**Material Studies of Metallic Fuels**

Uranium-10Zr (wt. %) is planned as nuclear fuel in several sodium-cooled fast reactors (SFRs) currently being designed as it has high thermal conductivity, high fission atom density, and ease of fabrication. Efforts toward developing advanced metallic fuel are being taken, with focus on reducing the fabrication cost, and increasing the fuel burnup and residence time. After surveying many options for commercial fabrication of alloy rods, hot extrusion is found to be a cost-efficient and high-throughput fabrication method. Understanding the microstructure and phase evolutions as a result of irradiation and thermal cycling is of paramount importance for the development and ultimately licensing of this fuel form. Studying U-Zr alloys with higher Zr compositions is also important due to the Zr-rich fuel center region resulted from U-Zr redistribution. This talk will discuss the evolutions of microstructure and phase of several U-Zr alloys with different compositions, obtained from the time-of-flight thermal cycling neutron diffraction data, with a focus on lattice parameter, atom ordering, texture, and weight fractions. This talk will also discuss the lanthanide-induced fuel-cladding chemical interaction (FCCI) effect, which is the primary effect limiting fuel burnup and residence time. Unlike UO2 where lanthanides are stabilized with oxygen, the lanthanides in U-Zr fuels are free atoms and thus migrate along the gradients of temperature and concentration. The presenter and collaborators elucidated the FCCI mechanism and developed the state-of-the-art FCCI mitigation method. In addition, radionuclide release in the primary sodium coolant, due to fuel and cladding failures, is a concern associated with the development and deployment of SFRs. Tellurium as one of the main radionuclides released from the fuel fragmentations, as well as an active corrosive agent, has been detected in the primary sodium coolant. Isothermal corrosion investigations therefore were carried out to study the Te-induced corrosion effects on structural materials.



Bio:

**Dr. Yi XIE** is an Assistant Professor in the School of Nuclear Engineering at Purdue University. Prior to joining Purdue in 2020, she was the inaugural Glenn T. Seaborg Distinguished Post-Doctoral Fellow and then a Staff working at the Materials and Fuels Complex of Idaho National Laboratory. She has participated multiple research projects and has extensive experience of experimental platform development. Dr. Xie’s research areas include advanced nuclear fuels and material degradation behaviors in extreme environments. She investigates steam corrosion, electrochemical corrosion, and stress corrosion cracking of nuclear-grade materials. She develops advanced manufacturing technologies and conducts characterization and testing for metallic and ceramic fuels. Dr. Xie is the Virtual Labs Faculty Fellow and Paul Zmola Scholar at Purdue University. She has more than 50 peer-reviewed journal articles and conference papers. She is a member of American Nuclear Society (ANS) and The Minerals Metals and Materials society (TMS). Dr. Xie is a member of the Program Committee of ANS’ Materials Science & Tech Division. Dr. Xie and collaborators organize and chair two symposia about nuclear fuel and additive manufacturing in the TMS 2023 Meeting. She is an active technical reviewer for 10+ journals, U.S. DOE CINR/SBIR/RTE programs, and U.S. NRC UNLP program. She obtained Ph.D. degree of Nuclear Engineering from The Ohio State University and Bachelor degree from University of Science and Technology of China.