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 (UO_2/UC/UC_2) 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 SiO2.

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.

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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.

Session Name: Session 10

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.