INL/RPT-22-69421

University Contributions to the Versatile Test Reactor (VTR) – FY-2022

September 2022

Book of Abstracts

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Students were an integral part of the work done on the Versatile Test Reactor (VTR) this fiscal year (FY). In fact, these students participated in the research, development, and deployment of technologies needed to move the design of advanced experimental vehicles forward. This document summarizes the work performed by these students in several areas, including the Extended Length Test Assembly (ELTA)-Sodium-cooled Fast Reactor (SFR); ELTA-Molten Salt Reactor (MSR); ELTA- Lead/lead-bismuth cooled Fast Reactor (LFA); ELTA-Gas-cooled Fast Reactor (GFR); ELTA-Materials (M); and Cross-Cutting Technologies areas. The author wishes to thank these students and their professor/mentors for the invaluable work they performed in FY-2022.

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ACRONYMS

AMI	absolute mixing index
ANS	American Nuclear Society
CCL	cartridge closed loop
CFD	computational fluid dynamics
DCPD	direct current potential drop
DNS	direct numerical simulation
DOE	U.S. Department of Energy
DP	double-pulse
EAC	environmentally assisted cracking
EBR-II	Experimental Breeder Reactor II
ELTA-[XXX]	Extended Length Test Assembly-(SFR, MSR, LFR, GFR, M)
FPVS	fission product venting system
GCI	grid convergence index
GFR	gas-cooled fast reactor
H2TS	hierarchal two-tiered scaling
IVTR	Integral Versatile Test Reactor
LES	large eddy simulation
LFR	lead/lead-bismuth cooled fast reactor
LIBS	laser-induced breakdown spectroscopy
LOFA	loss of flow accident
LWR	light water reactor
M	materials
MCNP	Monte Carlo N-Particle
MiNES	Materials in Nuclear Energy Systems
MSLF	molten salt load frame
MSR	molten salt reactor
OSU	Oregon State University
POD	proper orthagonal decomposition
ppm	parts per million
PU	Purdue University
RDT	reactor development and technology
Re	Reynolds number
RELAP	Reactor Excursion and Leak Analysis Program

SEC	spectroelectrochemical cell
SFR	sodium-cooled fast reactor
SiC	silicon carbide
SNR	signal-to-noise ratio
TAMU	Texas A&M University
TI	turbulence intensity
TIG	tungsten inert gas
UH	University of Houston
UI	University of Idaho
UM	University of Michigan
UNM	University of New Mexico
UW	University of Wisconsin
VCU	Virginia Commonwealth University
VTR	Versatile Test Reactor

University Contributions to the Versatile Test Reactor (VTR) – FY-2022

1. EXTENDED LENGTH TEST ASSEMBLY–SODIUM-COOLED FAST REACTOR (ELTA–SFR)

1.1 Redevelopment of Vanadium Wire Sodium Oxygen Measurement Technique and Correlation to Plugging Meter for VTR Applications

Andrew Napora – University of Wisconsin–Madison (UW–Madison)

2020 American Nuclear Society (ANS) Virtual Winter Meeting

To accurately measure oxygen concentration in the Versatile Test Reactor's Sodium Fast Reactor cartridge loop, a small-footprint oxygen detection method must be correlated to existing methods. This will identify the accuracy and correlation between existing techniques (such as a plugging meter). In support of this goal, UW-Madison has undertaken a redevelopment of the vanadium wire sodium oxygen measurement technique and will correlate these results with an RDT standard plugging meter sized for smaller loops. A test facility has been built and is fully operational. Two vanadium wire measurements have been made. Further testing will occur through the range of 4-20 ppm oxygen, with oxygen concentration controlled by a cold trap and independently measured by a plugging meter. A correlation will be developed between readings from a modified RDT standard plugging meter, cold trap temperatures, and the vanadium wire measurements.

1.2 Oxygen-Getting Performance of Zirconium Granules in a Liquid Sodium Hot Trap

Andrew Napora – UW–Madison

Zirconium hot trapping has been selected as a method of oxygen control in the VTR's SFR cartridge loop. Zirconium exposed to liquid sodium coolant at temperatures over 1000°F will form zirconium oxide with free oxygen impurities, thereby purifying the coolant. While the oxidative weight gain rate constant for this phenomenon has been established for chemically polished strip, no data exists for the hot trapping performance of unalloyed zirconium granules. An existing sodium facility was modified to examine the hot trapping capability of zirconium in this form factor. For granules exposed to sodium at 1200°F, the anticipated reduction in sodium oxide concentration was not observed. This is hypothesized to result from the unintentional trapping of oxide in unheated tube stubs acting as informal cold traps, artificially increasing the sodium oxygen inventory. Further tests will increase the mass of zirconium used as getter material, sizing the hot trap to deplete all potential sources of oxygen in the loop.

2. EXTENDED LENGTH TEST ASSEMBLY–MOLTEN SALT REACTOR (ELTA–MSR)

2.1 The Effect of Impurities and Geometry on the Corrosion and Thermodynamic Behavior of Molten Salts

Robin V. Roper, University of Idaho (UI)

Molten salt systems are highly relevant to the energy industry and have become a focus of much research in recent years, particularly in nuclear energy and thermal energy storage. Many types of molten salt are used for various systems, including fluoride, chloride, and nitrate salts. These types of salts have thermodynamic and chemical properties, which make them advantageous to energy production systems. Molten salt systems are superior to other systems in many ways, including design simplification, atmospheric operating pressures, higher heat capacities, safer operational parameters, and the possibility for online salt processing. Despite the advantages of a molten salt system, these concepts can also contain more complexity than many other energy storage systems due to the chemistry of the salt mixtures. In particular, molten salt systems can be particularly susceptible to corrosion. A risk assessment was performed as part of this dissertation, determining that corrosion risk is of particular concern in designing a molten salt system.

To address the risk of corrosion in a molten salt system design, two phases of research were conducted, which are described in this work. The first phase aimed to determine the effect of corrosion on the thermodynamic behavior of molten salt. In particular, the melting behavior of two types of molten salt was investigated, with additions of impurities that could result from corrosion. The second phase of research aimed to determine the effect of system design on molten salt corrosion. Through the use of COMSOL multiphysics, four different basic geometries were modeled, which were representative of geometries that could be found in a large-scale system. Molten salt laminar flow was modeled in these geometries, with corrosion behavior coupled to the laminar flow physics. This model was the first of its kind and was simple in form. It provided information on the general impact of design changes on the velocity profile and corrosion behavior in representative geometries.

The combination of experimentation and modeling determined that design changes can be made to reduce the impact of corrosion on a molten salt system. Extending the system temperature to above the entire melting range of a molten salt will ensure an operating fluid that is entirely liquid in phase. Modifying geometry to reduce areas of peak velocity will reduce the intensity of local corrosion and reducing the overall system velocity will also reduce corrosion. Incorporating system design changes such as these in the preliminary phases of design can reduce overall corrosion risk, extending the lifetime of the system.

2.2 Review, Assessment, and Learning Lesson on How to Design a Spectroelectrochemical Experiment for the Molten Salt System

Dimitris Killinger – Virginia Commonwealth University (VCU)

This work provided a review of three techniques for molten salt medias: (1) spectrochemical; (2) electrochemical; and (3) spectroelectrochemical. A spectroelectrochemical system was designed by utilizing this information. Here, we designed a spectroelectrochemical cell (SEC) and calibrated temperature controllers and performed initial tests to explore the system's capability limit. There were several issues, and a redesign of the cell was accomplished. The modification of the design allowed us to assemble, align the system with the light sources, and successfully transfer the setup inside a controlled environment. A preliminary run was executed to obtain transmission and absorption background of $NaCl$-$CaCl_2$ salt at 600°C. It shows that the quartz cuvette has high transmittance effects across all wavelengths and there were lower transmittance effects at the lower wavelength in the molten salt media. Despite a successful initial run, the quartz vessel was mated to the inner cavity of the SEC body. Moreover, there was shearing in the patch cord, which resulted in damage to the fiber optic cable, deterioration of the SEC, corrosion in the connection of the cell body, and fiber optic damage. The next generation of the SEC should attach a high-temperature fiber optic patch cord without introducing internal mechanical stress to the patch cord body. In addition, MACOR should be used as the cell body materials to prevent corrosion of the surface and avoid the mating issue and a use of an adapter from a manufacturer that combines the free beam to a fiber optic cable should be incorporated in the future design.

3. EXTENDED LENGTH TEST ASSEMBLY–LEAD/LEAD-BISMUTH-COOLED FAST REACTOR (ELTA–LFR)

3.1 Computational Methods, Investigations, and Codes to Support Corrosion Experiments in Molten Lead and Transfer to Reactor Conditions

Khaled A. Talaat – University of New Mexico (UNM)

Lead-cooled fast reactors have many potential economic advantages over other Generation IV reactor designs due to the high boiling point of lead (~1750°C) at atmospheric pressure and excellent neutronic properties, which have made them attractive to the commercial energy sector in recent years. However, they remain hampered by challenges in cladding material compatibility with heavy liquid-metal coolant. A forced circulation loop was established at the UNM's Lobo Lead Loop to prequalify materials for VTR testing and improve the understanding of flow accelerated corrosion in a molten lead environment. Corrosion in a lead-cooled environment is understood to be predominantly a mass transfer problem that is exacerbated by erosion of protective oxide layers and lead penetration. Quantitative transfer of the out-of-pile experiments at the loop-to-reactor conditions is necessary for the durability assessment of structural materials, safety analyses (e.g., to avoid clogging due to corrosion product precipitation), and validation of the current theoretical understanding of lead induced corrosion in a reactor environment. Many empirical and physics-based models of corrosion have been introduced in the literature based on mass transport and oxide layer modeling, but with a limited out-of-pile application and no application in reactor conditions due to the difficulty of obtaining inputs to these models (e.g., oxide layer thickness and composition, transport coefficients especially in oxide layers, reactor temperature distribution and flow conditions). The present work aims to support out-of-pile experiments in the Lobo Lead Loop at UNM and lay the foundation for efforts to study corrosion in reactor conditions and transfer the out-of-pile experiments to reactor conditions, within a proposed framework. The proposed framework involves the coupling of experiments with computational fluid dynamics (CFD) simulations of the flow to enable the development of multivariate correlations, molecular dynamics simulations to estimate transport coefficients, and neutronics-thermal-hydraulics coupling for transfer of the experiments to reactor temperature distribution and flow conditions.

CFD simulations are utilized to study the flow conditions in the loop, design components to meet experimental operation targets, and examine shear stresses on the specimens to inform erosion modeling efforts, which can be pursued after the experiments are conducted. Lagrangian-Eulerian coupled simulations are used to study convective mass transport to understand how the convective properties of lead differ from other coolants, such as lead-bismuth eutectic, sodium, and water. The simulations also explore the sensitivity of convective transport to surface roughness of the specimens, temperature of lead, and mean flow velocity. It is then proposed that the coupling between neutronics and mass transport can be leveraged to monitor and study corrosion of cladding materials in reactor conditions. The simulations employed a TRIGA model that was modified to be lead-cooled and that demonstrated positive reactivity is added due to mass transfer corrosion, which removes nickel and other absorbers from the active core. This proposed approach to be made practical, however, necessitates that reactivity and neutronics contributions from other sources (e.g., temperature distribution changes and burn up) be discerned from relatively small mass transfer contributions, which necessitates advanced multiphysics coupling. A platform is developed for geometry-blind, multi-server steady-state coupling of prompt neutronics using Monte Carlo N-Particle version 6.1 (MCNP6.1) and thermal-hydraulics (OpenFOAM/STAR-CCM+), which accounts for the effects of power distribution from neutronics on heat transfer in the simulated system and accounts for effects of temperature of densities, surface expansion, and Doppler broadening of cross-sections through MAKXSF. As data on transport coefficients needed for mass transfer modeling and thermal-hydraulics simulations, particularly in oxide layers that form with varying thicknesses and compositions, remains scarce, molecular dynamics simulations were proposed as part of the framework. Non-equilibrium molecular dynamics simulations are preferred for their ability to study size effects at the nano and low-micron scales. As an effort to improve the reliability of molecular dynamics simulations, the method of Shannon entropy for convergence assessment of the fission source distribution in MCNP is adapted and introduced to molecular dynamics. Application of the approach to simulations of thermal conductivity, radiation damage, and fluid flow shows the potential for generalization to other areas of molecular dynamics.

3.2 Joining of Candidate Materials for Lead-Cooled Fast Reactors

Brandon Bohanon, Jake Noltensmeyer, and Ashley Machado – UNM

The lead-cooled fast reactor (LFR) is a fourth-generation reactor design characterized by high temperatures, neutron dosage, and a highly corrosive/erosive environment due to high flow velocities of molten lead. Therefore, materials compatibility testing must be performed to assess and qualify structural and system components that withstand this harsh environment. In this work, mechanical and microstructural properties of base and conventionally joined FeCrAl-based ferritic alloys, as well as additively manufactured alternatives before and after molten lead exposure at 500°C, are investigated. An analysis of dissimilar joints of candidate alloys was also performed.

Candidate materials include SS316L, FeCrAl alloys (e.g., MA956 and APMT), and additively manufactured AM100. Additionally, similar and dissimilar joints formed from these materials were also examined. The conclusions from these studies are as follows:

- Testing with SS316L showed that it retains good ductility even with hardening caused by welding. Testing in the Lobo Lead Loop showed that SS316L is still resistant to corrosion attacks up to 1,344 hours in 500°C molten lead flowing at 2 m/s. While the erosion in the welded SS316L sample was greater, Pb penetration into the matrix was not seen.
- While MA956 is an ideal candidate for LFRs, it is not readily weldable by conventional fusion methods such as tungsten inert gas (TIG). The agglomeration of Y_2O_3 dispersoids and the coarsening of its fine grain structure reduces the high-temperature strength. Cracking in the weld pool was extensive and porosity was found throughout the weld pool and heat affected zone with large macroscopic pores forming at the interface. Corrosion testing on welded samples showed that up to 1,344 hours in 500°C molten Pb flowing at 3 m/s, no signs of significant liquid-metal corrosion was observed. However, porosity caused by fusion welding provided an avenue for possible increased erosion rates. Operational experience with the Lobo Lead Loop, which consists mostly of MA956, shows that pores in welded pipes can lead to the formation of leaks.
- APMT has larger grains and dispersoids than MA956. Microscopy also saw porosity in its as-received state. An examination of factory TIG-welded APMT revealed lack-of-fusion defects in the form of a groove inside the piping and a channel pore through butt-welded plates. After polishing, the same cracking inside the weld pool seen in the lab-welded specimens was observed. While no adverse effects were seen during hardness testing, these cracks initiated fractures in room and elevated temperature testing. However, when compared to SS316L, the APMT welds had greater yield strength and ultimate tensile strength in all tests except at room temperature. As seen in previous studies, when exposed to molten lead, the base APMT specimens internally oxidized forming mixed oxide nodules throughout the surface. Based on corrosion results from the AM100 specimen, pre-oxidizing APMT prior to lead exposure may help reduce the formation of oxide nodules. After 672 hours, lead penetration into the material was not observed.
- AM100 showed anisotropic grain structure along with intergranular cracking and areas of partially melted powder. Porosity was spread throughout the specimen causing a density of 95.6% of theoretical values. Precipitates could also be seen at possible melt tracks, but were not characterized.

In tensile testing AM100 behaved similarly to base APMT with better ductility and strength when compared to welded APMT specimens. Pre-oxidation treatments formed a 1 µm thick alumina layer prior to testing. Even at areas of highest erosion, an alumina layer was still present and prevented lead penetration.

From the work with dissimilar joints, the most prevalent defect is widespread porosity. While the size and location of pores depended on the alloy being joined, porosity was spread throughout the weld pools of all joints made. Cracking was also common in most of the joints. Ion images showing what appeared to be un-melted powder in one of the weld pool cracks may suggest a change in filler metal. Optimization of the chamfer angle, preheat treatment, and welder current may improve issues during welding, such as lack-of-fusion and lack-of-penetration. Softening of the FeCrAl alloys was seen after post weld heat treatments. This could be caused by changes in the distribution of the dispersoids during welding or the coarsening of grains or a combination of both. Further analysis needs to be done before drawing conclusions.

4. EXTENDED LENGTH TEST ASSEMBLY–GAS-COOLED FAST REACTOR (ELTA–GFR)

4.1 Development of Innovative Measurement Techniques for Fission Product Transport Quantification

B.H. Choi – Texas A&M University (TAMU)

This research presents the methodology and results of numerical studies on turbulent flows in a piping system that resembles the geometry of a fission product venting system (FPVS). The fission product includes graphite particulates carried by coolant as it flows through the core, as well as fission gases like krypton, xenon, cesium, and iodine produced from fisson reactions. Knowing the location of mixing and the magnitude of various gaseous components is necessary to manage and mitigate the adverse effects of fission products in the cartridge loop. In this work, scaling analysis was used to identify the geometry, surrogate gases, and surrogate particles. The turbulent flow fields in the piping system were simulated using OpenFOAM v7 and large eddy simulation (LES). The grid independence was established using the grid convergence index (GCI) concept. In addition, numerical simulations were validated against the experimental data and direct numerical simulation (DNS) data reported in the literature. The quantities of interest, such as reattachment length and critical point of flows through axisymmetric expansion and the absolute mixing index (AMI) through a piping system with a 90 degree bend, were calculated. The reattachment length represents the length of the recirculation region, and the critical points (Cr_1 and Cr_2) represent the crossover points from laminar to transition region, and the transition region to turbulent flow. A series of parametric runs were made by varying flow Reynolds number (*Re*), turbulence intensity (TI) at the inlet, and surrogate gas concentration at the injection point. Using the parametric runs design, correlation expressions for reattachment length and the critical point were developed. The gradient of $\log_{10}$ AMI represents the mixing rate, and the highest value is observed downstream of the expansion valve. The AMI is the standard deviation of the concentration of the surrogate gas Argon in the piping system. The proper orthogonal decomposition (POD) technique was used to study the coherent turbulent structure downstream of a sudden expansion. Finally, simulation of the surrogate solid particles was carried out using the Lagrangian approach. The deposition velocity particles in a square horizontal channel were estimated using a discrete random walk model.

4.2 Fission Product Detection in Gas-Cooled Fast Reactors

Londrea Garrett – University of Michigan (UM)

There is significant interest in the nuclear power community in the development of Generation IV helium-cooled fast reactors due to their anticipated longer lifetimes and safer operational conditions as compared with traditional reactor systems. These reactors would greatly benefit from the ability to detect the increased presence of fission fragments such as certain Xe isotopes within the coolant stream, which can be an indication of the onset of fuel failure. A previous study demonstrated that laser-induced breakdown spectroscopy (LIBS) could detect concentrations as low as 0.2 μmol mol^{-1} parts per million (ppm) of Xe within an ambient He environment with 10^4 laser shots. Fuel failure monitoring would benefit from even better sensitivity. We investigate collinear double-pulse (DP) LIBS to improve upon the previously observed limit of detection. Measurements were performed using a 300-mJ, 10-ns Nd:YAG laser (Quantel, Evergreen) operating at 1064 nm and 10 Hz. The beam was focused using a 10-cm focal length lens into the experimental chamber containing the certified Xe-He mixtures. Emitted light was collected using a collimator and directed to an intensified charged-coupled device (ICCD) (Princeton Instruments, PI-MAX4) attached to a Czerny-Turner spectrogram (Horiba, MicroHR). The optimal gate width and gate delay for measurements were established through performing a systematic scan between 0.1 µs and 1000 µs for each parameter using the signal-to-noise ratio (SNR) as the figure of merit. Results suggest that DP-LIBS increases the sensitivity by one order of magnitude and, equivalently, decreases the required detection time by two orders of magnitude compared to SP-LIBS.

4.3 Effect of Surface Emissivity on Heat Transfer for the GFR Cartridge Loop (ELTA-GFR)

Ephraim Coelho – University of Idaho (UI)

The VTR continues to be the focus of study at UI, particularly in the development of comparative studies towards heat transfer behavior of its cartridge closed loop (CCL), which is currently under development. A fuller understanding of the behavior of silicon carbide (SiC) in the VTR test environment is crucial for safety and continuous operation of the reactor. CFD simulations have been performed previously in order to determine theoretical surface emissivity of SiC and to better understand the thermal radiation behavior in the CCL. Temperature samples collected from the test apparatus developed by UI were used to determine the experimental surface emissivity SiC.

4.4 Preliminary Design of a Thermal-Hydraulics Test Bed to Support Development of High-Temperature Gas-Cooled Cartridge Loop

Joshua Young – UI

NURETH-19

The high-temperature nuclear reactor designs are pushing the limit on components, available materials, and instrumentation. In this study, the focus is on designing a thermal-hydraulics test bed that will contain two prototypic test stands/loops: a high-pressure, high-temperature helium gas-cooled cartridge loop and a relatively high-temperature liquid-metal system. The overall objective of these two test loops is to provide a flexible testing environment to support the development of a gas-cooled fast reactor cartridge loop for the Versatile Test Reactor, a liquid-metal cooled fast-spectrum test reactor being developed by the US Department of Energy. The design of the helium prototypic test stand will allow for a gas cartridge and its components to be tested at high temperatures ranging between 650°C and 850°C and pressures up to 13.3 MPa. The liquid-metal prototypic test stand will operate near the atmospheric pressure. A comparative study with different surrogates for liquid metal is carried out, where comparisons are made with sodium to understand thermal hydraulic characteristics. The helium test stand being designed will consist of a test section, an external heater, and a cooler, as well as a helium compressor. Process modeling software is utilized to carry out a parametric study to optimize the design of the test stands and ensure sufficient flexibility for a broad spectrum of testing.

4.5 *In-Situ* Thermal Properties Measurement of SiC Composites Under Irradiation

Ken Ssennyimba – University of Houston (UH)

SiC composites are candidate materials for cladding and control rods in the GFRs because of their excellent thermal and mechanical properties in harsh environments. However, one major drawback to their application is the lack of irradiation data sufficient for the prediction of their performance up to peak temperatures (~1500°C) and neutron irradiation doses (~350 dpa). A new *in-situ* technique for rapidly determining the thermal properties of irradiated samples based on infrared thermography and finite element analysis was developed. Thermal properties of an SiC composite, irradiated with carbon up to a dose of 5.6 dpa at 300°C were studied. Results depict trends of thermal conductivity degradation in accordance with previous studies. This technique is expected to have a significant contribution to the rapid and accurate prediction of thermal properties of irradiated materials in the GFR cartridge loop.

5. EXTENDED LENGTH TEST ASSEMBLY–MATERIALS (ELTA–M)

5.1 Design, Development, and Testing of an Advanced Nuclear Reactor *In-Situ* Creep Capsule that Accommodates Multiple Specimen Geometries

Dulus Owen – Purdue University (PU)

Nuclear reactors operate under extreme environmental conditions, such as neutron bombardment, elevated temperatures, and high pressures. Over time, the harsh environmental conditions affect the material properties of structural materials and fuels. Studying the mechanical properties of structural materials and advanced fuels is common practice that is required to validate the material performance for deployment within the next generation reactors. Next generation reactors, such as Generation IV reactors, will operate in more extreme environments than the current fleet of power reactors, with temperatures reaching potentially over 1,000°C and the use of corrosive coolants, such as lead, lead-bismuth, and liquid sodium. Studying *in-situ* mechanical properties, such as irradiation creep, is challenging, particularly in next generation reactor conditions. The instruments used to measure *in-situ* irradiation creep must collect data in real-time while experiencing harsh in-reactor conditions. Many historical *in-situ* creep capsules have implemented a variety of designs to measure irradiation creep. The current study designed, developed, and tested a novel, modular *in-situ* creep capsule to address the challenges of testing candidate materials for next generation reactors. The *in-situ* creep capsule utilizes modern manufacturing methods, instrumentation, and alloying to address extreme environmental temperatures. Implementing modern technology has positioned the critical components of an *in-situ* creep capsule near the specimen, improving the accuracy of measuring irradiation creep in real-time. The modular design of the *in-situ* creep capsule allows the testing of various specimen geometries, thus making it a first-of-a-kind.

5.2 *In-Situ* Mechanical Testing in FLiNaK Molten Salts

Xavier Quintana, Jake Quincey, and Samuel Briggs – Oregon State University (OSU)

OSU has developed a molten salt load frame (MSLF) capable of performing *in-situ* fracture mechanics-based environmentally assisted crack growth testing in high-temperature molten salts. The MSLF allows for real-time load control and monitoring of crack initiation and growth via the use of the direct current potential drop (DCPD) method. The utilization of a pumped cold leg in the MSLF allows for the simulation of temperature gradient-driven mass transport effects is seen in non-isothermal salt flow loops. This work aims to elucidate the effects of stress on environmentally assisted cracking (EAC) in fluoride salts. Slow strain rate tests with simultaneous molten salt exposure have been performed on SS316L in FLiNaK. These tests are performed under simulated MSR-relevant conditions at 650°C under an Argon cover gas.

5.3 Fracture Mechanics-Based Testing and DCPD in FLiNaK

Xavier Quintana, Peter Beck, Jake Quincey, George Young, and Samuel Briggs – OSU

2021 Materials in Nuclear Energy Systems (MiNES)

Advanced reactor coolants such as molten salts pose unique challenges for test systems designed to study environmentally assisted cracking (EAC). Challenges with *in-situ* testing include immersing the sample in molten salt while preventing air exposure and providing sample access for real-time load control and monitoring of crack initiation and growth. An *in-situ* mechanical test system has been developed that addresses these challenges and is capable of state-of-the-art, fracture mechanics-based EAC testing in molten salt. Slow strain rate and K-controlled fatigue crack growth tests were performed utilizing DCPD for *in-situ* data collection. Initial tests have been conducted using 316 stainless steel in molten FLiNaK at temperatures up to 700°C. Samples were characterized post-test via scanning electron microscopy to elucidate the salt's influence on stress corrosion cracking. Inductively coupled plasma-based salt impurity measurements are also used to correlate specimen performance in the salt environment.

5.4 Materials Testing in FLiNaK

Xavier Quintana, Dustin Mangus, Jake Quincey, Julie D. Tucker, and Samuel Briggs – OSU

2023 TMS Annual Meeting

Fluoride molten salts have been targeted as a promising coolant for next generation reactors due to their favorable thermophysical properties, but present unique materials challenges primarily driven by the aggressive corrosion attack due to the reducing environment. Stress present in structural materials exacerbate degradation. Challenges with *in-situ* data collection mean data for this system is sparse and the degree of which stress accelerates degradation is not fully realized. Oregon State University's Molten Salt Load Frame (MSLF) enables environmentally assisted cracking (EAC) tests to be conducted via slow strain rate and fatigue crack growth tests. Tests are conducted with 316L stainless steel. Samples are immersed in flowing, convection driven LiF-NaF-KF eutectic (FLiNaK) simulating MSR-like conditions and are subject to variable fatigue or constant strain rate until failure. Pre- and post-immersion characterization and fractography is conducted to identify failure modes and evaluate the effects of molten salt on 316L.

6. CROSS-CUTTING TECHNOLOGIES

6.1 Quantifying System Distortions Between the VTR and IVTR Facilities through H2TS Scaling Methodology and RELAP5-3D Simulations

Juwan A. Johnson – OSU

SFRs operate at high temperatures and have a high neutron flux. This environment can be challenging; thus, research and development of small-scale and large-scale sodium facilities are needed to bridge the gap in understanding. Scaling analyses develop ratios to assist with this gap in understanding to address identified thermal hydraulic phenomena that govern SFRs. The purpose of this research is to provide justification for the development of the Integral Versatile Test Reactor (IVTR) facility, which is a sub-scaled conceptual facility that simulates the system behavior of the Versatile Test Reactor (VTR). To justify the feasibility of the IVTR, a one-dimension computational simulation was developed to simulate the governing thermal hydraulic processes expected during steady-state and transient operations using the Reactor Excursion and Leak Analysis Program (RELAP). Second, a hierarchal two-tiered scaling (H2TS) analysis was performed on the IVTR and VTR facilities to understand the important thermal hydraulic phenomena that should be analyzed for severe accident issue resolution with regards to sodium fast reactors. With the use of RELAP5-3D and H2TS scaling methodology, a comparison between the IVTR and VTR facilities were made and distortions between the two were determined. These analyses will be used to further nuclear energy research in the field of SFR development and commercialization.

6.2 Quantifying System Similarity Between EBR-II and the VTR Integral Test Facility through Hierarchical Two-Tiered Scaling and RELAP5 Simulations

Brett Leitherer – OSU

Advanced reactor designs offer many advantages over typical light water reactors (LWRs), including fast neutron capability, passive safety features, and improved performance. While the future for these reactors seems promising, operating and test experience is necessary to further validate such claims. The development of full-scale facilities has proved to be exceptionally costly and time-consuming, with some failing to ever reach completion. In order to streamline the research and development of advanced reactor concepts, the implementation of scaled-down integral test facilities is required. Constructing small-scale facilities assists in reducing required materials, build time, and overall cost. By applying scaling methodology, the scaled-down facilities can be designed to conserve significant phenomena present in the full-scale model, despite geometric differences.

The VTR is an SFR concept that is currently under development to support fast reactor research in the United States. A scaled-down integral test facility of VTR has been proposed to support such efforts through the validation of design and phenomenological behavior. With a theoretical model for the IVTR in place, further verification is necessary to confirm a working design. This study assists in identifying phenomena prominent in SFRs during a loss of flow accident (LOFA) by highlighting key nondimensional groups through the application of H2TS. Significant Π-groups were then used to quantify distortions present between IVTR and Experimental Breeder Reactor II (EBR-II). With EBR-II having undergone rigorous testing experience during normal and off-normal operating conditions, it provided a potential baseline for comparison with IVTR. Models were developed in RELAP5 to simulate transient operating conditions for IVTR and EBR-II and supply necessary parameters to quantify system distortions among facilities. It was concluded that while each test facility contained similar transient response trends, there was considerable distortion among them. This would suggest that EBR-II is not an ideal baseline model for comparison with IVTR and further work is needed.