuels

Removal and Deactivation of Bond Sodium from Fast Reactor Materials

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Motivation for Treating Sodium-Bonded Materials

- Sodium-bonded Fermi-1 blanket material along with sodium-bonded driver fuel and blanket material from Experimental Breeder Reactor-II (EBR-II) and the Fast Flux Test Facility (FFTF) – a total of 60 MTHM – have been stored at INL for nearly 50 years.
- The U.S. Department of Energy (DOE) has committed to remove sodium-bonded materials from INL by 2035 per a settlement agreement with the State of Idaho.
- The bond sodium's reactive characteristic inhibits its direct disposal in a repository.
- A DOE Record of Decision in 2000 identified electrometallurgical treatment as the preferred alternative for roughly 26 MTHM in EBR-II and FFTF materials; however, DOE opted to continue storage of the Fermi-1 blanket material while alternatives were evaluated.
- One alternative is a Melt-Drain-Evaporate (MEDE) process that has been developed at INL to remove bond sodium from blanket materials.

Bond sodium reaction in water

Motivation for Treating Sodium-Bonded Materials (cont.)

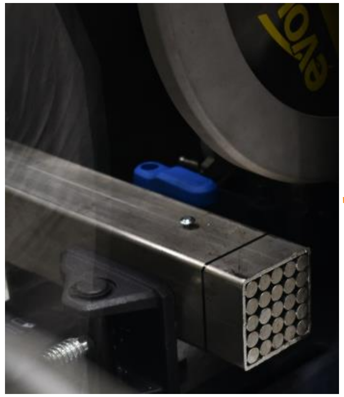
- Once the bond sodium is removed, the fuel/blanket material may be dispositioned appropriately; however, the sodium requires subsequent deactivation for its disposal.
- Traditional methods of bulk sodium deactivation (e.g., from primary and secondary coolant systems) have involved water and air/water systems; however, bond sodium constitutes substantially smaller volumes but with appreciable quantities of radioactive cesium.
- Consequently, a dry technique was developed to facilitate sodium deactivation within an inert atmosphere shielded enclosure like that needed for a MEDE process.

INL's Sodium Process Facility (Na + water systems)

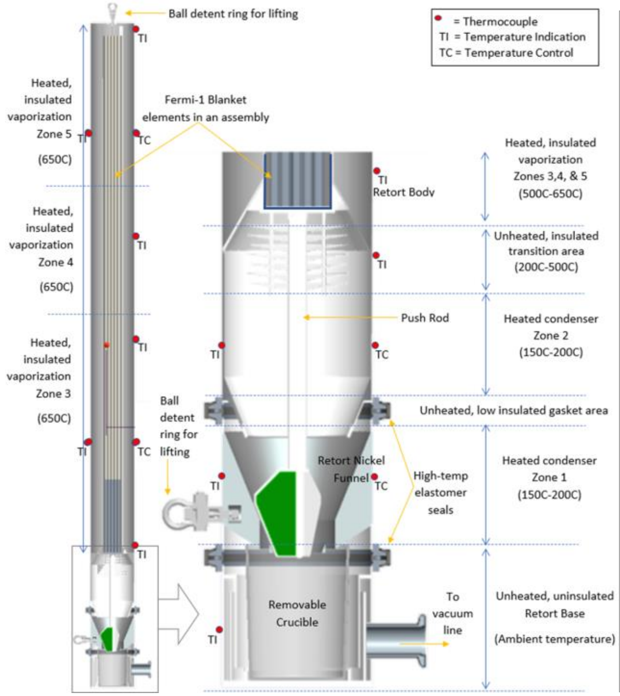
INL's Sodium Component Maintenance Shop (Na + air/water systems)

Overview of MEDE Process

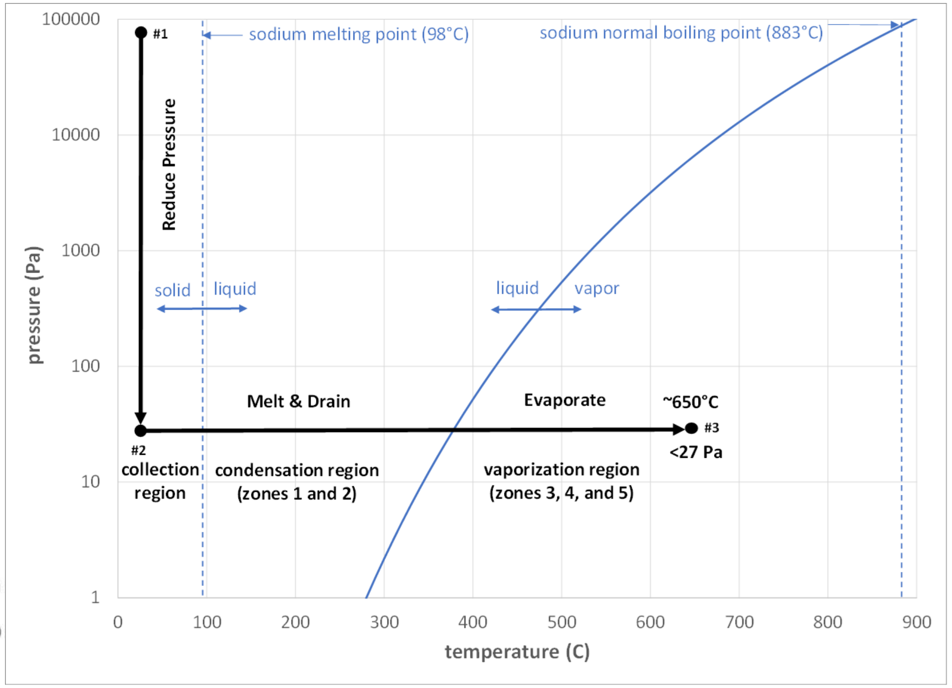
- The unit is comprised of three main parts – retort, furnace, and control system; the retort includes vaporization, condenser, and collection zones.
- A MEDE demonstration revealed ≥99.9998% removal of bond sodium from unirradiated Fermi-1 blanket elements and assembly.[1]



Breaching of Fermi-1 blanket assembly



Sectional view of MEDE retort



Sodium vaporization curve

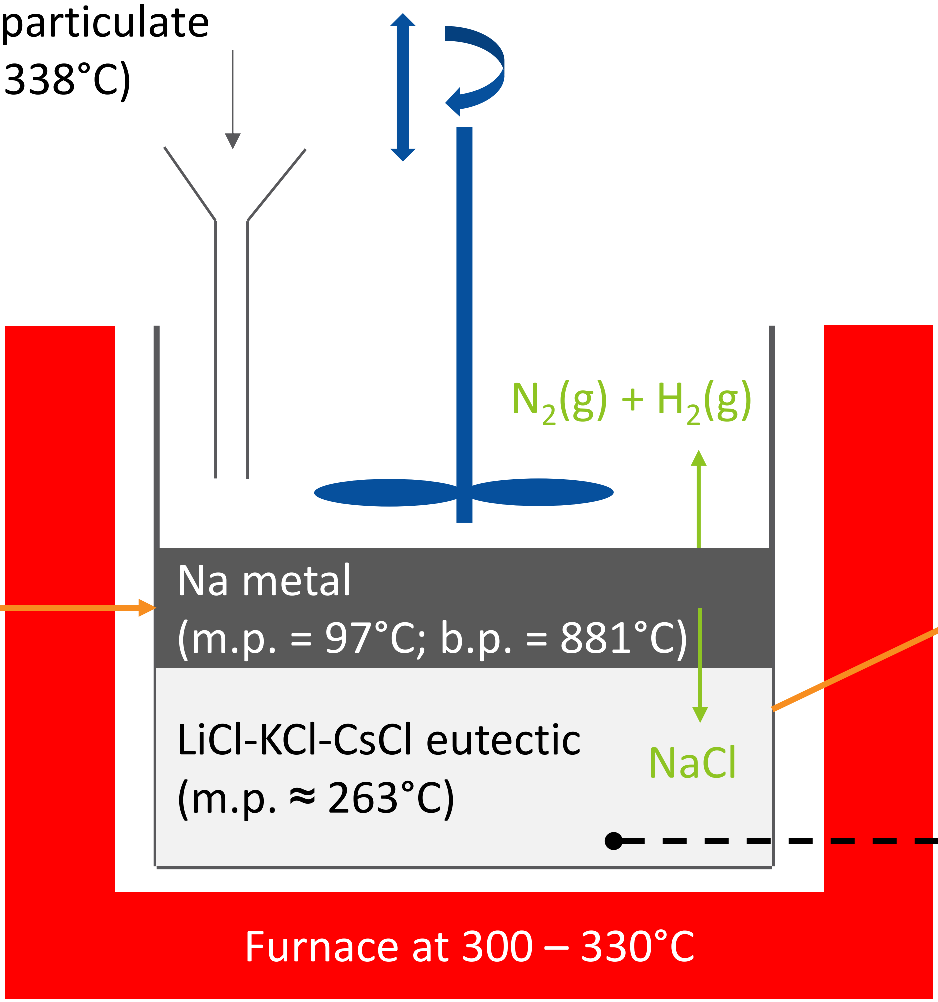
Recovered bond sodium from Fermi-1 blanket assembly

[1] S. D. Herrmann, et al., "Removal of bond sodium form Fermi-1 blanket assemblies using a melt-drain-evaporate process," *Progress in Nuclear Energy*, **163**, 104832 (2023).

Overview of Process for Dry Deactivation of Sodium Metal[2]

$2\ Na + 2\ NH_4Cl = 2\ NaCl + N_2(g) + 4\ H_2(g);\ \Delta H_{rxn,300C} = -207\ kJ$

Bond sodium from MEDE process with Fermi-1 blanket material



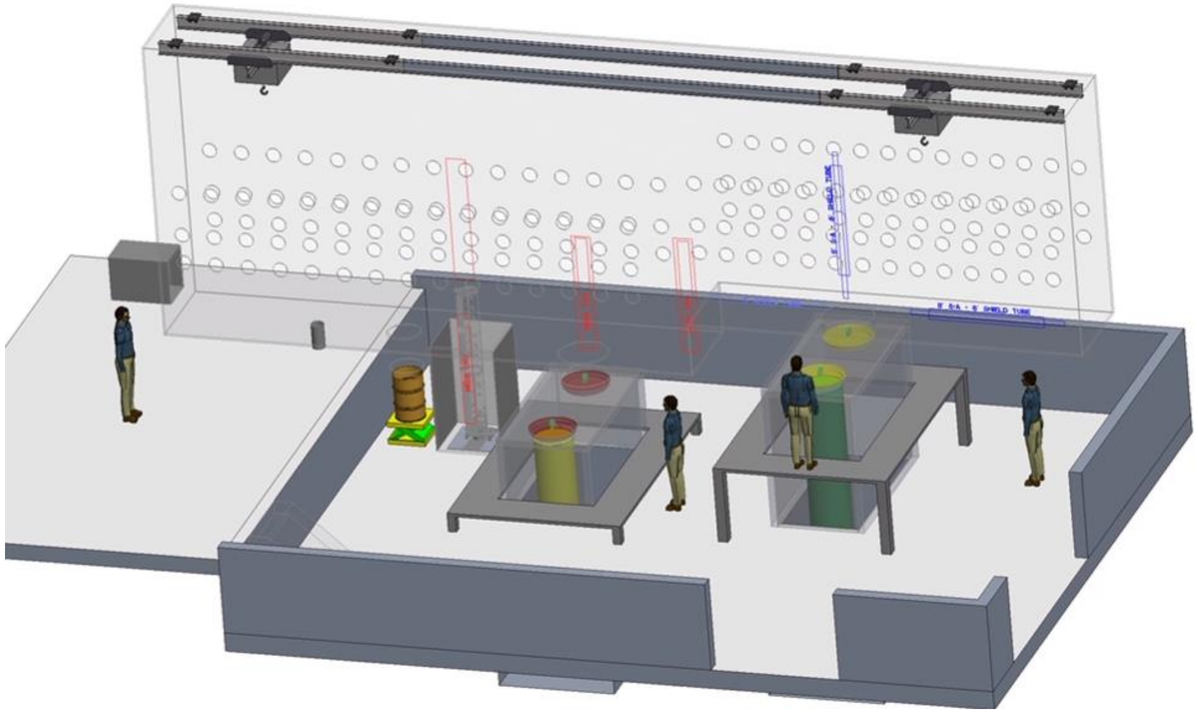
Halved salt product ingot (40 mol% NaCl)

Molten salt pool serves as an absorption medium for NaCl reaction product and a heat sink for associated heat of reaction.

[2] S. D. Herrmann, et al., "Dry Deactivation of Sodium Metal in a Molten LiCl-KCl-CsCl-NaCl System," *Journal of Hazardous, Toxic, and Radioactive Waste*, in review.

Potential Application of Integrated MEDE and Sodium Deactivation Processes

- The demonstrated sodium removal and deactivation processes substantiate a path forward for the disposition of 34 MTHM in irradiated sodium-bonded Fermi-1 blanket material.[3]



Conceptual design of a shielded glovebox for treatment of 34 MTHM in Fermi-1 blanket material and associated dry deactivation of bond sodium

- The demonstrated processes are amenable to other sodium-bonded materials in storage at INL (e.g., EBR-II blanket material) as well as future sodium-bonded materials.

[3] B. D. Preussner, et al., "Conceptual Design of Equipment and Operations for Treatment of Fermi-1 Blanket Material," *proceedings of ANS Annual Meeting, Las Vegas, NV (2024).*

Questions?

Idaho National Laboratory

Battelle Energy Alliance manages INL for the U.S. Department of Energy's Office of Nuclear Energy. INL is the nation's center for nuclear energy research and development, and also performs research in each of DOE's strategic goal areas: energy, national security, science and the environment.

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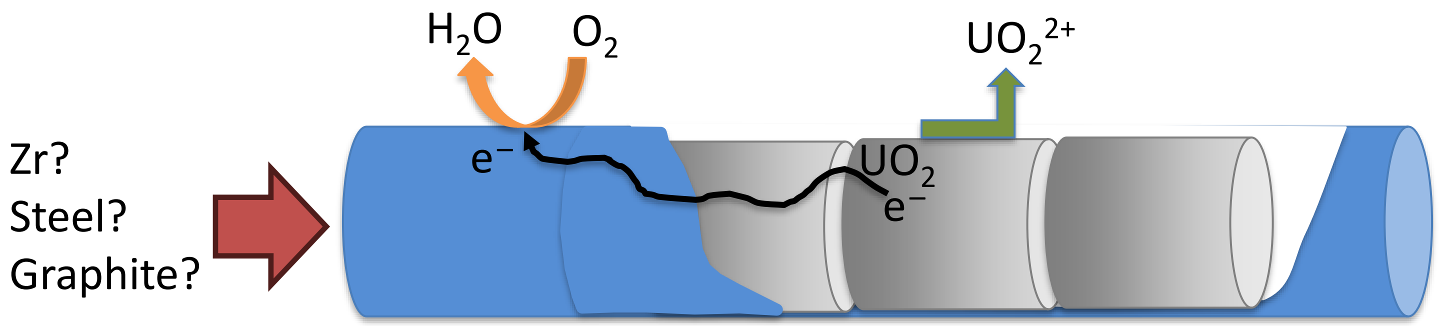
Materials interactions leading to enhanced dissolution or protection of spent fuel in long-term storage

Jamie Noël and David Shoesmith
Department of Chemistry
The University of Western Ontario
London, ON Canada

Galvanic Corrosion

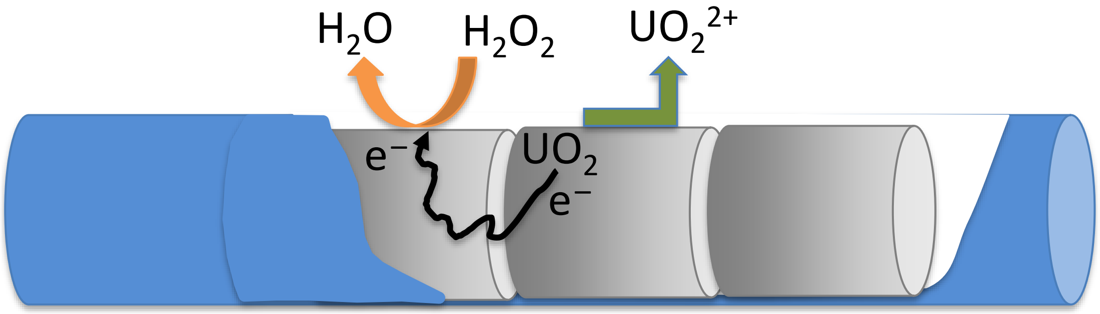
- We are bound by the law of conservation of charge
- Every electron produced by an oxidation reaction must be consumed (in real time) in a reduction reaction.
- The oxidation and reduction reaction rates must be equal.
- If the reduction reaction rate can be increased, then the oxidation reaction (spent fuel degradation) could occur at a higher rate.
- If the spent fuel is in contact with a material on which the reduction reaction can be fast or one with a large surface area, then the oxidation rate of the fuel will be enhanced.



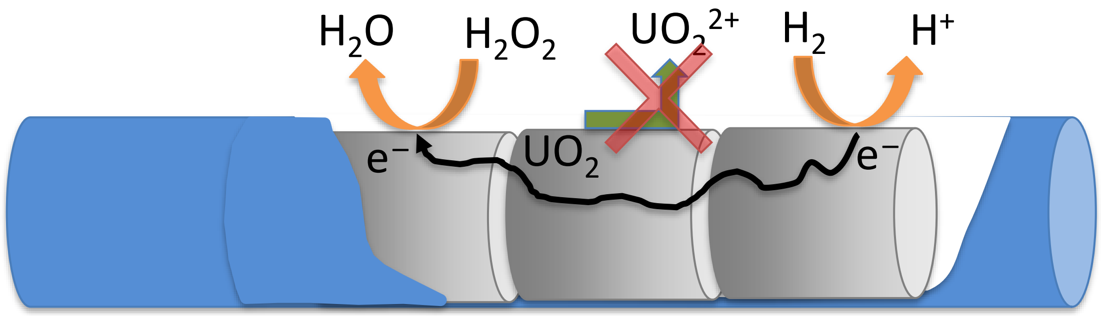
Other Coupled Redox Reactions

- Water radiolysis produces $OH^{\cdot}$, H_2O_2, $H^{\cdot}$, H_2, etc.
- Corrosion of metals may produce H_2, Fe^{2+}, etc.
- These oxidants and reductants can also react on the spent fuel surface, for better or for worse....

Peroxide enhances the corrosion of spent fuel.

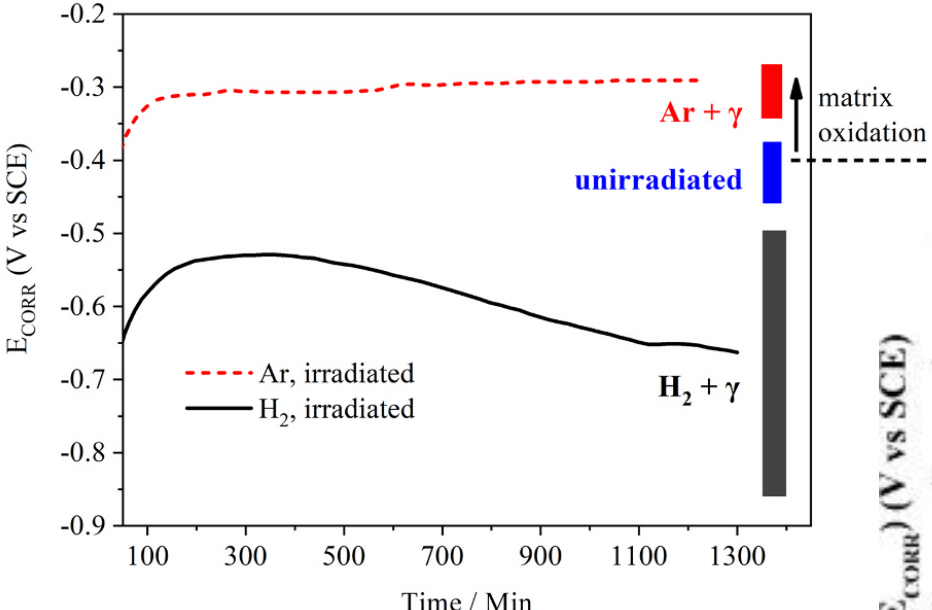


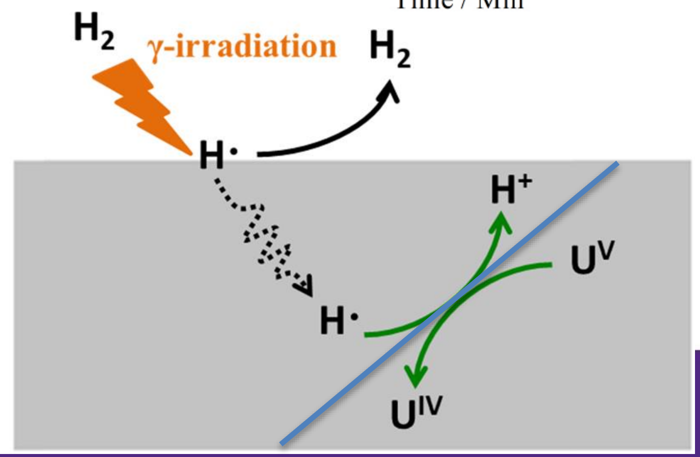
Hydrogen may protect spent fuel from corrosion by peroxide.

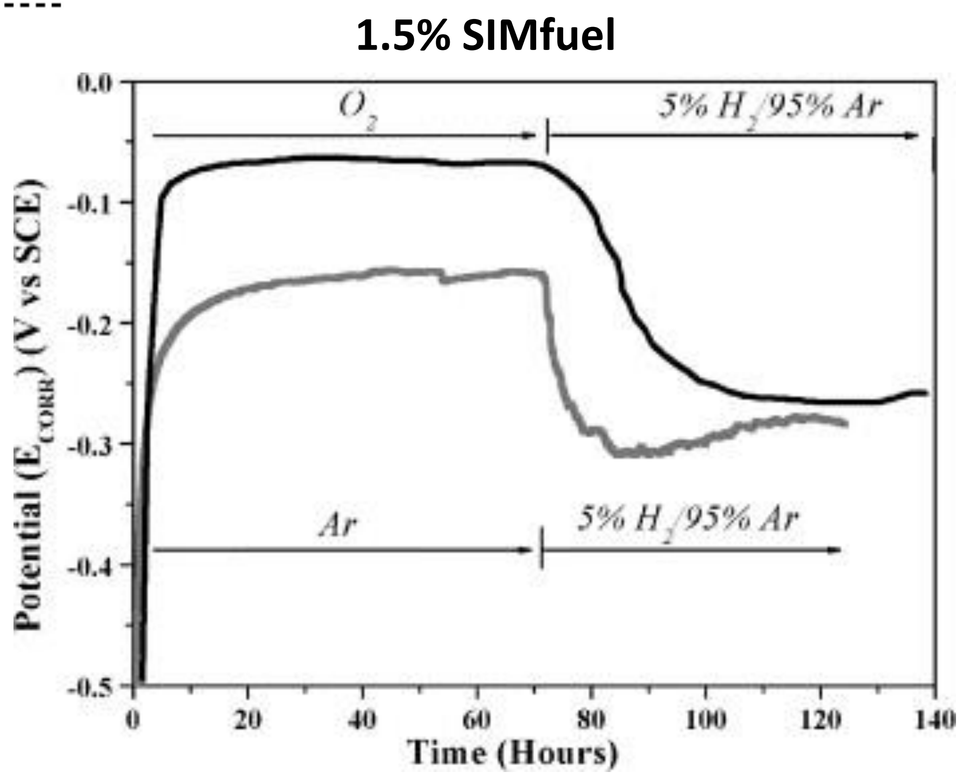


Current Research

- On CANDU fuel, SIMfuel, and UO_2 derived from metallic fuels







Published on this topic

1. Broczkowski, M. E., J. J. Noël, and D. W. Shoesmith. "The influence of dissolved hydrogen on the surface composition of doped uranium dioxide under aqueous corrosion conditions." *Journal of Electroanalytical Chemistry* 602, no. 1 (2007): 8-16.
2. Broczkowski, M. E., P. G. Keech, J. J. Noël, and D. W. Shoesmith. "The role of dissolved hydrogen on rare earth–doped uranium dioxide corrosion in the presence of hydrogen peroxide." *Journal of the Electrochemical Society* 158, no. 12 (2011): C439.
3. Badley, Martin DM, David W. Shoesmith, and James J. Noël. "Effect of Hydrogen on the Dissolution of Uranium Dioxide in Peroxide-Containing Environments." *Journal of The Electrochemical Society* 170, no. 9 (2023): 096506.
4. Liu, Nazhen, Fraser King, James J. Noël, and David W. Shoesmith. "An electrochemical and radiolytic study of the effects of H_2 on the corrosion of UO_2-based materials." *Corrosion Science* 192 (2021): 109776.
5. Liu, Nazhen, Linda Wu, Zack Qin, and David W. Shoesmith. "Roles of radiolytic and externally generated H_2 in the corrosion of fractured spent nuclear fuel." *Environmental science & technology* 50, no. 22 (2016): 12348-12355.
6. Liu, Nazhen, Zack Qin, James J. Noël, and David W. Shoesmith. "Modelling the radiolytic corrosion of α-doped UO_2 and spent nuclear fuel." *Journal of Nuclear Materials* 494 (2017): 87-94.
7. Razdan, Mayuri, and David W. Shoesmith. "The influence of hydrogen peroxide and hydrogen on the corrosion of simulated spent nuclear fuel." *Faraday Discussions* 180 (2015): 283-299.
8. Broczkowski, M. E., P. G. Keech, J. J. Noël, and D. W. Shoesmith. "Corrosion of uranium dioxide containing simulated fission products in dilute hydrogen peroxide and dissolved hydrogen." *Journal of the Electrochemical Society* 157, no. 8 (2010): C275.
9. Broczkowski, M. E., J. J. Noël, and D. W. Shoesmith. "The inhibiting effects of hydrogen on the corrosion of uranium dioxide under nuclear waste disposal conditions." *Journal of Nuclear Materials* 346, no. 1 (2005): 16-23.
10. Wu, Linda, Nazhen Liu, Zack Qin, and David W. Shoesmith. "Modeling the radiolytic corrosion of fractured nuclear fuel under permanent disposal conditions." *Journal of The Electrochemical Society* 161, no. 8 (2014): E3259.
11. Broczkowski, M. E., D. Zagidulin, and D. W. Shoesmith. "The role of dissolved hydrogen on the corrosion/dissolution of spent nuclear fuel." In *Nuclear Energy and the Environment*, pp. 349-380. American Chemical Society, 2010.
12. Broczkowski, Michael E., R. Zhu, Z. Ding, J. J. Noël, and D. W. Shoesmith. "Electrochemical, SECM, and XPS Studies of the Influence of H_2 on UO_2 Nuclear Fuel Corrosion." *MRS Online Proceedings Library* 932 (2006): 1-8.
13. Wu, Linda, Yannick Beauregard, Zack Qin, Sohrab Rohani, and David W. Shoesmith. "A model for the influence of steel corrosion products on nuclear fuel corrosion under permanent disposal conditions." *Corrosion Science* 61 (2012): 83-91.

NUCLEAR WASTE MANAGEMENT ORGANIZATION

SOCIÉTÉ DE GESTION DES DÉCHETS NUCLÉAIRES

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Future work: Galvanic coupling between TRISO kernels and graphite….

Western Science

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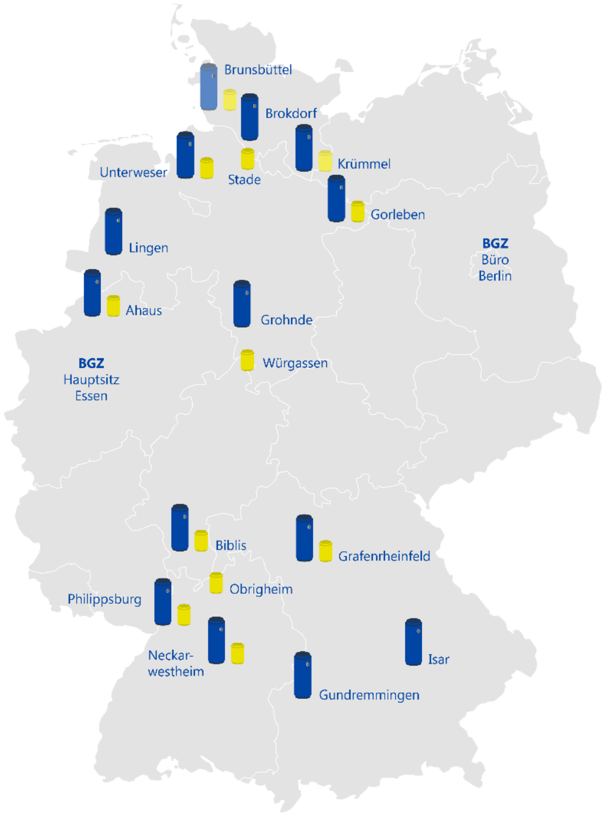
Dry Storage of THTR Spent Fuel in Germany

Dr. Maik Stuke, Ralf Schneider-Eickhoff

2024 Workshop on Storage and Transportation of TRISO and Metal Spent Nuclear Fuels

December 3rd to 5th, 2024

BGZ's task: Safe interim storage of LLW and HLW



- BGZ has been **commissioned by the German government** to provide safe and reliable interim storage of nuclear waste arising from the operation of the German NPPs.
- BGZ operates by now **13 Interim Storage Facilities for HLW**:
  - 2 centralized facilities,
  - 11 facilities at the site of NPPs.
- Due to the restart of the search for a final HLW repository, the originally approved storage period of 40 years is no longer sufficient:
  - **Storage periods of at least 100 years** must be considered.

The THTR-300 nuclear power plant

- THTR-300 NPP operated by Hochtemperatur-Kernkraftwerk GmbH (HKG)
  - helium-cooled thorium high-temperature reactor of the pebble bed type
  - thermal power of 750 MW → electrical output of 300 MW
  - intended as commercial prototype reactor

- Key data of reactor operation
  - Commissioning (self-sustaining chain reaction): September 13, 1983
  - First supply of electricity to the grid: November 16, 1985
  - First time 100% capacity reached: September 23, 1986
  - Full commercial operation: June 1, 1987
  - Start of decommissioning: September 29, 1989

- From 1985 until its closure in 1989: 16,410 operational hours → 423 full-load days

THTR operating elements

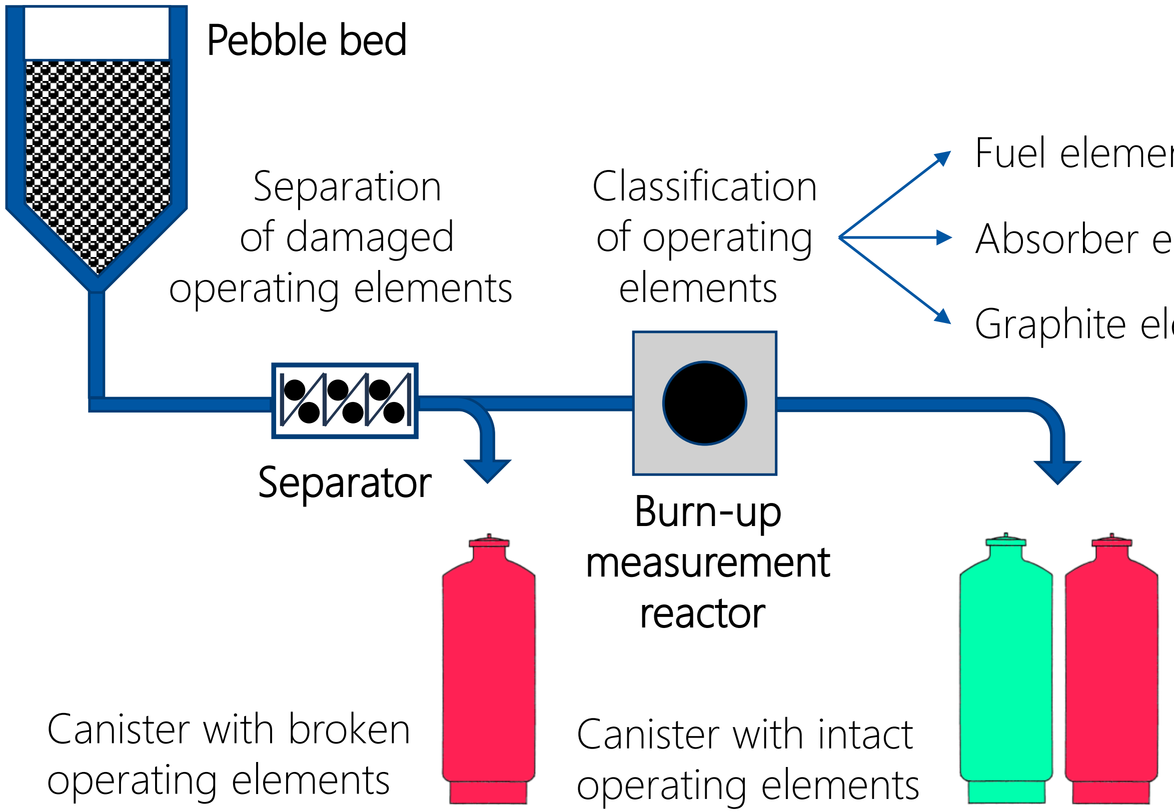
Gesellschaft für Zwischenlagerung mbH

- The THTR-300 reactor core contained approx. 670,000 operating elements, of which approx. 85% were fuel elements:
  - 563,000 fuel elements
  - 76,000 graphite elements
  - 31,000 absorber elements
- Diameter of the operating elements: 60 mm
- Unirradiated THTR fuel elements (weight ca. 200 g) contain approx.
  - 1 g of highly enriched uranium (93% U 235) and
  - 10 g of thorium
  - $(Th,U)O_2$ kernel surrounded by a HTI-BISO coating
- Absorber elements contain hafnium and boron in the graphite matrix
- Graphite elements consist of pure graphite

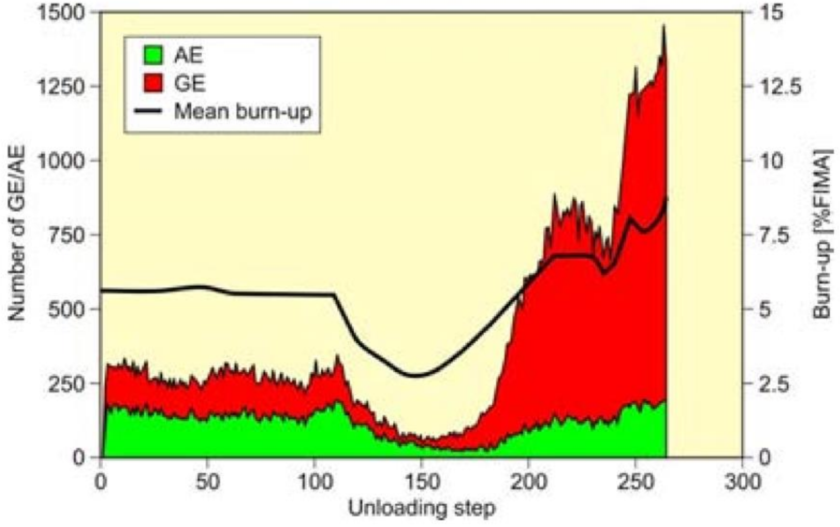
PLÄTZER, S., MIELISCH, M., Unloading of the Reactor Core and Spent Fuel Management of THTR-300 (Proc. Tech. Mtg, Jülich, 8-10 Sept, 1997), IAEATECDOC-1043, IAEA, Vienna (1998)

Management of unloading & packaging of the THTR fuel

BGZ Gesellschaft für Zwischen-lagerung mbH



Number of graphite/absorber elements and mean burn-up per unloading step with a total of 2.100 fuel elements

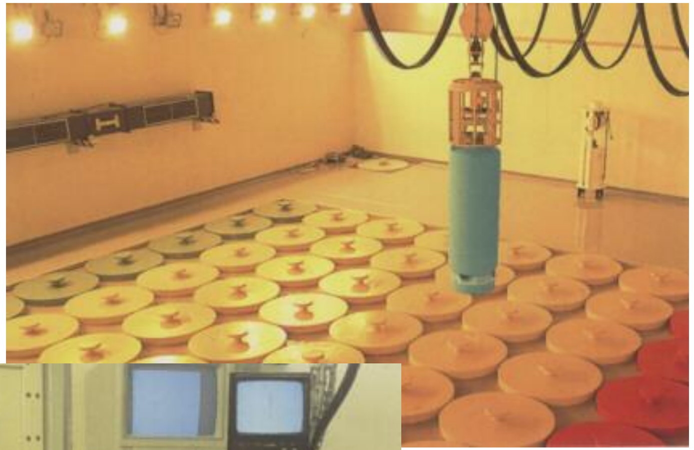


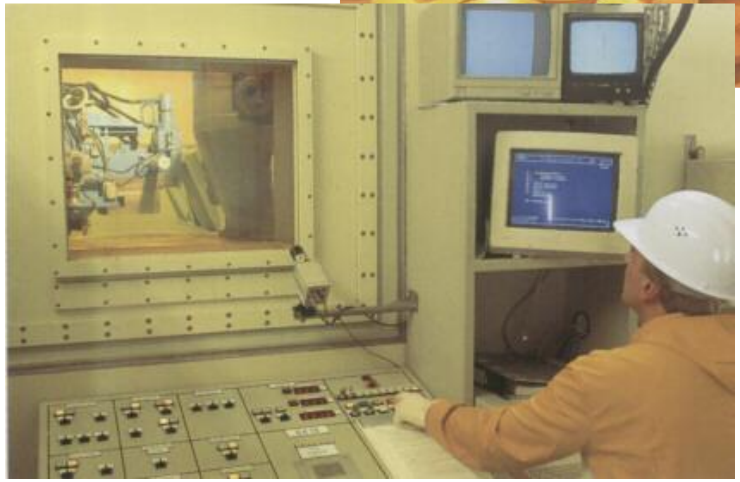
PLÄTZER, S., MIELISCH, M., Unloading of the Reactor Core and Spent Fuel Management of THTR-300 (Proc. Tech. Mtg, Jülich, 8-10 Sept, 1997), IAEATECDOC-1043, IAEA, Vienna (1998)

Management of unloading & packaging of the THTR fuel

BGZ Gesellschaft für Zwischen-lagerung mbH

- Stepwise unloading:
  - Separation of damaged fuel during normal operation
  - Unloading of the pebble bed reactor
  - Dismantling of the burn-up measuring reactor (AMR)
- Filling the steel canisters with intact only or mixed intact/defective operating elements
- Excess graphite/absorber elements are sorted out and treated as LLW
- Packaging of AMR fuel plates in 4 specific canisters (2 per cask)
- Buffer storage of the loaded canisters in cavities within the operating element storage facility
- Loading of the casks on demand:
  - Inside the shielded loading station, insertion of the canister into the cask cavity and preliminary bolting of the primary lid, both remotely operated by a manipulator
  - Outside the loading station, application of full torque to the primary lid bolts, back-filling the cask cavity with helium, fitting of the secondary lid and leak testing of both lid systems





Shipment of the casks

BGZ Gesellschaft für Zwischen-lagerung mbH

- Between June 1992 and April 1995, a total number of approx. 620,000 spent fuel elements had been transported in 305 CASTOR casks from THTR to BZA in 57 shipments, usually 6 casks on 2 railway wagons per shipment.
- Unloading operation (December 1993 - October 1994)
  - Achievement of a maximum processing rate of 11 CASTOR casks per week in 3-shift operation, 6 days a week, with optimized handling

    PLÄTZER, S., MIELISCH, M., Unloading of the Reactor Core and Spent Fuel Management of THTR-300 (Proc. Tech. Mtg, Jülich, 8-10 Sept, 1997), IAEATECDOC-1043, IAEA, Vienna (1998)

Year	No. of loaded/shipped casks
1992	14
1993	6
1994	278
1995	7

Storage of SNF at BGZ storage facility in Ahaus

BGZ Gesellschaft für Zwischen-lagerung mbH

- First licensing of Ahaus interim storage facility: April 10, 1987
- Licensing of CASTOR THTR/AVR for storage of THTR fuel elements in Ahaus: March 17, 1992
- Commissioning of the interim storage facility with the first storage of CASTOR THTR/AVR casks: June 25, 1992

- Approved / max. total heat load of the ISF: 17 MW / < 0.1 MW
- Approved / used number of storage positions: 420 / 50

- Current inventory:
  - 305 CASTOR THTR/AVR
    with THTR fuel (operating elements / burn-up measuring reactor)
  - 3 CASTOR V/19 } with LWR fuel assemblies from NPPs
  - 3 CASTOR V/52 } with LWR fuel assemblies from NPPs
  - 18 CASTOR MTR 2
    with fuel elements from the Rossendorf VVR-SM research reactor

CASTOR THTR/AVR in storage configuration

- The monolithic cask body of the CASTOR THTR/AVR is made of ductile cast iron, without additional moderator or cooling fins
  - Wall thickness (side/bottom): 370 mm

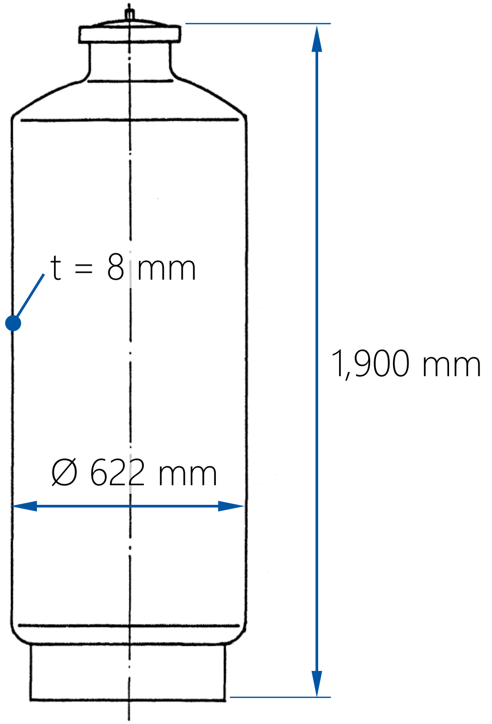
- Overall length: ca. 2.8 m
- Outer diameter: ca. 1.4 m

- Weight cask body: 23.0 t
- Gross weight loaded cask: 26.1 t

- Weight steel cannister, loaded: 770 kg
- Weight steel cannister, empty: 350 kg

- Loading: 1 steel canister with approx. 2,100 operating elements
  - Cannister is helium filled (1 bar) and sealed (< 10^{-4} Pa·m³/s)

Steel canister for intact THTR-OE



Operational experinces

- The design and approved heat load is approx. 200 watts per cask
  - Real burn-up (max. 8.8% fima per cask, mean < 6% fima per cask) significant lower than design burn-up (mean 11.4% fima per cask)
  - Actual heat load per cask approx. 100 W at the time of loading
- Due to the lower heat load, risk of falling below the dew point
  - Moisture may cause corrosion at unprotected cask components
- Countermeasures:
  - Closure of the vents to reduce the amount of moisture entering with the outside air and to increase the minimum temperature inside
  - Additional corrosion protection measures at the outer cask surfaces → Maintenance campaign in the early 2000s

- Regular 10-year inspection carried out on six casks to date
  - No unexpected effects, except for normal signs of wear and aging

Summary

- BGZ, by far the largest operator of interim storage facilities in Germany, has extensive operational experience, including with dual-purpose casks for spent THTR fuel.
- The entire high-level radioactive inventory of the THTR-300 nuclear power plant was packed into a total of 305 CASTOR THTR/AVR casks and transported to the Ahaus interim storage facility, where it has been safely stored for more than 30 years.
- The long-term safety of the inventory was assessed as part of the licensing procedure. Due to the low loads during storage compared to normal operation in the reactor core and the high safety margins in the cask design, no safety-relevant ageing effects are expected.

BGZ Gesellschaft für Zwischenlagerung mbH

Thank you for your interest !

BGZ's research program

Management and Disposal of U.S. Department of Energy's TRISO- and Metallic-based Spent Nuclear Fuel and Preliminary Considerations for Waste Resulting from Advanced Nuclear Reactors

Presented to:
U.S. Nuclear Regulatory Commission

Presented By:
Bret Leslie

December 5, 2024

2024 Workshop on Storage and Transportation of TRISO and Metal Spent Nuclear Fuels

Independent Federal Agency

The U.S. Nuclear Waste Technical Review Board (Board) was established by Congress as an independent federal agency in the 1987 amendments to the Nuclear Waste Policy Act

Board Mission

- The Nuclear Waste Policy Act
    - ➢ Established a federal responsibility framework for disposal of commercial and atomic energy defense spent nuclear fuel (SNF) and high-level radioactive waste (HLW) in a deep geologic repository and the requirements for implementation

- The Board evaluates the "technical and scientific validity" of U.S. Department of Energy (DOE) activities related to implementing the Nuclear Waste Policy Act, including
    - ➢ Activities related to the packaging of SNF and HLW and transportation of the wastes to a federal storage or disposal facility
    - ➢ Site characterization, design, and development of facilities for disposing of SNF and HLW

DOE Activities Relevant to Workshop

- Gathering information needed to address contracts for acceptance, transportation, and disposal of wastes from advanced reactors
  - [… "the Secretary [DOE] is authorized to enter into contracts with any person who generates or holds title to high-level radioactive waste, or spent nuclear fuel, of domestic origin for the acceptance of title, subsequent transportation, and disposal of such waste or spent fuel."] (Nuclear Waste Policy Act, Section 302)

- Storage, packaging, transportation, and disposal of DOE SNF, including TRISO and metal fuels
  - DOE actively manages more than 250 types of SNF (e.g., wet and dry storage and treatment of sodium-bonded metal fuel)
  - DOE developed waste system acceptance criteria for SNF and HLW
  - DOE analyzed disposal of its SNF and HLW at the proposed Yucca Mountain repository

Board Review of Relevant DOE Activities

- DOE's gathering of information needed to address contracts for acceptance, transportation, and disposal of wastes from advanced reactors
  - Summer 2023 Board meeting addressed
    - DOE's Back-End Management of Advanced Reactors (BEMAR) effort and
    - DOE's *Advanced Reactor Fuel Gap Analysis* and *Features, Events, and Processes Analyses* activities
    - Review results transmitted to DOE [Siu 2024]
- Storage, packaging, transportation, and disposal of DOE SNF, including TRISO and metal
  - Comprehensive Board report on management and disposal of DOE SNF [NWTRB 2017]

Board Findings Summer 2023 Meeting

The Nuclear Waste Policy Act, as amended, stipulates that an operating license for a reactor cannot be issued by the U.S. Nuclear Regulatory Commission unless the applicant has entered into a contract with the Secretary of Energy for disposal of SNF and HLW generated from the reactor's operation [Siu 2024].

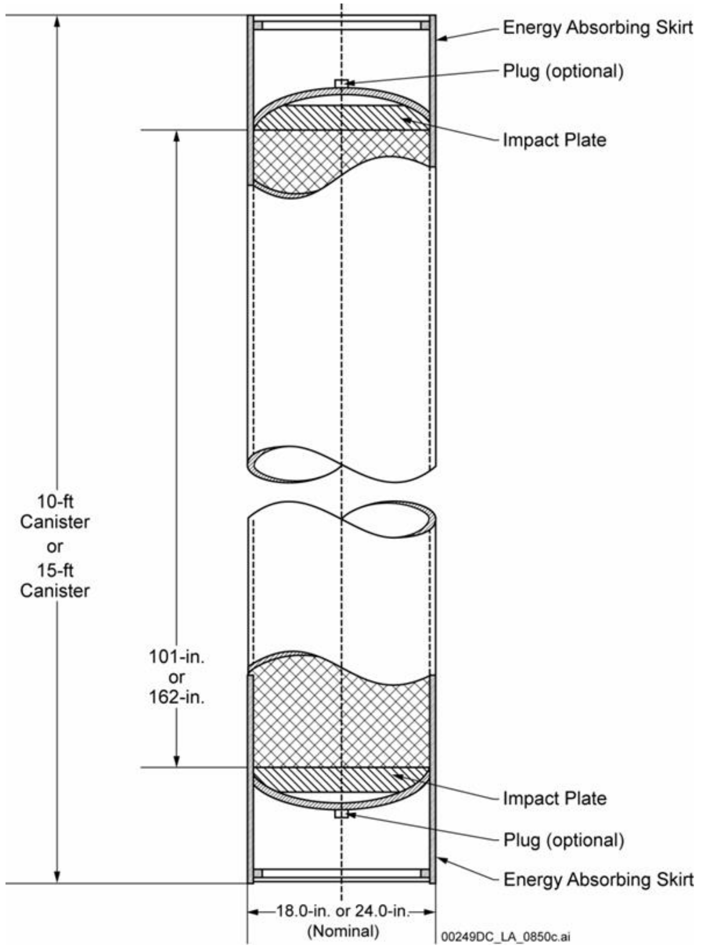
Finding: DOE has initiated an effort to assess the potential impacts of various advanced nuclear fuels on storing, transporting, and disposing of SNF and HLW by requesting data from advanced reactor vendors. DOE is also developing a strategy to identify knowledge gaps and outline areas where further research would contribute to a well-defined disposition pathway for SNF and HLW resulting from advanced reactor operations. This effort will inform DOE decisions concerning how to proceed and how to deal with the impacts. The Board commends DOE for initiating this assessment and strongly endorses the effort. [Siu 2024]

Board Review on Management of DOE SNF

- Board report [NWTRB 2017]
  - Presented the characteristics of DOE SNF and its management (storage, packaging, and transportation) and disposal
  - Described the legal and regulatory constraints on management and disposal
  - Included Board recommendations to DOE related to
    - Aging management,
    - Measuring and monitoring conditions during storage,
    - Drying procedures,
    - Packaging facilities,
    - Waste acceptance system requirements, and
    - Generic disposal research effort

Key Considerations for Disposal

- Packaging of DOE SNF in "small diameter" multiple-purpose (storage, transportation, and disposal) canisters (i.e., the DOE standardized canister)
  - ➢ Fuel and neutron absorbers, depending on characteristics of the fuel (e.g., enrichment of fissile isotopes)
  - ➢ One DOE standardized canister in "larger" co-disposal waste package (roughly comparable in size to commercial dual-purpose canisters [DOE 2009]) containing five HLW glass canisters
- Probabilistic performance assessment of repository safety, including analysis of features, events, and processes that differ from existing commercial SNF
  - ➢ Potential gas generation
  - ➢ Waste form degradation rates



DOE standardized canister [DOE 2009]

[NWTRB 2017]

Key Considerations for Disposal (continued)

- Disposal criticality [NWTRB 2017]
  - DOE's licensing approach relied on demonstrating that the total probability of criticality for all waste forms would not exceed one chance in 10,000 over the first 10,000 years after permanent closure of the repository
  - DOE defined nine groups of DOE SNF, analyzed criticality for each group using a representative SNF, and determined whether, or what, neutron absorbers (i.e., baskets or baskets and supplemental pellets) were required to be packaged with the SNF
- Sodium-bonded SNF [NWTRB 2017]
  - Inventory was not included or assessed for proposed Yucca Mountain repository
  - Waste forms from electrochemical processing of sodium-bonded SNF
    - Not included or assessed for proposed Yucca Mountain repository
    - The Board recommended DOE Office of Nuclear Energy (DOE-NE) should implement the existing waste acceptance system requirements to increase the likelihood that SNF managed by DOE-NE and that waste forms resulting from electrochemical processing of sodium-bonded SNF would be acceptable for geologic disposal in a repository

Summary

- Nuclear Waste Policy Act requirements drive the need for information to assess the disposability, not just storage and transportation, of SNF and HLW from advanced reactors prior to NRC licensing

- Disposability of TRISO and metal-based SNF has only been assessed for an unsaturated oxidizing repository (Yucca Mountain) and required the use of smaller diameter canisters in which neutron absorbers were packaged with higher-enrichment non-TRISO fuels

- Assessments of disposability for SNF and HLW from advanced reactors, including consideration of SNF packaging requirements during reactor operations and the assessment of features, events, and process, has not yet been completed for other disposal environments (e.g., reducing and saturated conditions in crystalline and clay/argillite host rocks)

References

DOE, *Yucca Mountain Repository License Application Safety Analysis Report,* DOE/RW-0573, Rev. 1. February 2009

NWTRB, *Management and Disposal of U.S. Department of Energy Spent Nuclear Fuel*, December 2017

N. Siu, April 24 letter from Nathan Siu, Chair, U.S. Nuclear Waste Technical Review Board to Dr. Kathryn Huff, Assistant Secretary for Nuclear Energy, 2024

PROJECTION OF TRISO SPENT NUCLEAR FUELS AND RELATED ISSUES

T. K. KIM
Manager of Nuclear Systems Analysis Department
Nuclear Science & Engineering Division
Argonne National Laboratory

December 2024
Workshop on Storage and Transportation of TRISO and Metal Spent Nuclear Fuels

CONTENTS

- **TRISO Fuel Characteristics**
  - Generation of TRISO SNF mass and volume
  - Projection of required Canisters for accommodating unprocessed TRISO SNF
- **Projection of TRISO SNF**
  - Nuclear capacity expansion scenario
  - Projection of TRISO SNF
- **Flooded criticality of microreactor transportation**
- **Conclusions**

GENERATION OF SPENT NUCLEAR FUELS

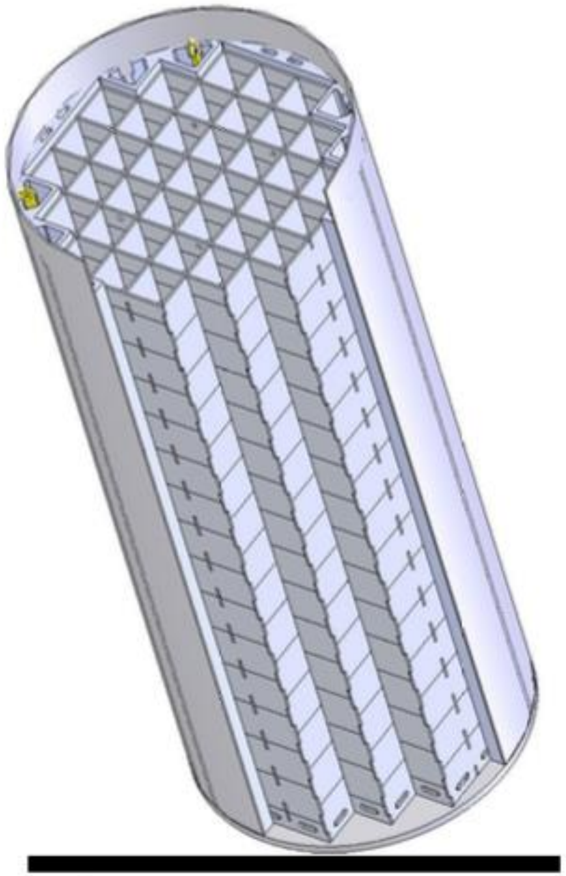
	PWR w/ oxide	SFR w/ metal	HTGR w/ TIRO
Example reactor	AP1000	Natrium [a)]	Xe-100
Power, MWt/MWe	3400 / 1117	840 / 345	200 / 80
Thermal efficiency	33%	41%	40%
Fuel form	UO_2	U-Zr w/o Na-bond	TRISO/Pebble
Burnup, GWd/t	50.0	146 [b)]	169
Uranium enrichment, %	4.2	16.5	15.5
Uranium mass, kg-U/assembly or pebble	539	114.3[b)]	7.0E-3
Assembly or pebble volume, m^3	0.220	0.104	1.13E-04
Fuel element volume-to-mass, m^3/t-HM	0.408	0.912 (2.2)	21.8 (53.4)
SNF mass, t/GWe-year	22.2	6.01 (0.27)	5.41 (0.24)
SNF volume, m^3/GWe-year	9.1	5.6 (0.62)	118 (13.0)

a) While Natrium power is slightly larger than SMR boundary of 300 MWe, it was included because most design features are close to SMRs than conventional reactors.
b) Due to the lack of information, data were obtained from a PRISM/Mod-B design that was revised to have the discharge burnup close to the Natrium design burnup of ~150 GWd/t.
*) T. K. Kim, L. Boing, B. Dixon., “Nuclear Waste Attributes of Near-Term Deployable Small Modular Reactors,” 2023 International Congress on Advanced in Nuclear Power Plants, April 2023.

- **SNF values are normalized to unit electricity generation (GWe-yr) and compared to PWR.**
- **HTGR generates a factor of 4.1 smaller SNF mass but produces a factor of 13 larger volume.**

CANISTER REQUIREMENTS

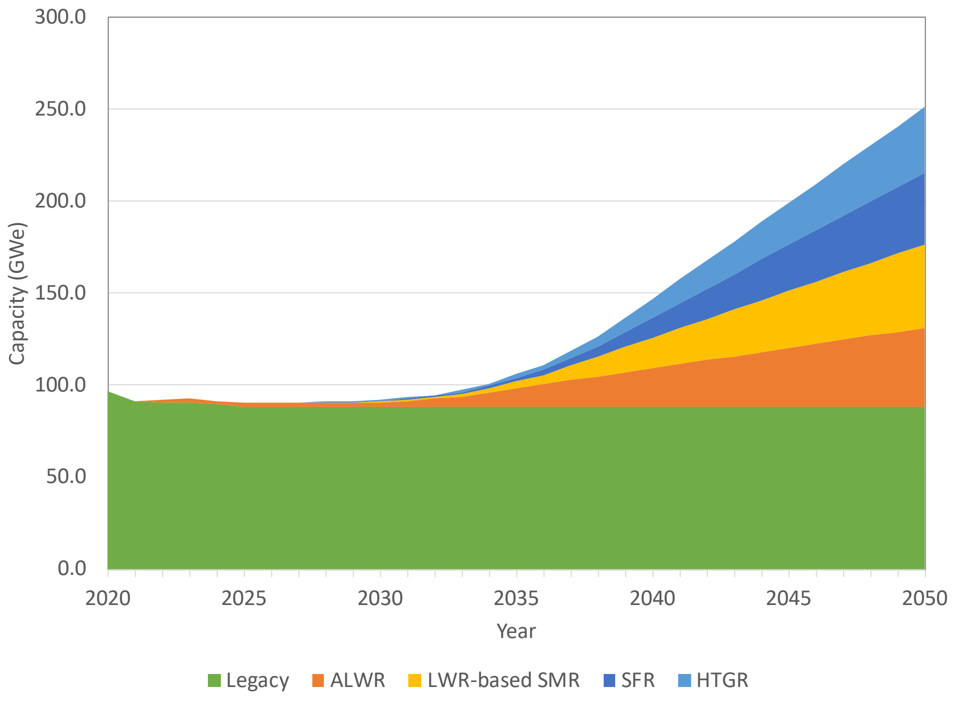
	PWR w/ oxide	SFR w/ metal	HTGR w/ TRISO
Example reactor	Ap1000	Natrium	Xe-100
Required canisters, #/GWe-yr	1.1	1.5	20.3
SNF loading, assemblies or pebbles	37	37	39,006
Heavy metal loading, initial MT-U/canister	19.9	4.1	0.3
Decay heat @ 5 yrs, kW/canister	56.5	21.9	2.4
Decay heat @ 50 yrs, kW/canister	16.7	6.9	0.6
Decay heat @ 1,000 yrs, kW/canister	1.5	0.6	0.03
A2-values @ 5 yrs/canister	1.0E+07	2.8E+06	3.4E+05
A2-values @ 50 yrs/canister	8.4E+06	2.1E+06	2.2E+05
Gammas (>1 MeV) @ 5 yrs/canister	1.0E+16	2.3E+15	3.0E+14
Gammas (>1 MeV) @ 50 yrs/canister	2.2E+14	7.0E+13	6.0E+12
Fissile mass of fresh fuel, kg/canister	839	722	42
Fissile mass of discharged fuel, kg/canister	305	450	4



MPC-37

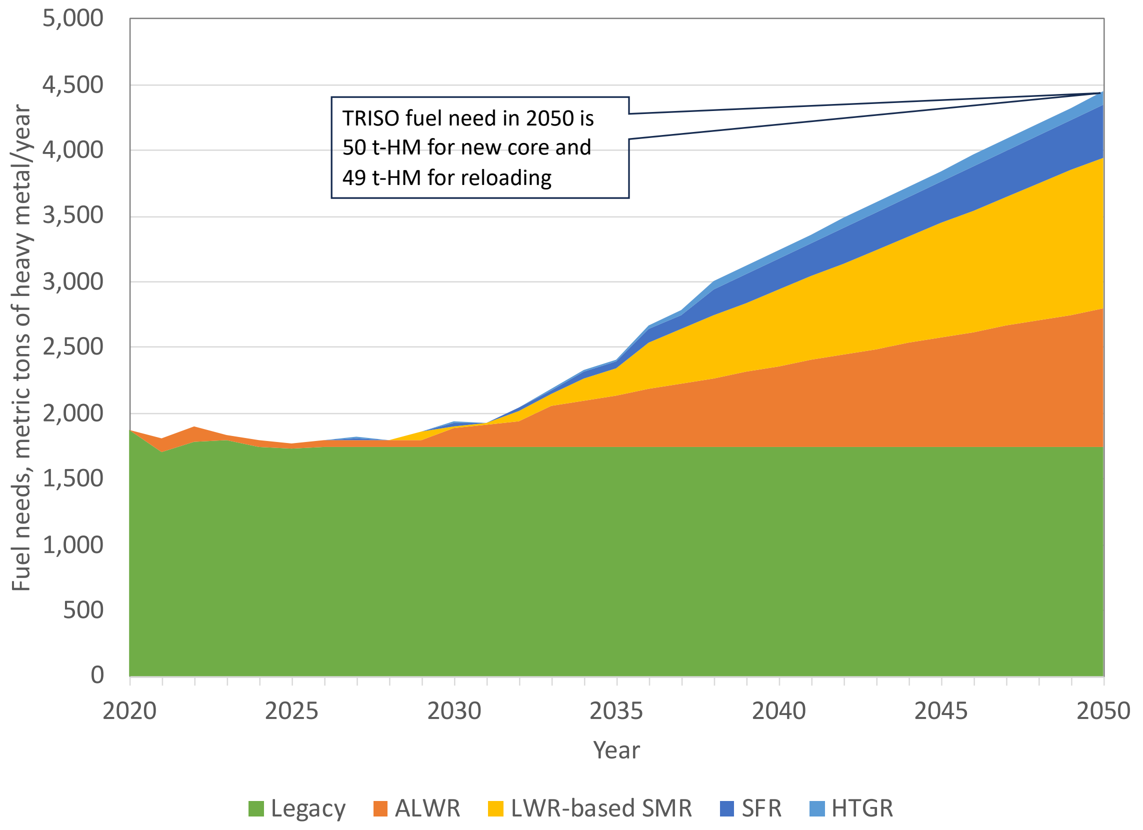
PROJECTION OF TRISO FUELS

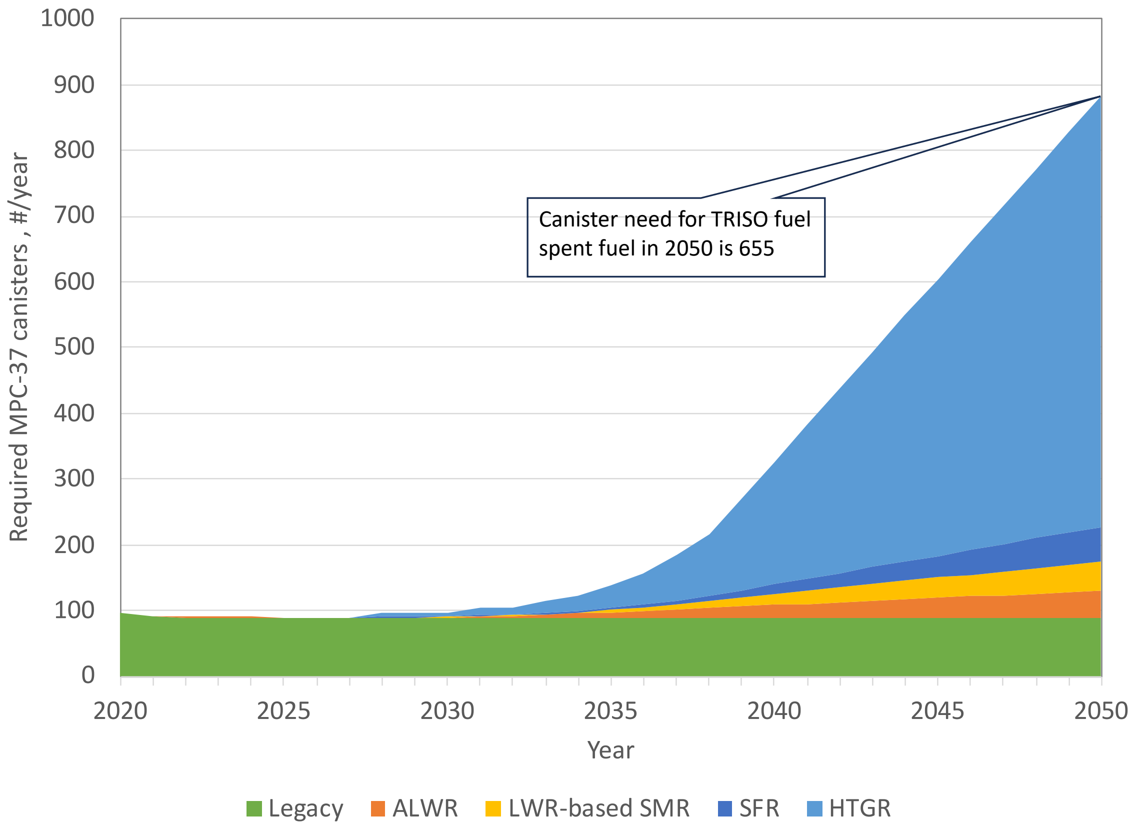
- **Nuclear capacity expansion scenarios**
  - Total nuclear capacity is ~250 GWe by 2050.
- **Reactor capacity mix (even distribution)**
  - 25%: Light water reactors (LWRs) with oxide fuel
  - 25%: LWR-based SMR with oxide fuel
  - 25%: SFR with metallic fuel
  - 25%: HTGR with TIRSO fuel
- **Reactor Representation**
  - Advanced LWR (ALWR) by AP-1000 design (1.1 GWe)
  - LWR-based SMR by NuScale (0.72 GWe/12-pack)
  - SFR by the Natrium (0.35 GWe)
  - HTGR by the Xe-100 design (0.32 GWe/4-pack)



- **Reactor capacity in 2050**
  - Legacy LWR: ~88 GWe with LEU
  - ALWR: ~43 GWe with LEU
  - LWR-based SMR: ~45 GWe with LEU
  - Small SFR: ~39 GWe with HALEU
  - Small HTGR: ~36 GWe with HALEU

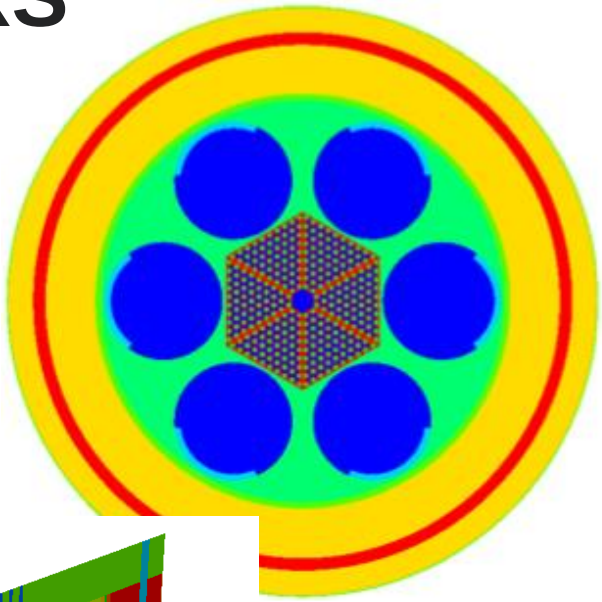
FUEL AND CANISTER NEEDS





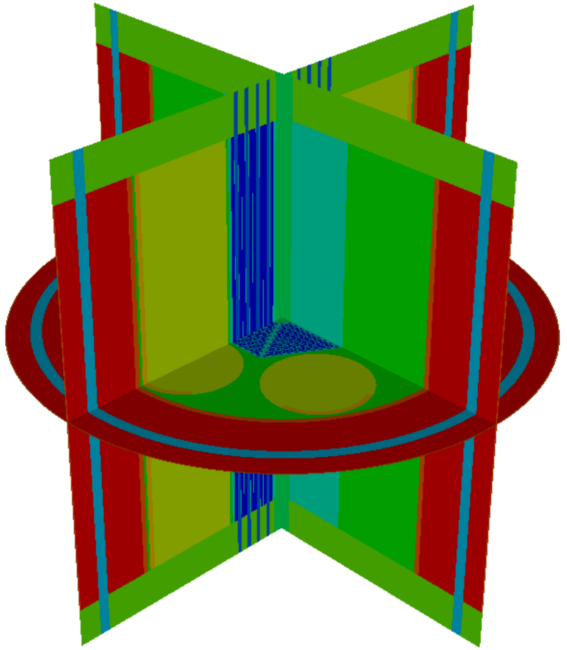
- **In 2050, the annual needs of TRISO fuel and MPC Canisters will be ~99 MT-U and 655, respectively, which are ~2% and ~74% of the total needs of the nuclear fleet.**

FLOODED CRITICALITY OF MICROREACTORS



- **The k-effective of several prismatic microreactors with TRISO fuels was calculated for flooded cases.**
  – Core size: various depending on the designs
  – With moderators and absorbers in transportation cases
  – Reactivity control
    - Control drums
    - Burnable poison
- **Flooding of gas channels with cold water raises k-eff more than 20,000 pcm.**

Flooded?	k-eff	Δk [pcm]	Control drum location	Additional absorber for transportation
Yes	> 1.00	-	Facing out	No
No		21,000	Facing center	No
Yes			Facing center	No
No		23,000	Facing center	Yes
Yes			Facing center	Yes



CONCLUSIONS

- **TRISO SNF characteristics were normalized to unit electricity generation (GWe-yr) and compared with oxide and metallic fuels.**
- **HTGR with TRISO fuel generates less spent fuel mass, but more volume.**
  - TRISO SNF contains less decay heat, gammas, and A-2 than oxide or metallic SNFs.
  - However, the required canisters for unprocessed TRISO SNF is a factor of 18 larger than oxide and metallic SNFs, which may affect the back-end fuel cycle costs and required capacity repository.
- **An appropriate reactivity control feature is required for the transportation of microreactors.**
  - K-effective increases by 21,000 – 25,000 pcm when gas channels are flooded with cold water.

Management of TRISO spent fuel using a Universal Canister System

Jesse Sloane, PE
Executive VP of Engineering
Deep Isolation

Steven Sisley
Program Manager, Cask Development Projects
NAC International, Inc.

December 5, 2024

ONWARDS
UPWARDS

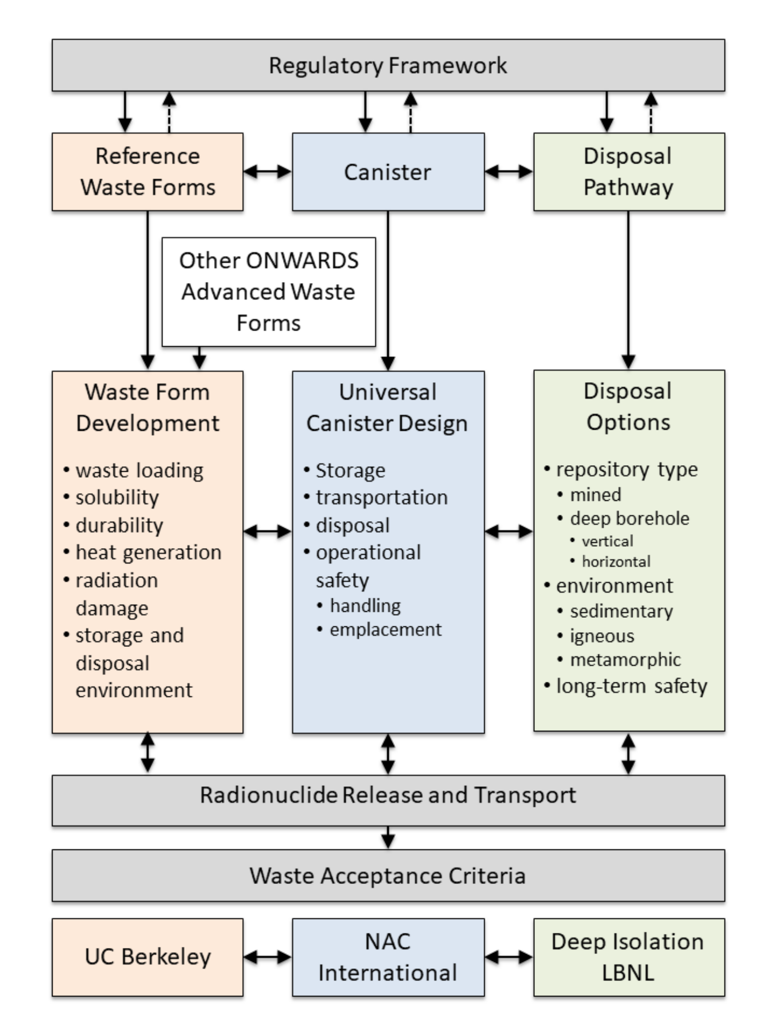
DEEP ISOLATION®
BERKELEY LAB
Berkeley UNIVERSITY OF CALIFORNIA
NAC INTERNATIONAL

Project Overview

- Project **UPWARDS**

  U*niversal* ***P****erformance Criteria and Canister for Advanced Reactor* ***W****aste Form* ***A****cceptance in Borehole and Mined* ***R****epositories Considering* ***D****esign* ***S****afety*

- ~2 years into 3-year project
- Four workstreams:
  1. Waste Form Development – UCB
  2. Canister Design and Prototype – NAC
  3. Models and Generic Performance Assessment – DI/LBNL
  4. Regulatory Review and Waste Acceptance Criteria – DI

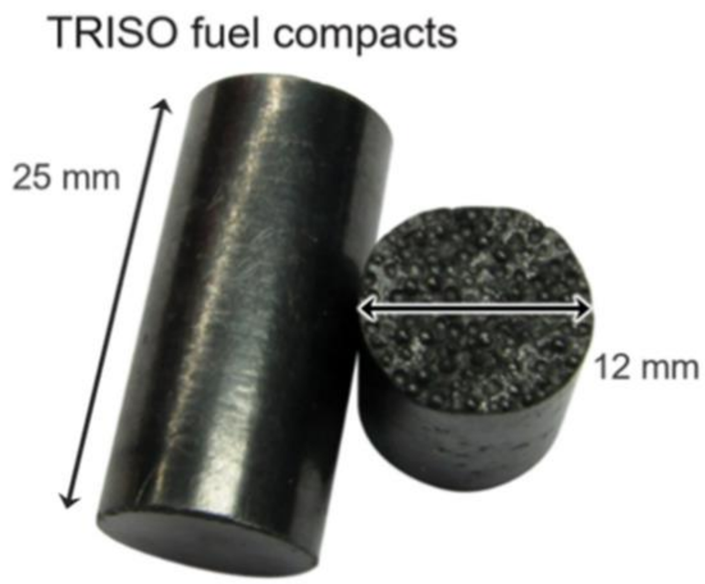


Workstream 1: Waste Form Development

- Waste form research
  - Lanthanide Borosilicate (LaBS) glass
  - TRISO
  - Intact halide salt
- Developed standards & specifications gap analysis
- Running experiments to determine safety-relevant parameters
- Culminates in recommended standards and specifications for waste forms

→ performance assessment inputs

Berkeley
UNIVERSITY OF CALIFORNIA



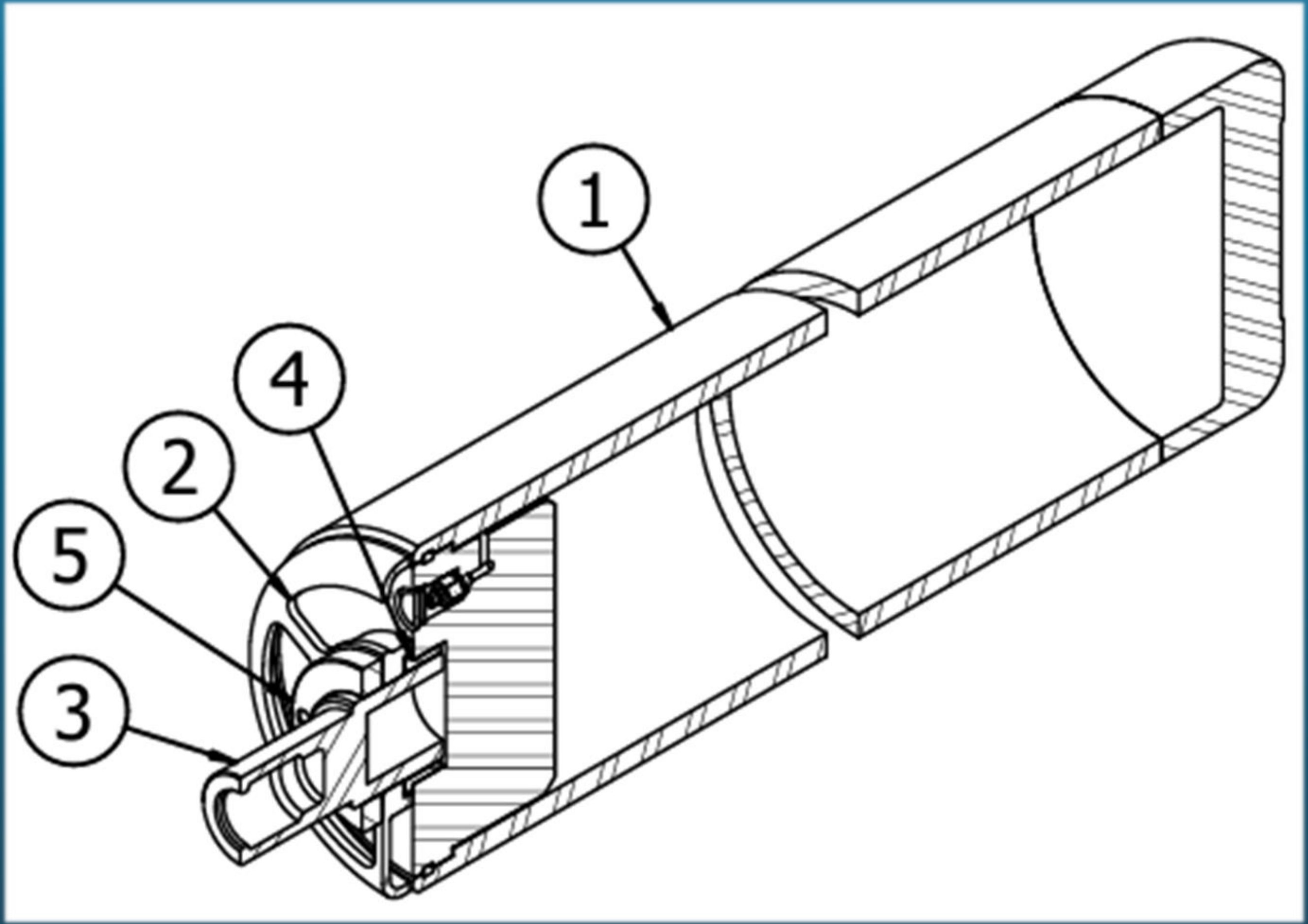
Demkowicz, Paul, and Hunn, John D. *Two decades of DOE investment lays the foundation for TRISO-fueled reactors*. United States: N. p., 2020. Web.

Workstream 2: Preliminary Universal Canister Design

- Design & Analysis
  - Functional Requirements Specification
  - Design Specification
  - Preliminary Design & Analysis
    - Evaluations of bounding configurations for the most limiting storage, transportation, and disposal conditions
- Prototype Fabrication

Discipline	Evaluation Description
Structural	UCS External Pressure and Stacking Analysis (Borehole Disposal)
	UCS Handling Stress Analysis
	UCS Transfer 20-foot End Drop Evaluation
	UCS Borehole Drop
Nuclear	UCS Shielding Evaluation for TRISO Fuel
	UCS Criticality Evaluation for TRISO Fuel (Transportation and Disposal)
Thermal	UCS Thermal Analysis for Borehole Disposal

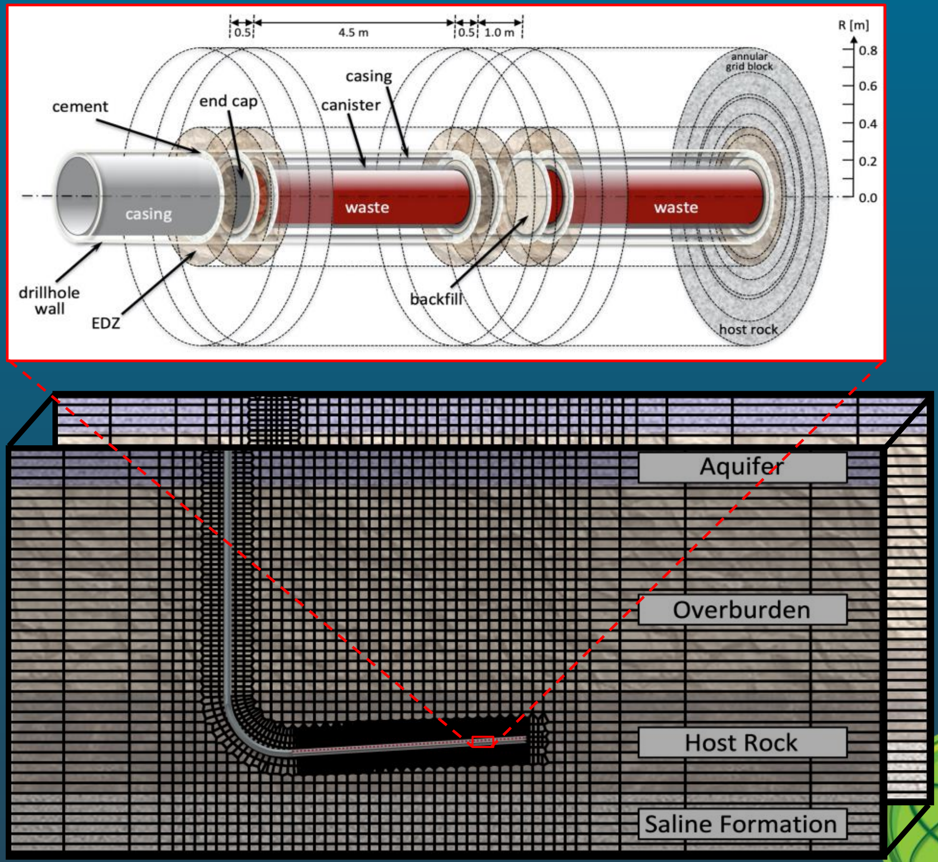
UCS Preliminary Design



Item No.	Description
1	UCS Assembly
2	Cap Plate
3	Lift Adapter
4	Key Plate
5	Jam Plate

Workstream 3: Integrated Safety & Performance Assessments

- Developed
  - Representative safety & performance assessment models for mined and borehole repositories
  - Source-term models
- In Progress
  - Calculating safety & performance to determine performance envelope
    - ➢ Input to Workstream 4 (WAC development)
  - Documentation of results



Workstream 4: Development of Waste Acceptance Criteria

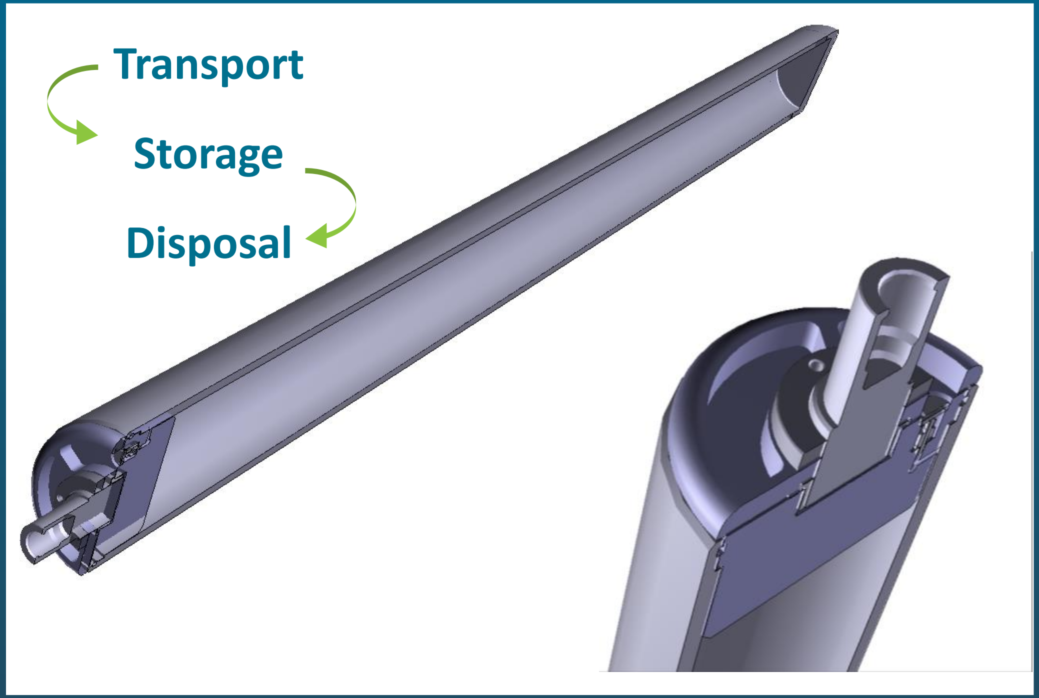
- Developed
  - Review of existing regulatory requirements and waste acceptance criteria
- In Progress
  - Formulation of generic WAC
  - WAC coupling against disposal configurations

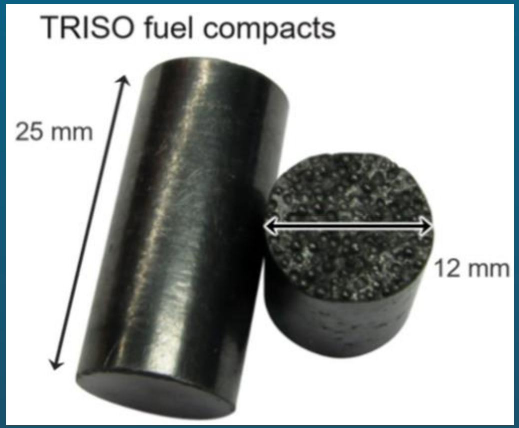
DEEP ISOLATION®

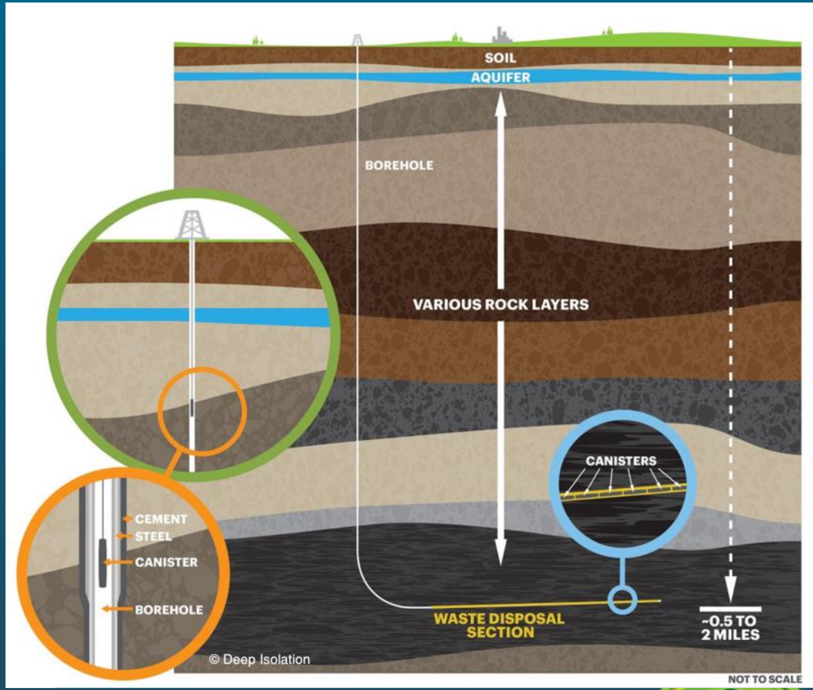
Regulatory Topic	Consideration
Borehole Conditions	Extent of design flexibility
	Credit for overly robust boreholes
UCS Design Credit	Extent to which canister factors into "canister plus waste acceptability"
	Credit for radiological, chemical, and mechanical stability
Retrievability Period	Assumed timing for retrievability, backfilling, and sealing
	Minimum conditions thereafter
Managing AR Waste versus Legacy Waste	Extent to which lessons learned from LWR mgt informs AR guidance

Configuration Applicability & Pairing Matrix

Canister Class ⟷ Waste Form ⟷ Repository Configuration







International Applications

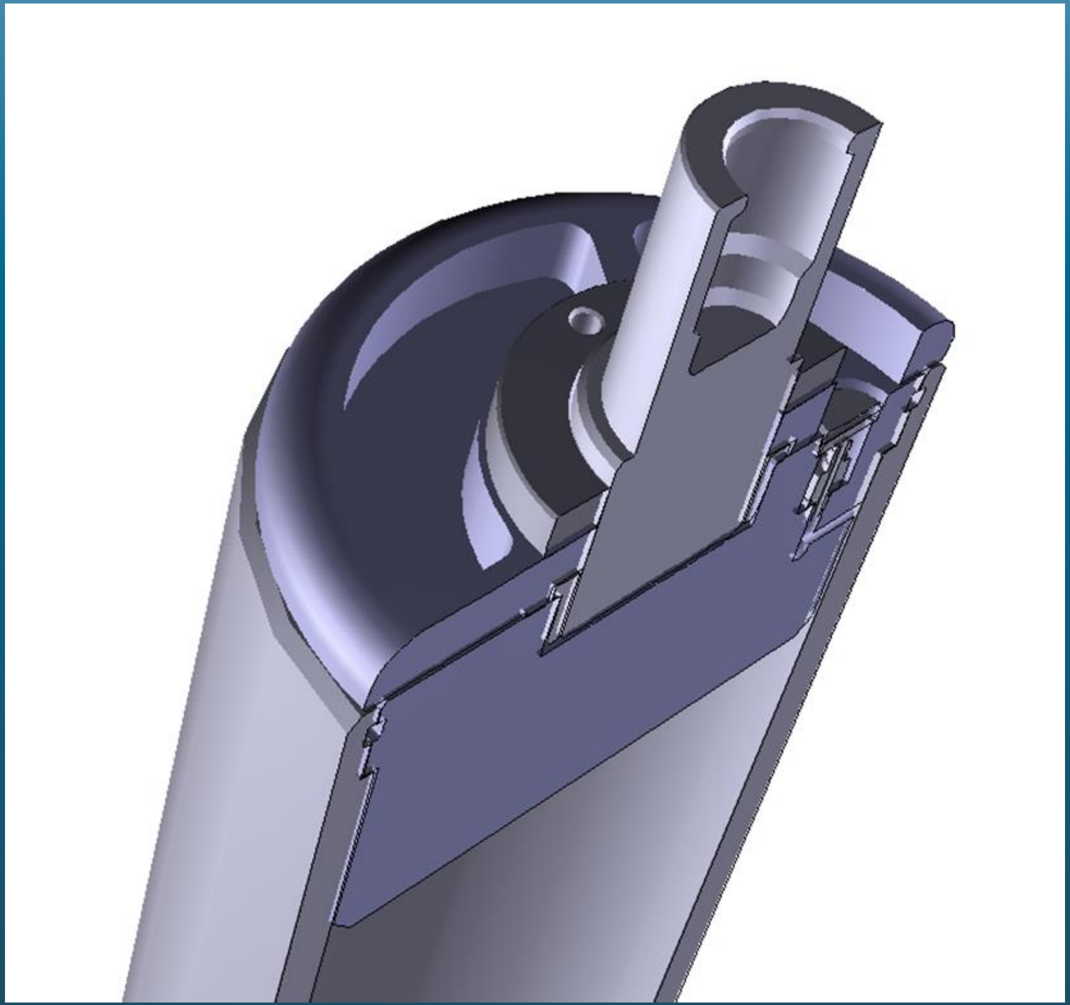
Joint Project on Waste Integration for Small and Advanced Reactor Designs (WISARD)

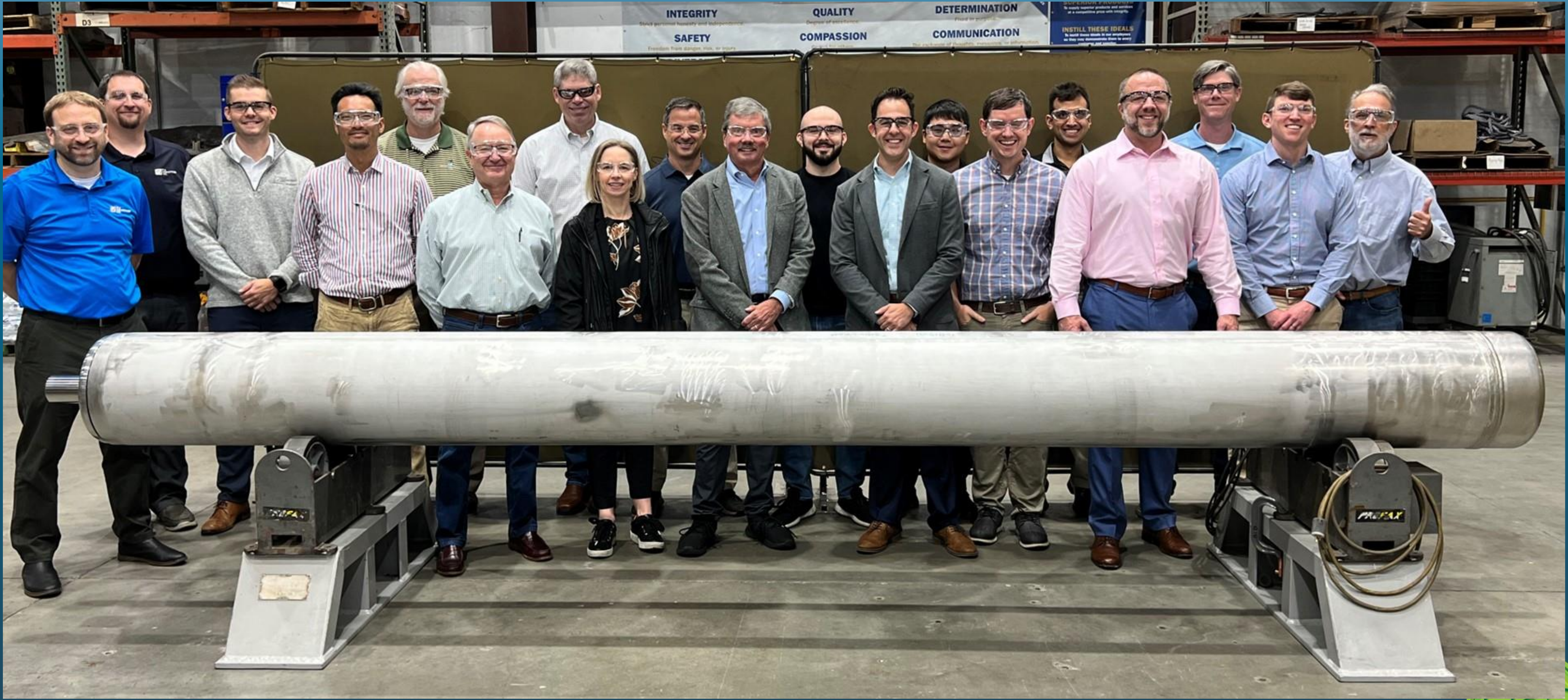
- Treatment, recycling or reprocessing
- Spent fuel and waste storage
- Transportation
- Disposal

NEA
NUCLEAR ENERGY AGENCY

Conclusion

- Project concludes in July 2025
- Will deliver first-of-a-kind fully-integrated waste management system
  - Accommodate 3 waste forms
  - Compliant to the design and analysis
  - Demonstrated suitability in generic performance envelope
  - Capable of pairing with waste forms and disposal methods using WAC




INTEGRITY
QUALITY
DETERMINATION
SAFETY
COMPASSION
COMMUNICATION
INSTILL THESE IDEALS

Thank you!

PROJECT UPWARDS

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Microreactor Transportation Emergency Planning Challenges

Steven J. Maheras

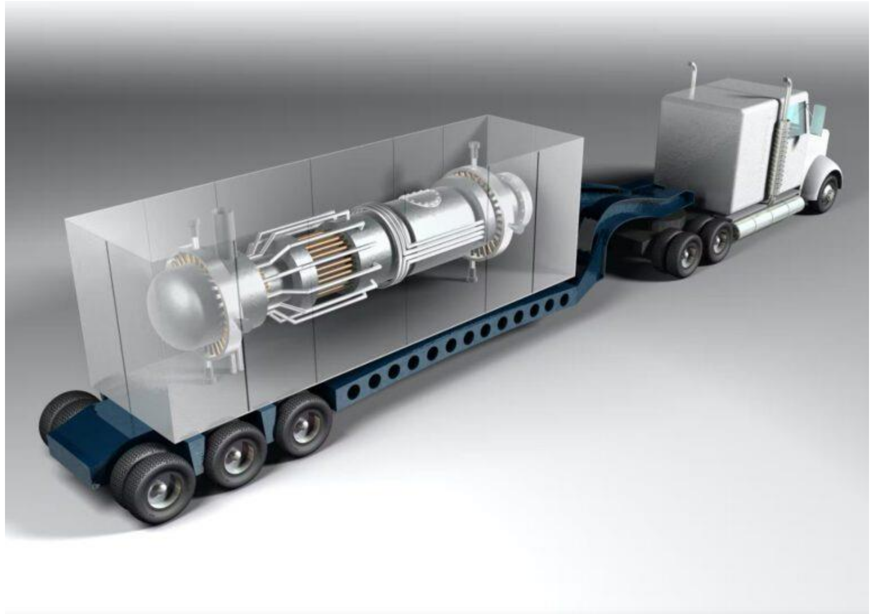
Pacific Northwest National Laboratory

2024 Workshop on Storage and Transportation of TRISO and Metal Spent Nuclear Fuels

December 3-5, 2024

Microreactors (Transportable Nuclear Power Plant or TNPPs)

- Microreactors are a class of very small modular reactors targeted for non-conventional nuclear markets
  - Also known as transportable nuclear power plants or TNPPs
- There are a variety of microreactor/advanced reactor designs, including gas, liquid-metal, molten-salt, and heat-pipe-cooled concepts
- Potential microreactor applications are:
  - Remote communities
  - Mining sites
  - Remote defense bases
  - Applications such as back-up generation for power plants
  - Humanitarian aid and disaster relief missions



Semi-Tractor and Trailer Carrying Reactor Module

MRP Microreactor Program

Key Attributes of Microreactors

- Microreactors have key features enabled by their small size that distinguish them from other reactor types mainly large reactors (LWRs) and small modular reactors (SMRs).
- These are:
  - Typically produce less than 20 MW thermal
    - Smaller size needed to remain transportable/deployable
  - Smaller footprint
  - Factory fabrication
  - Transportable
  - Self-regulating (enabling remote and semi-autonomous microreactor operation)
  - Rapid deployability and availability during emergency response
  - Possible operation up to 10 years or more

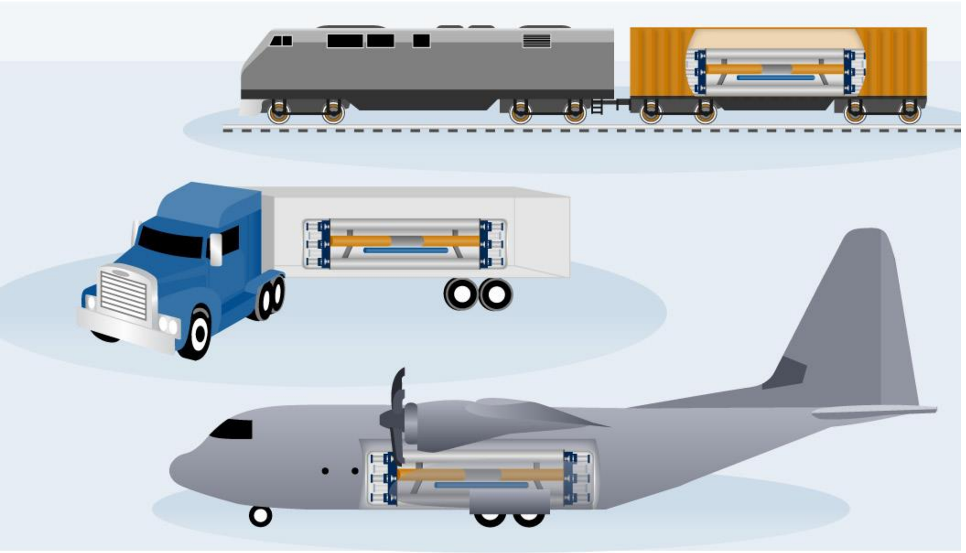
MRP Microreactor Program

Commercial Microreactor Developers and Types

Developer	Name	Type	Power Output (MWe/MWth)	Fuel	Coolant	Moderator	Refueling Interval
Aalo Atomics	Aalo One	STR	7 MWe/20MWth	U-Zr-H	Sodium	H	3-5 years
Alpha Tech Research Corp	ARC Nuclear Generator	MSR	12 MWe/30 MWth	LEU	Flouride salt		intermittent
Antares Industries	R1	Sodium Heat Pipe	1.2 MWth	TRISO	Sodium	Graphite	
BWXT	BANR	HTGR	17 MWe/50 MWth	TRISO	Helium	Graphite	5 years
Deep Fission	DB-PWR	PWR	1-15 MWE	LEU	Water	Water	4-6 years
General Atomics	GA Micro	HTGR	1-10 MWe		Gas		
HolosGen	HolosQuad	HTGR	13 MWe	TRISO	Helium/CO2		10 years
Micro Nuclear, LLC	Micro Scale Nuclear Battery	MSR/Heat Pipe	10 MWe	UF4	FLiBe	YH	10 years
Nano Nuclear	Zeus/Odin	HTGR/MSR	1.0 MWe/2.5 MWth	UO2	Helium		
NuCube	Nu3	Heat Pipe	1 MWe/3 MWth	TRISO	Sodium	Graphite	10+ years
NuGen, LLC	NuGen Engine	HTGR	2-4 MWe	TRISO	Helium		
NuScale Power	NuScale Microreactor	LMTM/Heat Pipe	<10 MWe	Metallic	Liquid Metal	Liquid Metal	10 years
Oklo	Aurora	SFR	15 MWe	Metallic (U-Zr)	Sodium		10+ years
Radiant Nuclear	Kaleidos Battery	HTGR	1.2 MWe	TRISO	Helium	Graphite	4-6 years
Ultra Safe Nuclear	Micro Modular Reactor	HTGR	5 MWe/15 MWth	TRISO	Helium	Graphite	20 years
Westinghouse	eVINCI	Sodium Heat pipe	5 MWe/15 MWth	TRISO	Sodium	Graphite	8 years
X-Energy	XENITH	HTGR	5 MWe/10 MWth	TRISO	Helium	Graphite	3+ years

Microreactor Transportation

- Current microreactor concepts are to transport the microreactor containing its unirradiated or irradiated fuel
- A microreactor with its unirradiated or irradiated contents is unlikely to meet the entire suite of regulatory requirements in U.S. Nuclear Regulatory Commission (NRC) transportation regulations (10 CFR Part 71)
- A risk-informed process will likely be used initially for NRC transportation package approval
  - Demonstrate equivalent safety and that risk to the public is low
  - This will probably require the use of compensatory measures



Source: GAO. | GAO-20-380SP

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Current Transportation Approach

- The microreactor shipment would be a commercial shipment and would receive transportation package approval from the NRC using a risk-informed process
- Strategy is Crawl-Walk-Run
  - Concentrate on highway transport first
  - Then other surface modes (rail and barge/ship) – evaluation of transport by vessel has just started
  - Finally air transport
- The microreactor containing its irradiated fuel would contain a *highway route-controlled quantity* (HRCQ) of radioactive material (i.e., > 3000 A_2)
  - For truck shipments this means that a Commercial Vehicle Safety Alliance (CVSA) Level VI inspection and safety permit would be required (see 49 CFR 385 and 49 CFR 397)
  - For rail shipments this means that the transportation planning requirements in 49 CFR 172.820 would apply
- The microreactor would be fueled by LEU or HALEU (not HEU)
- For rail shipments, transport would be via Association of American Railroads (AAR) Standard S-2043 railcars
- Marine shipments would be via a Class INF-2 or INF-3 ship

LEU= Low Enriched Uranium (< 5 % U-235)
HALEU= High Assay Low Enriched Uranium (5-20 % U-235)
HEU= High Enriched Uranium (> 20% U-235)

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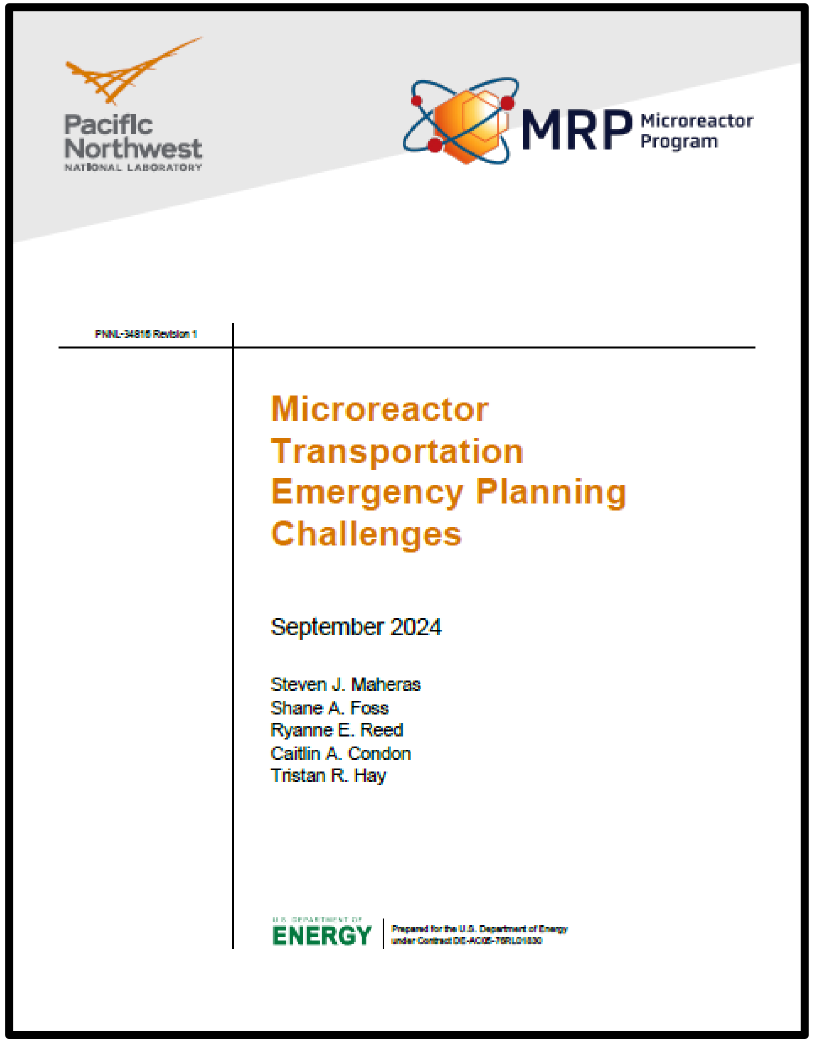
Microreactor Transportation Emergency Response Planning Challenges

- The U.S. Department of Energy Office of Nuclear Energy (DOE-NE) Microreactor Program is working to identify potential microreactor transportation emergency response planning challenges
- The U.S. Department of Energy (DOE) National Transportation Stakeholders Forum has been used to obtain input from the Tribes and States on these potential challenges


Pacific Northwest NATIONAL LABORATORY

MRP Microreactor Program

PNNL-34816 Revision 1

Microreactor Transportation Emergency Planning Challenges

September 2024

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U.S. DEPARTMENT OF ENERGY | Prepared for the U.S. Department of Energy under Contract DE-AC05-76RL01830

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Areas Examined In Identifying Microreactor Transportation Emergency Planning Challenges

• Assignment of Responsibility	• Accident Assessment
• Emergency Response Organization	• Protective Response
• Emergency Response Support and Resources	• Radiological Exposure Control
• Emergency Classification System	• Medical and Public Health Support
• Notification Methods and Procedures	• Recovery, Reentry, and Post-Accident Operations
• Emergency Communications	• Exercises and Drills
• Public Education and Information	• Radiological Emergency Response Training
• Emergency Facilities and Equipment	• Responsibility for the Planning Effort: Development, Periodic Review, and Distribution of Emergency Plans

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Results of Evaluation

- Microreactor transportation emergency planning challenges organized into cross-cutting challenges and specific transportation emergency response challenges
- This presentation will discuss several cross-cutting transportation emergency planning challenges
  - Use of hazardous materials in microreactor designs
  - Revisions to the DOT Emergency Response Guidebook
  - Potential microreactor fuel type issues
  - Criticality and Proposed Rule 10 CFR Part 53, Risk-Informed, Technology-Inclusive Regulatory Framework for Commercial Nuclear Plants
  - Potential compensatory measures
  - External Engagement, Emergency Response Training, and Accident Recovery Plans
- Discussions with States and Tribes also discussed

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Use of Hazardous Materials in Microreactor Designs

- Beryllium-containing materials are currently being investigated for use in microreactors as replacements for graphite as a neutron moderator (Cheng et al. 2022)
- Beryllium is a hazardous material and if these beryllium-containing materials were incorporated into a microreactor, the presence of these materials would have to be considered in the transportation emergency response planning for these specific microreactors
- Sodium-containing heat pipes are being investigated for use in some microreactors
- Sodium is a hazardous material and the presence of sodium would have to be considered in the transportation emergency response planning for these microreactors, specifically in two areas:
  - The ability of sodium in combination with water to exacerbate releases of radioactive material during a transportation accident, and
  - The need to modify transportation accident fire-fighting guidelines if sodium was present

Source: Cheng B., E. M. Duchnowski, D. J. Sprouster, L. L. Snead, N. R. Brown, and J. R. Trelewicz. 2022. "Ceramic Composite Moderators as Replacements for Graphite in High Temperature Microreactors." Journal of Nuclear Materials. Volume 563.

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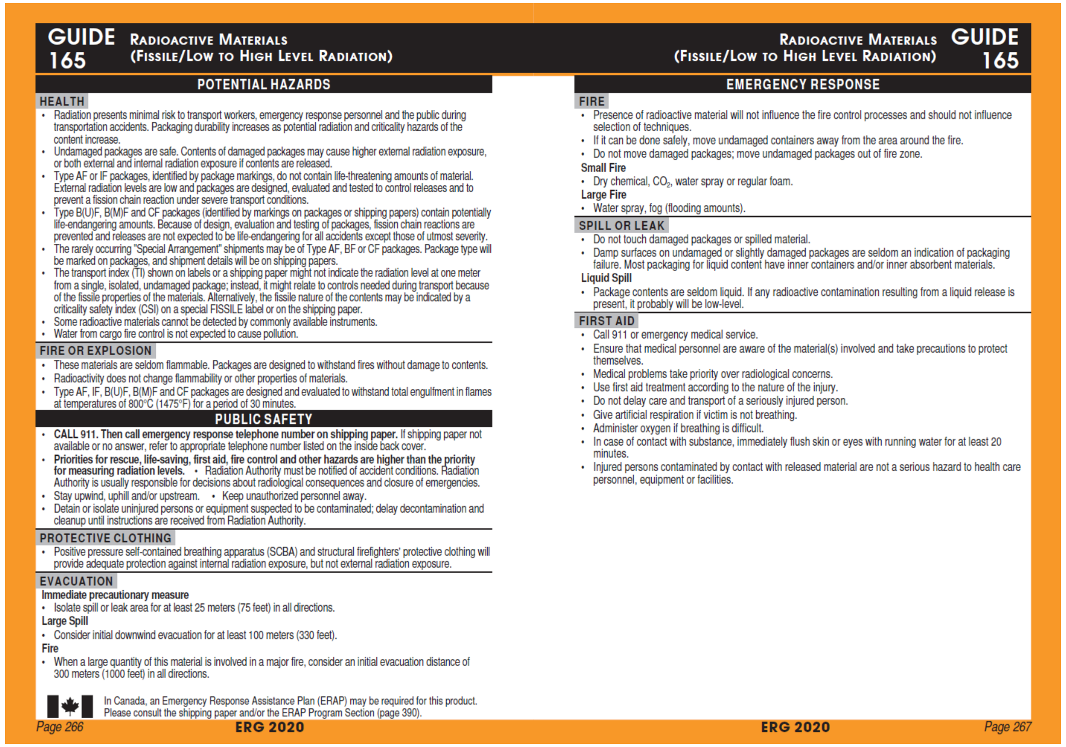
Emergency Response Guidebook (ERG)

- The U.S. Department of Transportation (DOT) Pipeline and Hazardous Materials Safety Administration (PHMSA) ERG provides first responders with a manual to help deal with hazardous materials transportation accidents during the critical first 30 minutes after the accident
- Emergency responders are trained to use the shipping papers, numbered placard, or orange panel number to determine which emergency response guide to use in responding to the accident
- The emergency response guides were not developed based on transportation accidents involving microreactors containing irradiated fuel
- The ERG would have to be expanded to include a guide that is specific to microreactor transportation accidents
- The guide may have to be fuel-type specific because of the differences in potential releases from different microreactor fuel types
- The guide may also have to be modified to account for the presence of hazardous materials such as beryllium or sodium



GUIDE 165 RADIOACTIVE MATERIALS (FISSILE/LOW TO HIGH LEVEL RADIATION)

POTENTIAL HAZARDS

HEALTH
- Radiation presents minimal risk to transport workers, emergency response personnel and the public during transportation accidents. Packaging durability increases as potential radiation and criticality hazards of the content increase.
- Undamaged packages are safe. Contents of damaged packages may cause higher external radiation exposure, or both external and internal radiation exposure if contents are released.
- Type AF or IF packages, identified by package markings, do not contain life-threatening amounts of material. External radiation levels are low and packages are designed, evaluated and tested to control releases and to prevent a fission chain reaction under severe transport conditions.
- Type B(U)F, B(M)F and CF packages (identified by markings on packages or shipping papers) contain potentially life-endangering amounts. Because of design, evaluation and testing of packages, fission chain reactions are prevented and releases are not expected to be life-endangering for all accidents except those of utmost severity.
- The rarely occurring "Special Arrangement" shipments may be of Type AF, BF or CF packages. Package type will be marked on packages, and shipment details will be on shipping papers.
- The transport index (TI) shown on labels or a shipping paper might not indicate the radiation level at one meter from a single, isolated, undamaged package; instead, it might relate to controls needed during transport because of the fissile properties of the materials. Alternatively, the fissile nature of the contents may be indicated by a criticality safety index (CSI) on a special FISSILE label or on the shipping paper.
- Some radioactive materials cannot be detected by commonly available instruments.
- Water from cargo fire control is not expected to cause pollution.

FIRE OR EXPLOSION
- These materials are seldom flammable. Packages are designed to withstand fires without damage to contents.
- Radioactivity does not change flammability or other properties of materials.
- Type AF, IF, B(U)F, B(M)F and CF packages are designed and evaluated to withstand total engulfment in flames at temperatures of 800°C (1475°F) for a period of 30 minutes.

PUBLIC SAFETY
- **CALL 911. Then call emergency response telephone number on shipping paper.** If shipping paper not available or no answer, refer to appropriate telephone number listed on the inside back cover.
- **Priorities for rescue, life-saving, first aid, fire control and other hazards are higher than the priority for measuring radiation levels.** • Radiation Authority must be notified of accident conditions. Radiation Authority is usually responsible for decisions about radiological consequences and closure of emergencies.
- Stay upwind, uphill and/or upstream. • Keep unauthorized personnel away.
- Detain or isolate uninjured persons or equipment suspected to be contaminated; delay decontamination and cleanup until instructions are received from Radiation Authority.

PROTECTIVE CLOTHING
- Positive pressure self-contained breathing apparatus (SCBA) and structural firefighters' protective clothing will provide adequate protection against internal radiation exposure, but not external radiation exposure.

EVACUATION

Immediate precautionary measure
- Isolate spill or leak area for at least 25 meters (75 feet) in all directions.

Large Spill
- Consider initial downwind evacuation for at least 100 meters (330 feet).

Fire
- When a large quantity of this material is involved in a major fire, consider an initial evacuation distance of 300 meters (1000 feet) in all directions.

In Canada, an Emergency Response Assistance Plan (ERAP) may be required for this product. Please consult the shipping paper and/or the ERAP Program Section (page 390).

Page 266 ERG 2020

RADIOACTIVE MATERIALS (FISSILE/LOW TO HIGH LEVEL RADIATION) GUIDE 165

EMERGENCY RESPONSE

FIRE
- Presence of radioactive material will not influence the fire control processes and should not influence selection of techniques.
- If it can be done safely, move undamaged containers away from the area around the fire.
- Do not move damaged packages; move undamaged packages out of fire zone.

Small Fire
- Dry chemical, CO_2, water spray or regular foam.

Large Fire
- Water spray, fog (flooding amounts).

SPILL OR LEAK
- Do not touch damaged packages or spilled material.
- Damp surfaces on undamaged or slightly damaged packages are seldom an indication of packaging failure. Most packaging for liquid content have inner containers and/or inner absorbent materials.

Liquid Spill
- Package contents are seldom liquid. If any radioactive contamination resulting from a liquid release is present, it probably will be low-level.

FIRST AID
- Call 911 or emergency medical service.
- Ensure that medical personnel are aware of the material(s) involved and take precautions to protect themselves.
- Medical problems take priority over radiological concerns.
- Use first aid treatment according to the nature of the injury.
- Do not delay care and transport of a seriously injured person.
- Give artificial respiration if victim is not breathing.
- Administer oxygen if breathing is difficult.
- In case of contact with substance, immediately flush skin or eyes with running water for at least 20 minutes.
- Injured persons contaminated by contact with released material are not a serious hazard to health care personnel, equipment or facilities.

ERG 2020 Page 267

Potential Microreactor Fuel Type Issues

- Many commercial and military microreactor designs use TRISO fuel
  - Releases from TRISO fuels from transportation accidents are expected to be low
  - However, there is a lack of impact testing information for TRISO fuels that can be related to the hypothetical accident conditions (HAC) in 10 CFR 71.73
  - This lack of impact testing information represents a data gap that would need to be resolved during the transportation package approval process
- For metal fuels, their performance during transportation accidents is likely to be less robust than TRISO fuels, presenting significant challenges
  - Vendors may choose to ship their microreactors unfueled

Criticality and Proposed Rule 10 CFR Part 53, Risk-Informed, Technology-Inclusive Regulatory Framework for Commercial Nuclear Plants

- Draft 10 CFR 53.620(d), Fuel Loading
  - A manufacturing license may include authorizing the loading of fresh (unirradiated) fuel into a manufactured reactor under Part 70
- Specifies required protections to prevent criticality
  - At least two independent physical mechanisms in place, each of which is sufficient to prevent criticality assuming optimum neutron moderation and neutron reflection conditions
- The way that this requirement is met may (will?) trickle into 10 CFR Part 71 space

Potential Compensatory Measures

- Microreactors containing irradiated fuel shipped by highway would be highway route-controlled quantity (HRCQ) (> 3000 A2) shipments and would need to meet the routing requirements in 49 CFR Part 397
  - The use of interstates, beltways around cities, state identified preferred routes could be considered as compensatory measures
- Microreactors will likely be overweight/overdimension and will require state permitting when transported by highway
  - Specific heavy haul truck or superload permit requirements could be considered as compensatory measures

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Other Potential Compensatory Measures Could Include

- Increased exclusion zone around the microreactor because of possible radiation dose rate increase
- Real time health/fitness onboard monitoring/diagnostics of reactor package
- Escorting of the reactor forward and aft for the entire route
- Rolling road closures
- Travel at reduced speeds
- Choosing a route that avoids bodies of water (balanced by quality of road)
- Controls for bridges over bodies of water (bridge inspection, speed reduction, close bridge to other traffic)
- Judicious use of time-of-day and day-of-week restrictions
- Avoid shipping during severe weather
- Conduct training and exercises for emergency responders along the route

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Potential Issue Associated with Compensatory Measures

- It is likely that NRC microreactor transportation package approval would be conducted using a risk-informed process and the microreactor containing irradiated fuel may not meet the 10 mrem/h (0.1 mSv/h) at 2 meters from the conveyance dose rate limit contained in 49 CFR 173.441 and 10 CFR 71.47
- As a result, the microreactor may require a stand-off distance of approximately 30 meters to obtain a dose rate of 10 mrem/h (0.1 mSv/h), depending on the amount of shielding and storage time
- This could have implications for transportation emergency response planning if external package dose rates keep responders and recovery crews from meeting necessary objectives for recovery and mitigation

External Engagement, Emergency Response Training, and Accident Recovery Plans

- Would need to conduct external engagement prior to transporting a microreactor containing its irradiated fuel
  - A microreactor containing its irradiated fuel has not been shipped in the U.S., and State and Tribal emergency responders along potential routes are likely to be unfamiliar with microreactor transport
  - This engagement could take 2 to 3 years
- Potential need to conduct emergency response training and exercises along transport routes and identification of potential safe havens
- Potential need to develop transportation accident recovery plans

Health Monitoring Information System

- Incorporating a health monitoring instrumentation system into the design of a microreactor would enable data on the status of the microreactor to be collected during its transport
- This data could be used to reduce the risks associated with transporting a microreactor containing irradiated fuel
- Three categories of data to be collected have been identified
  - Reactor Safety – neutron fluence and dose rate, temperature monitoring, vehicle shock and vibration data
  - Shielding – radiation dose rates around reactor and supplemental shielding
  - Radiological Containment – reactor vessel pressure, fission gas activity, intrusion monitors and alarms

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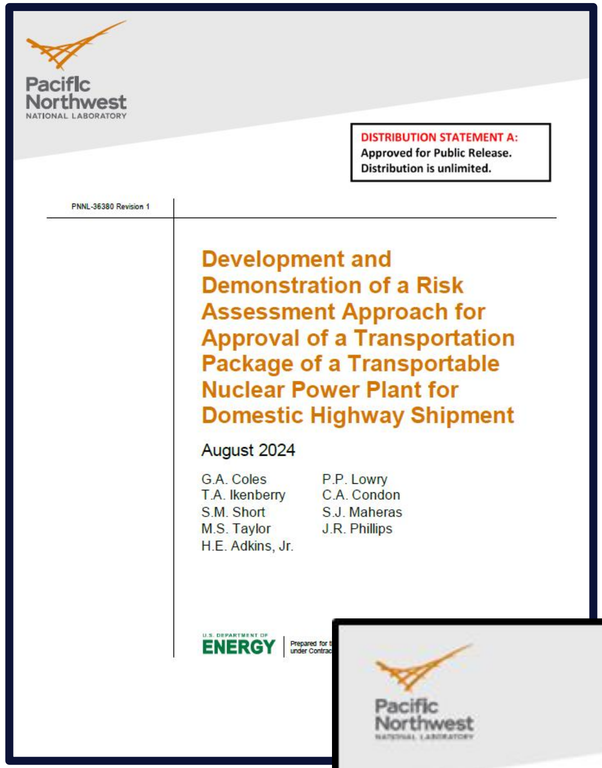
Discussions with States and Tribes

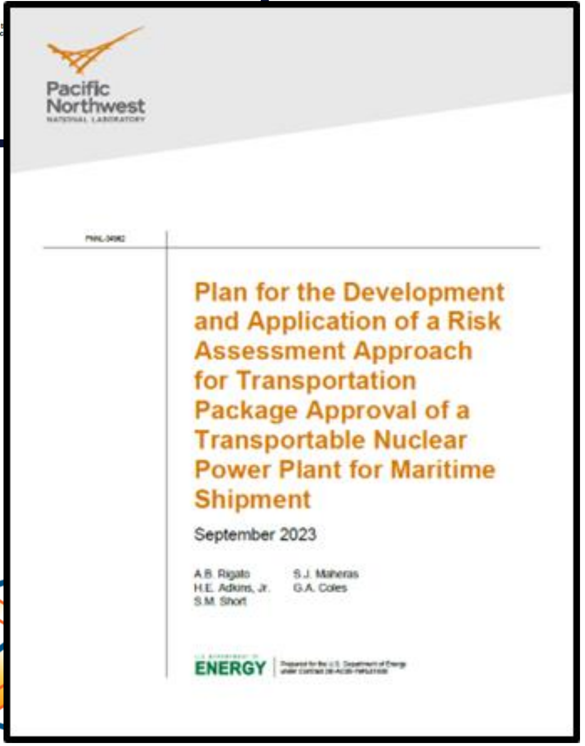
- In general, the transportation emergency response community is not familiar with microreactors or the concept of transporting a microreactor containing its irradiated fuel
- The purpose of the discussions is to obtain State and Tribal perspectives on the potential emergency planning challenges associated with the transportation of a microreactor containing its irradiated fuel
- The challenges may differ from shipments of spent nuclear fuel in Type B transportation casks (the current paradigm)
- Some challenges are likely to be mode-specific (i.e., different for shipment by truck, rail, air, and vessel)
- Some challenges will be design-specific, e.g., presence of other hazardous materials, or fuel type-specific

Program

Collaboration Activities

- Working closely with the U.S. Department of Defense (DoD) Strategic Capabilities Office (SCO)
  - Project Pele
  - Presented risk-informed transportation package approval methodology to U.S. Nuclear Regulatory Commission (NRC) Advisory Committee on Reactor Safeguards (ACRS)
    - ACRS Subcommittee – November 17, 2023
    - Full ACRS – December 6, 2023
    - Endorsement of methodology by NRC – October 2024
- Working closely with U.S. Army Reactor Office (ARO), U.S. Army Office of the Chief of Engineers (OCE), DoD Operational Energy Capability Improvement Fund (OECIF), and National Reactor Innovation Center (NRIC)
  - Current emphasis on maritime transport of microreactors





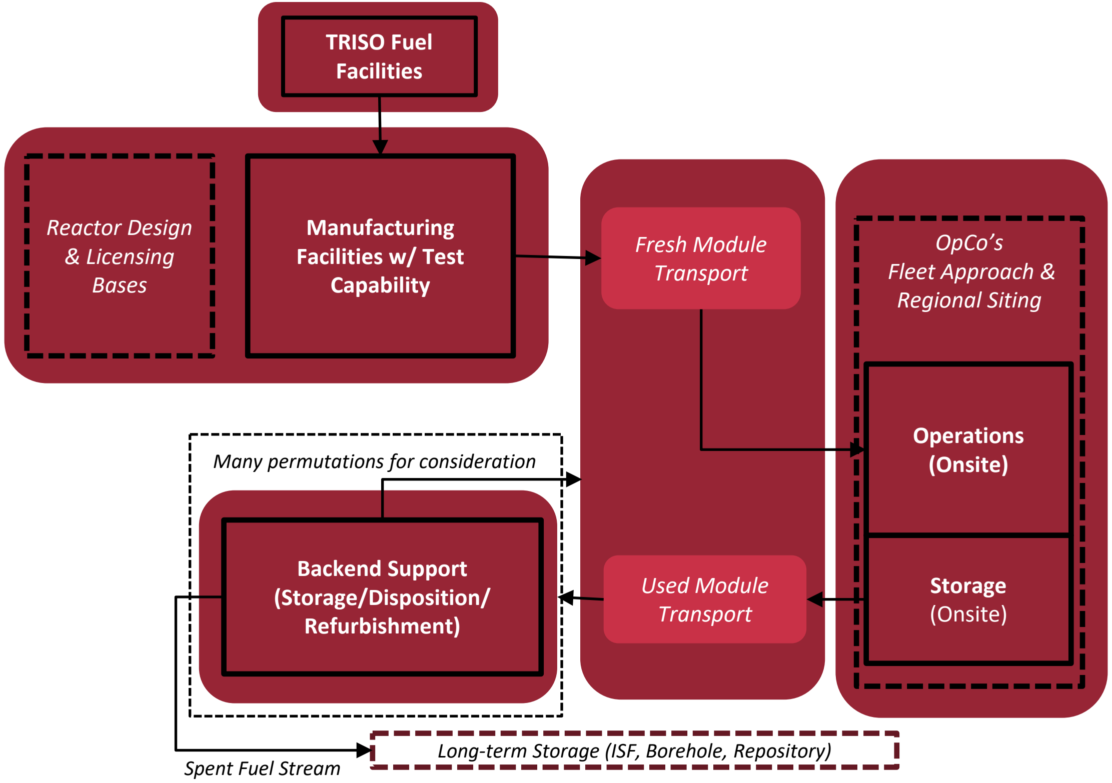
MRP Microreactor Program

Cross-Domain Development of Principal Design Criteria for Transportable Reactors

Travis A. Chapman
Director, Regulatory & Licensing
December 5, 2024
2024 Workshop on Storage & Transportation of TRISO & Metal Spent Nuclear Fuels

Transportable Reactor Technologies Cross Many Operational Domains

- Transportable microreactor business models incorporate a full lifecycle of activities from manufacture through end-of-life: regulator has multiple touchpoints. Regulatory framework addresses all domains
- Each element represents a potential “mode” for the microreactor and intersection with the regulatory framework
- Commercial success for any concept will depend on effective solutions to each element and their transitions



Principal Design Criteria: Tool for Regulatory Acceptance

Principal Design Criteria (PDC) are a key aspect of reactor-related safety review activities under 10 CFR 50 and 10 CFR 52 frameworks

- (10 CFR 50.34(a)) Applicants provide the PDC for the facility, the design bases and the relation of the two
- (10 CFR 50 App. A Introduction) *The [PDC] establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, [SSCs] that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public*

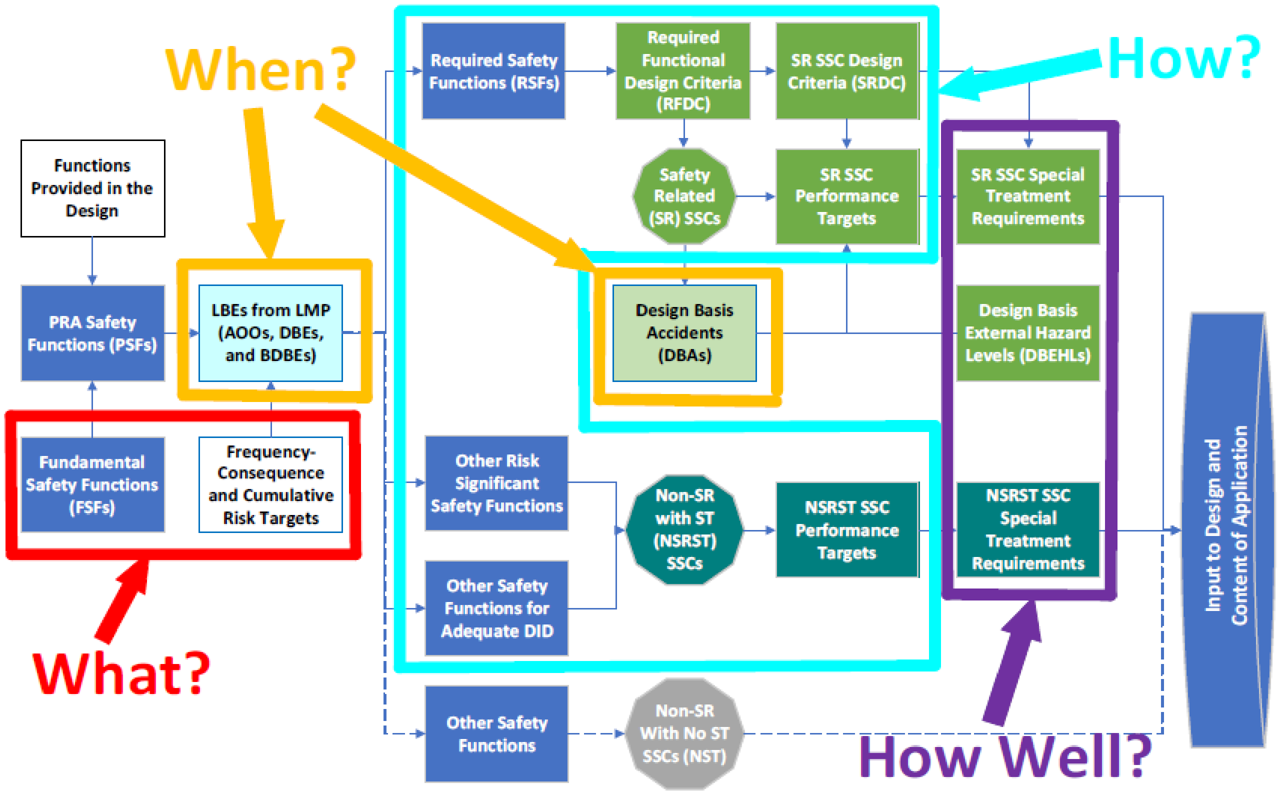
PDC represent design requirements used for two different purposes:

- Developers: PDC are informed by & inform system engineering-focused design requirements. Focus on design verification activities (i.e., NQA-1 Design Control)
- Regulators: PDC define acceptance criteria and conformance is a means of demonstrating safety

Similar design criteria exist, in form, in the regulations & regulatory guidance associated with spent fuel storage (NUREG-2215) and transport (NUREG-2216) activities

PDC as acceptance criteria represent a way for stakeholders to understand what is acceptable and guide how a demonstration of safety is developed, packaged for review, achieves acceptable evaluation findings, and is maintained through the technology lifecycle

PDC development from a Licensing Modernization Project perspective



Licensing Modernization Project (LM) guidance in NEI 18-04 and RG 1.233 provide a method of developing functional design criteria for how safety functions are allocated to SSCs, their performance requirements established, and "targets" for capability/reliability (how/how well) are set

Special Treatments establish controls to ensure functional design criteria remain met and monitored through the lifecycle (e.g., quality controls, surveillances/inspections, technical standards)

Approach is often combined with other regulatory guidance on PDC development (RG 1.232)

Source: ML21173A262 TICAP Presentation, Southern Company

PDC from a Storage & Transportation perspective

NUREG-2215 *Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities*

Section 3: Principal Design Criteria Evaluation

- Criteria associated with structural, shielding, confinement, radiation protection, criticality, material compatibility, retrievability, decommissioning
- Addresses a range of normal conditions, off-normal conditions, and accident conditions associated with storage activities (both facility and cask/canister systems)

NUREG-2216 *Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material*

- By-section, Regulatory Requirements and Acceptance Criteria & related guidance (e.g., RG 7.6)

  The package must have adequate structural performance to meet the containment, shielding, subcriticality, and temperature requirements of 10 CFR Part 71 under normal conditions of transport, hypothetical accident conditions, and air transport conditions, as applicable

- Feature-specific requirements associated with tamper-indication, positive closure, lifting/tie-down, heat load/thermal limits, and others

Can be translated into discrete functional design requirements and performance targets (what, when, how, how well) for spent fuel-related SSCs of interest: particle coatings, canisters, vaults, etc.

Challenges to Address

- Review approach: Discrete applications for each domain vs. integrated application for all domains
  - Economic consideration for applicants to manage development of application content (Safety Analysis Reports, Tech Specs, etc.), NRC review resources, schedules, etc.
  - SARs for Utilization Facilities vs. Certificates of Compliance have different information organizing principles
- Duration of “Licensed” activity scope
  - For operating reactors, applicability of PDCs align with the duration of the License (40+20+20+etc.)
  - Addressed in license renewal/subsequent license renewal with tools like aging management programs
  - Long-term storage (in reactor, canister, cask, etc.) should address applicable durations of interest
- Technology approach to storage & transport activities
  - For transportable reactors, many approaches are possible: reactor as a package, reactor becomes a package, or the reactor is contained within packaging
- Differences in evaluation approaches (i.e., PDC as Figures of Merit)
  - Transportation: NCT and HAC – deterministic, bounding, conservative
  - Site Operation: Risk-informed methods of establishing the range of licensing events to evaluate

BWXT

System Design and Safety Analysis Associated with Storage and Transportation

Prakash Narayanan

2024 Workshop on

STORAGE AND TRANSPORTATION OF TRISO AND METAL SPENT NUCLEAR FUELS

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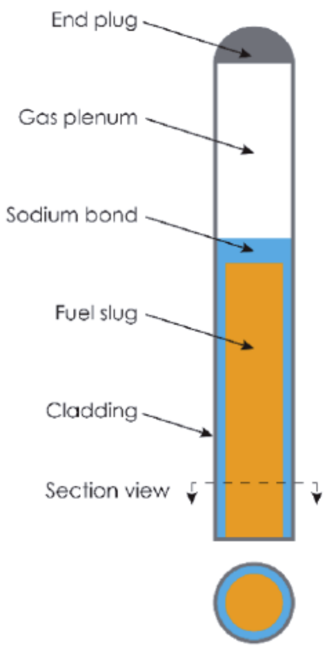
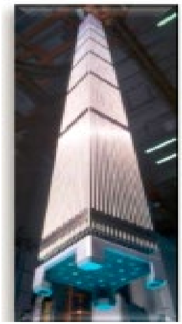
Overview of presentation

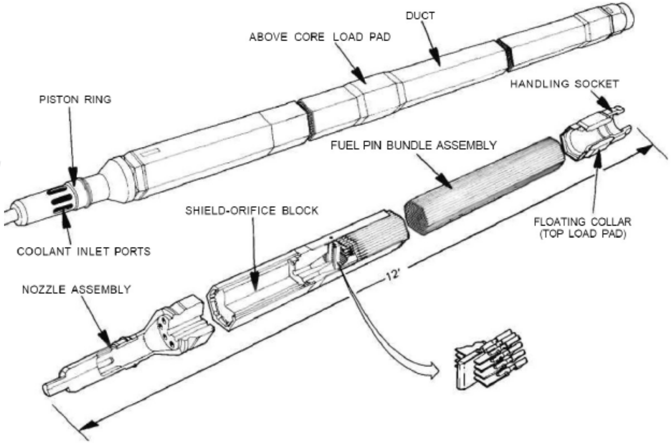
1. TRISO and Metallic Fuel
2. Storage and Transportation
3. Safety Analysis Criteria
4. Impact on System Design
5. Key Takeaways

1.1 TRISO and Metallic Fuel Types

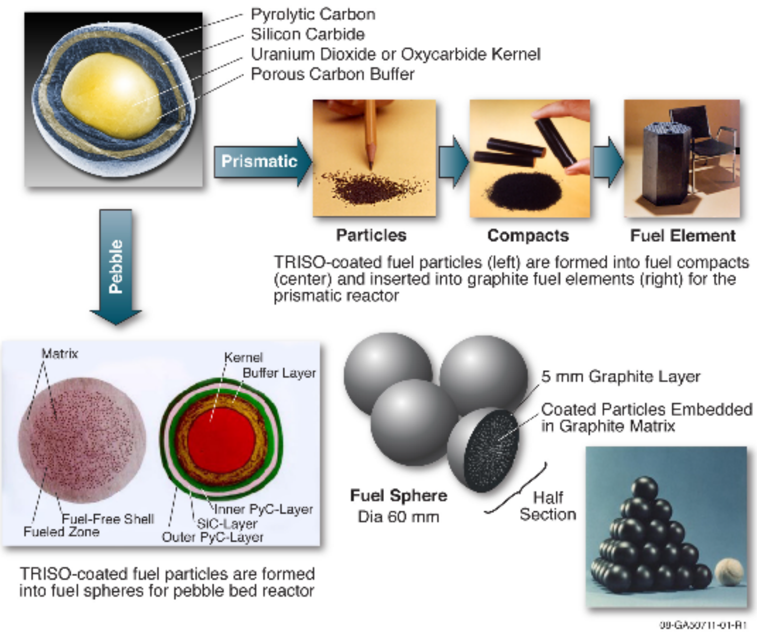
Fuel with cladding







TRISO fuel



TRISO-coated fuel particles (left) are formed into fuel compacts (center) and inserted into graphite fuel elements (right) for the prismatic reactor

TRISO-coated fuel particles are formed into fuel spheres for pebble bed reactor

1.2 Fuel Classification for Safety Analysis

Used Fuel Management approach based on Fuel Type and Reactor operational features

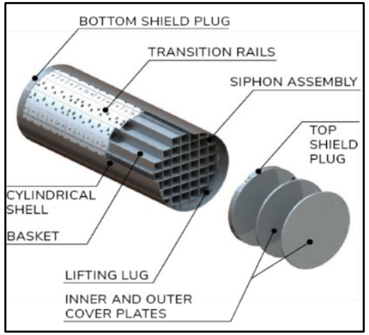
- **LWR Type fuel**
- **TRISO fuels**
- **Metallic fuels**

- **Decay Power density**
  - **Standard, Low & High**
- **Fission Product Barrier**
- **Refueling Interval**

2.1 Storage and Transportation

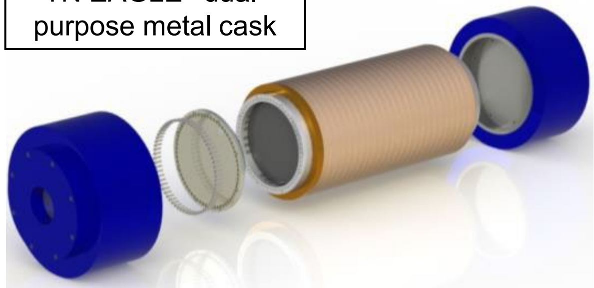
- **Storage and Transportation are critical part of the overall backend management**
- **Licensed solutions must be available for the various fuel types**
- **Existing solutions like EOS/EAGLE systems, licensed for LWR fuel, can be adapted/modified for such needs**

EOS ® Canister and Basket



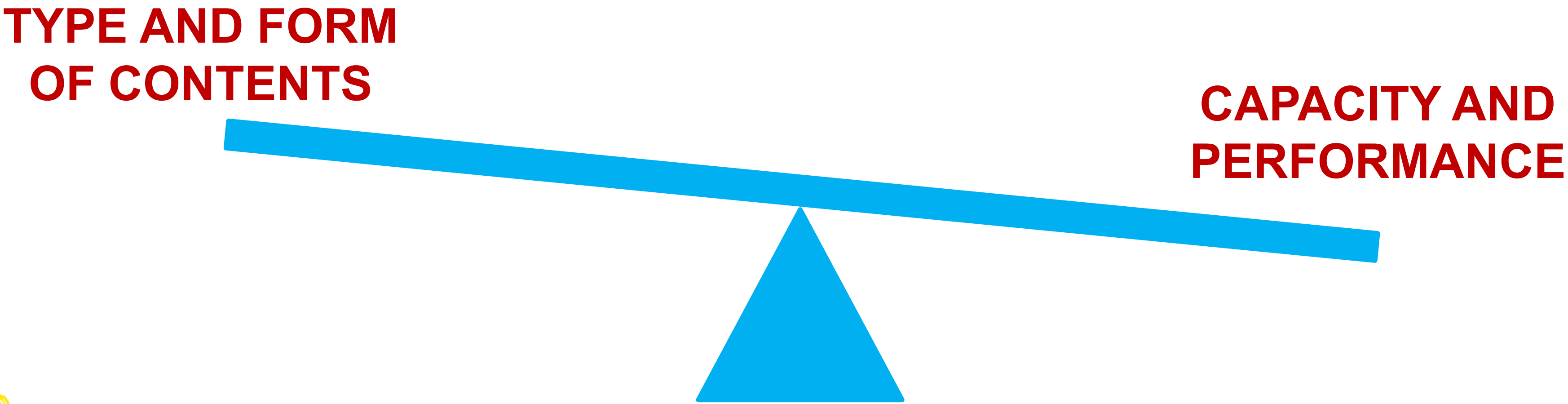
EOS ® HSM

TN EAGLE® dual-purpose metal cask



2.2 Readiness of Existing Systems

- **Storage & Transportation Systems have evolved over 40+ Years**
- **Adaptation is key for Storage and Transportation**
- **Significant Operational Experience including Aging Management**



orano

3.1 Safety Analysis Criteria

- **Requirements for Storage and Transportation Systems are both System and Contents dependent**
- **Depending on the Fuel Types, System Design could essentially remain unchanged**

Contents

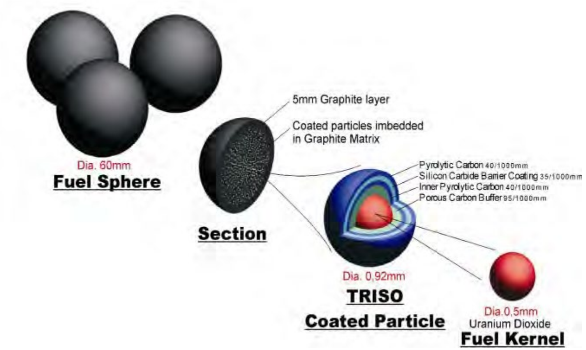
- **Fission Product Barrier**
- **Criticality Control**
- **Fuel Assembly Design**
  - **Type, Form, Weight**
  - **Decay Heat, Source Terms**

System

- **Physical Protection**
- **Containment / Confinement**
- **Heat Removal**
- **Radiation Protection**

3.2 Fission Product Barrier

- **Significant experience with LWR cladding integrity supplemented by HBU Demo Project**
- **Will require subject matter expertise to develop fuel / cladding integrity criteria**
- **TRISO-Type fuel is robust, will retain fission gas and fission products**
- **Canister-based retrievability maybe employed**
- **Regulatory guidance will need to be updated**



Fuel and/or Cladding integrity is critical for modeling

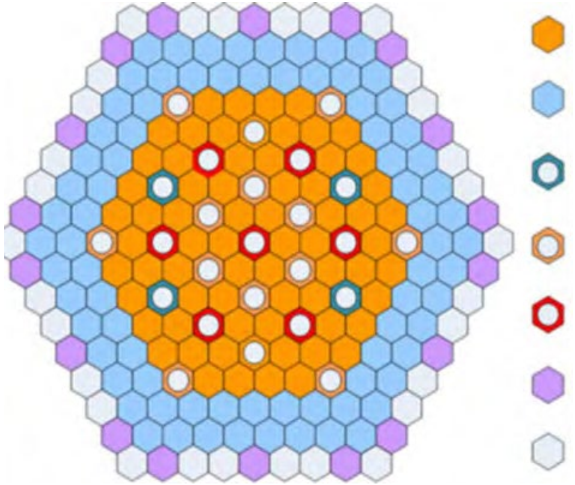
3.3 Criticality Control

- **Significant experience on criticality analysis - all types of designs with fresh fuel assumption**
- **Benchmark validations will be required for burnup or irradiation credit**
- **Fixed poison effectiveness different from LWR fuel**
- **Some reactor designs do not have water**
- **Low power density fuel will not need fixed poison**
- **Regulatory guidance will need to be updated**

Criticality Control will impact capacity optimization

3.4 Fuel Assembly Design

- **Weight, Dimensions, Fuel Type, Decay Heat, Source Terms etc**
- **Benchmark validations will be required**
- **High power density fuel will be capacity / geometry limited**
- **Low power density fuel will be mass limited – low fuel to non-fuel ratio**
- **Regulatory guidance will need to be updated**



Decay Heat will be a limiting factor for System Performance

4. Impact on System Design

Features	Current Generation	High Power Density	Low Power Density
Design Changes	None for Standard	Basket Layout Fuel Segment	Basket Layout
Criticality Control	Fixed Absorber, Burnup Credit	Basket Geometry, Fixed Absorber	Basket Geometry
Capacity	37 PWR 89 BWR	Variable	Limited by weight
Heat Load	~ 50 kW / DSC	> 50 kW / DSC	~ 30 kW Loaded
Cooling Time	> 2 Years	> 4 Years	> 45 Days
Dose Rates	No Change	Small Increase	Decrease

5. Key Takeaways

- **Safety Analysis for Storage and Transportation of Metallic and TRISO fuels can mostly be accomplished within the current framework**
- **Fuel and Contents related criteria will require Subject Matter Expertise which will lead to regulatory guidance**
- **The experience from current generation storage and transportation systems is immensely beneficial and applicable**
- **Regulatory changes may include classification of fuel including gross ruptures and cooling time**

QUESTIONS

COMMENTS

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Giving nuclear energy its full value

System Design and Safety Analysis Associated with Storage and Transportation

Prakash Narayanan

2024 Workshop on

STORAGE AND TRANSPORTATION OF TRISO AND METAL SPENT NUCLEAR FUELS

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Building on Established Knowledge to Inform the Regulatory Framework for TRISO and Metal Spent Nuclear Fuels

Rod McCullum, Nuclear Energy Institute
2024 NRC Workshop on Storage and Transportation of TRISO and Metal Spent Nuclear Fuels 12/5/2024

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Westinghouse

X energy

ZENO POWER SYSTEMS

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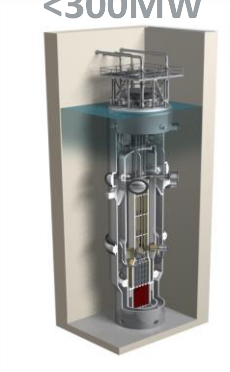
Expanding Versatility through Advanced Technology

Micro Reactors (< 20MW)

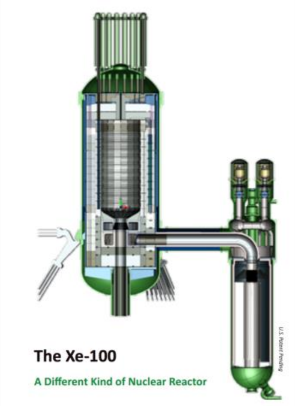
Oklo (shown)
Approximately a dozen in development

LWR SMRs <300MW



NuScale (shown)
GEH X-300
Holtec SMR-160

High Temp Gas Reactors

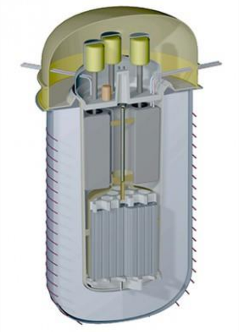


X-energy (shown)
Several in development

Liquid Metal Reactors

TerraPower Natrium (shown)
Several in development

Molten Salt Reactors



Terrestrial (shown)
Several in development

Non-Water Cooled
Most <300MW, some as large as 1,000 MW

NIA Technology Primer: https://nuclearinnovationalliance.org/sites/default/files/2022-07/ANRT-APrimer-July2022.pdf

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Advanced Reactor Progress

NEI

	Pre-Application	Under Review	Under Construction	Operating
Test Reactors		Kairos Power Hermes2; Natura Resources SUSTAINABLE ENERGY	Kairos Power Hermes	
Micro Reactors	Westinghouse; USNC ULTRA•SAFE•NUCLEAR; OKLO; GENERAL ATOMICS		BWXT BWX Technologies, Inc.	
Power Reactors	TERRESTRIAL ENERGY; GE HITACHI; DEEP FISSION; GENERAL ATOMICS; Westinghouse; HOLTEC INTERNATIONAL; arc CLEAN TECHNOLOGY; X energy	TerraPower; NUSCALE Power for all humankind		

Progress Being Made

Updated 09/25/2024

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Established Knowledge on SMR Spent Fuel

- **DOE BEMAR***

  Back End Management of Advanced Reactors

  (Activities driven by legal requirement for operators to have spent fuel acceptance contracts with the U.S. Department of Energy as a prerequisite to obtaining an operating license with the U.S. Nuclear Regulatory Commission)

- **ARPA-E UPWARDS**

  Universal Performance Criteria and Canister for Advanced Reactor Waste Form Acceptance in Borehole and Mined Repositories Considering Design Safety

- **Developer Specific Initiatives (US)**

- **International Initiatives**
  - **IAEA COGS** Challenges, Opportunities, and Gaps for Small reactors
  - **OECD/NEI WISARD** Waste Integration System for Advanced Reactor Designs

DOE BEMAR*

- Focused on Three Advanced Reactor Designs

Company	Reactor	Fuel Type (Composition)	Coolant	Construction Application	Estimated Operation
Kairos Power	Hermes 1 Hermes 2	TRISO Pebbles (UCO)	Molten fluoride salt	Approved Dec. 2023 Submitted July 2023	2026
X-energy	Xe-100	TRISO Pebbles (UCO)	Helium	Expected in 2024	2028
TerraPower	Natrium	Metallic with bonded sodium (U-10Zr)	Sodium	Submitted March 2024	2028

U.S. DEPARTMENT OF ENERGY | Office of NUCLEAR ENERGY | CONSENT-BASED SITING

- Analyze generic SNF types to answer the following
  - Can the SNF type be disposed of in a generic repository similar to past U.S. SNF/HLW design concepts?
  - If not, what type of treatment is needed to dispose of it?
  - How long it is going to take to treat it?
  - What are the disposal costs (including treatment, if necessary)?

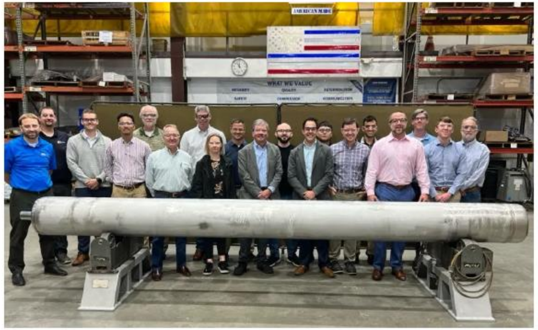
APRA-E UPWARDS* Project

DEEP ISOLATION®

SOLUTIONS TECHNOLOGY COMMUNITY ABOUT INSIGHTS NEWS & EVENTS

Deep Isolation Leads Third Technical Workshop for UPWARDS Project on Universal Canister System Development

Berkeley, CA – Deep Isolation, a leader in nuclear waste disposal solutions, successfully hosted its third technical workshop for the UPWARDS project, a project centered on the development of a Universal Canister System (UCS). The workshop, held at R-V Industries, Inc. in Honey Brook, Pennsylvania, focused on large-scale manufacturing and commercialization of this pioneering waste management solution.



The UPWARDS project, funded by the Department of Energy's Advanced Research Project Agency – Energy (ARPA-E), is developing the UCS and associated Waste Acceptance Criteria (WAC) for advanced reactor waste forms. Deep Isolation leads this initiative in collaboration with partners NAC International, Inc., the University of California, Berkeley, and Lawrence Berkeley National Laboratory.

Attendees of the workshop included members of the project's Technical Advisory Committee, who toured R-V Industries' facility and observed the UCS prototype canister in its final stages of fabrication. The committee includes representatives from across the nuclear industry who provide ongoing technical and strategic guidance for the project.

Rod McCullum, Senior Director of Decommissioning and Used Fuel at the Nuclear Energy Institute (NEI) and a member of the UPWARDS Technical Advisory Committee, emphasized the importance of the UCS initiative: "The Universal Canister System represents a pivotal step toward safe and sustainable nuclear waste management. This innovative approach not only reflects our commitment to the long-term safety of nuclear technologies that will benefit future generations but also reaffirms the nuclear industry's commitment to transparency and community engagement."

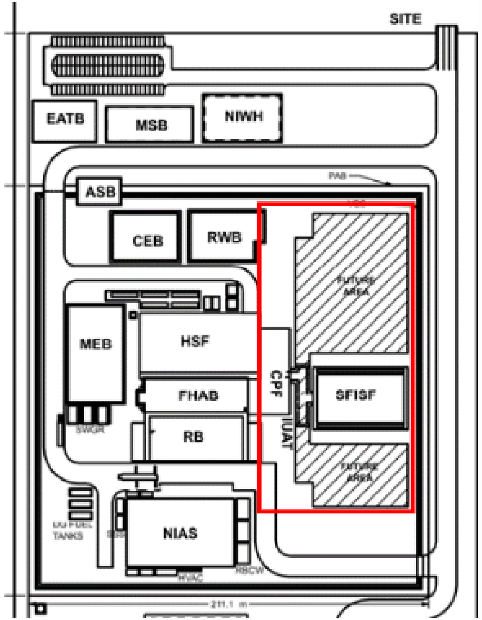
Advanced Reactor Waste Streams Considered for UCS

- SMR PWR Assemblies
- TRISO Fuel Blocks
- TRISO Compacts/Pebbles
- Vitrified Recycling Waste
- Molten Salt Waste

Developer Specific Example – X-Energy

Spent Fuel Area Overview

- All structures have been moved to above grade structures for the Long Mott site.
- Single Canister Processing Facility (CPF) Servicing all 4 units adjacent to the Fuel Handling Annex Building.
- Initially only one Spent Fuel Intermediate Storage Facility (SFISF) will be constructed with room on the Nuclear Island left for an additional 3 standalone SFISFs.
- Combined, all 4 SFISFs can store 60 years' worth of spent fuel in addition to irradiated fuel free graphite pebbles.
- The CPF is connected to the SFISF though the Inter-unit Access Tunnel (IUAT) which will be constructed progressively as additional SFISFs are constructed.
- Fuel from the reactors will be combined in storage cannisters and approximately 1 canister per week will be transferred from the CPF to the SFISF during equilibrium operations with higher transfer rates during certain operational modes.
- Plans for packaging and shipment offsite have not been finalized but are expected to include use of CPF for loading one or more spent fuel canisters into overpacks for shipment offsite.



Recycling will likely be a key part of U.S. SMR used fuel management strategy

NEI

Values of Recovered Uranium from HALEU Used Nuclear Fuels, Revision 1

Nuclear Fuel Cycle and Supply Chain

Prepared for
U.S. Department of Energy
Systems Analysis and Integration

T. K. Kim, N. Stauff, A. Abdelhameed, E. Hoffman (ANL)
A. Cuadra, C. Lu (BNL)
E. Davidson (ORNL)

March 28, 2024
ANL/NSE-23/77, Rev.1

Executive Summary

The value of the recovered uranium (RU) from high assay low-enriched uranium (HALEU) used nuclear fuels was evaluated. Three utilizations of the recovered uranium were considered in this study, which include the cases that RU is used as a fissile material of nuclear fuel, RU is reused in the original advanced reactor after reenrichment, and RU is reused in conventional light water reactors after down-blending. In this study, the RU values were identified by comparing the cost of making a unit mass of fuel with RU versus the fuel cost with the equivalent fresh enriched uranium (EU). A series of bounding analyses for calculating the fuel costs were conducted using several selected reactor types, which include microreactors, advanced thermal reactors, and fast reactors having a burnup of 2 – 165 GWd/t (with residual U-235 content in discharged fuels of 0.8 - 19.6%).

This study concludes that RU having a residual U-235 content higher than ~7% would cost less than the fresh EU. The affordability increases as the residual U-235 content in RU increases. For instance, the fuel cost with RU having the residual U-235 content of 19.6% is about 85% cheaper than the fuel cost with the equivalent fresh EU. This study observed that reusing RU after reenrichment in the original microreactor is impractical because the U-235 content in the re-enriched RU fuel would need to be higher than the limit for low-enriched uranium (<20%) to provide the same burnup performance due to parasitic absorption from U-236.

It is noted that this study focused on the recovery of uranium only, and the value of other fissile materials (such as Pu) in the used nuclear fuel was not considered even though those are bred significantly in fast reactors. In addition, the impacts of uncertainties in the cost data and the value of RU of TRISO fuels were not evaluated in this study due to the limited information on the cost data uncertainties and the separation cost from TRISO fuels.

IAEA COGS

Coordinated Research Project on *Challenges, Gaps and Opportunities for Managing Spent Fuel from SMRs*

MAIN OUTPUTS

- Development of ***specific roadmaps*** for **managing spent fuel from the different SMR technologies**, identifying what can be derived, optimized or adapted from existing practices, or what needs to be fully developed
- To compare various SMR systems, in terms of efforts required to develop and implement an SFM strategy
  - Nuclear fuel cycle facilities
  - Nuclear materials involved
  - Infrastructures (e.g., human resources, financing)
  - R&D / Demonstrations
  - Technology readiness level
  - Enablers/Synergies

OECD/NEA WISARD

The WISARD Joint Project will consist of six Tasks spread over three years as shown in Figure 1.

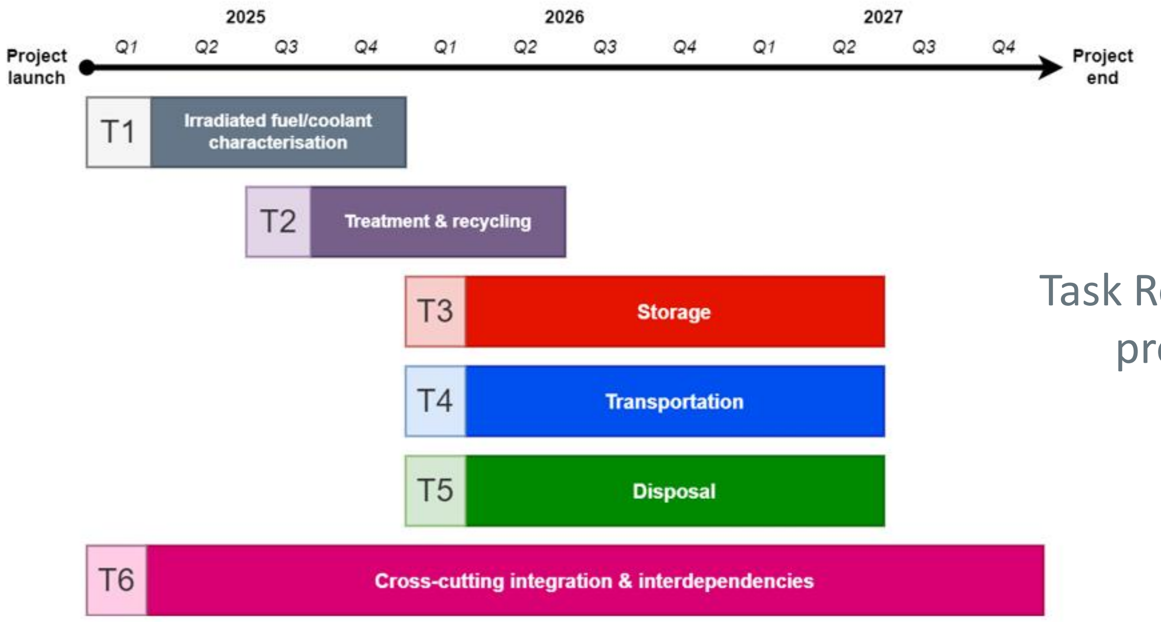


Figure 1: WISARD Task Timeline

Task Results will be proprietary

Conclusion

- Considerable work is already underway on all aspects of advanced reactor used fuel (including TRISO and Metal Spent Nuclear Fuels)
- All of this work is being conducted with the objective that the storage and transportation of these spent fuels will be conducted under existing regulations
- If there is a need for any new regulatory tools, these ongoing programs are sufficient to identify that need it in a timely manner

