INL/MIS-21-62159-Revision-0

FY2021 February Monthly Status Report for the VTR

March 2021

Changing the World's Energy Future

Jordi Roglans-Ribas, George Malone, Thomas Fanning, Kevan D Weaver

Idaho National Laboratory

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SOLVING ENERGY CHALLENGES THROUGH SCIENCE

Versatile Test Reactor

Program Highlights

Jordi Roglans-Ribas, Program Overview

The Program hosted the quarterly integration meeting in late February which focused on technical status for fuel fabrication, fuel supply and performance, experiments, nuclear design, and plant design, as well as discussions regarding revised Fiscal Year (FY) 2021 priorities, activities, and goals. The meeting was held virtually and was available to VTR staff at the partner laboratories. Representatives from the Nuclear Regulatory Commission (NRC) and the Office of Project Management (OPM) participated in the meeting, as did GE-Hitachi Nuclear Energy, LLC (GEH), Bechtel National, Inc. (BNI), and TerraPower (TP). Finally, the Technical Advisory Committee attended the meeting and provided a written report with observations and recommendations. The revised set of milestones discussed at the meeting were entered into the Department of Energy, Office of Nuclear Energy (DOE-NE) management tool, Program Information Collection System (PICS).

VTR began planning for the start of site characterization, including field work, with the objective to initiate field work for the preferred site in the summer. Specific scope for the capital funds that the project has received is also being identified and will be fully defined in March.

Negotiations continued for the design-build contract. A new release under the current contract with GEH will be established to cover May and June activities, after which the intent is to transfer the plant design scope to the design-build contract.

VTR completed the migration from SharePoint to the new Confluence Site. User functionality and access is consistent with the previous SharePoint Site. Since Confluence is cloud-based, reliability should be improved. The experiments area of SharePoint will be migrated in early March. To use the Confluence Site, Chrome, Microsoft Edge, or Firefox browsers must be used as Confluence is not compatible with Internet Explorer.

A compliance-based audit was performed as part of the verification activity to determine the effectiveness of the VTR project in implementing their Quality Management System (QMS). The audit concluded VTR personnel follow the approved processes to reliably and efficiently produce the expected result and the results consistently meet or exceed expectations. Additionally, the VTR process/actions are documented, compliant, understood, and consistently implemented by the appropriate personnel, specifically in the areas of configuration management (CM) and contractor assurance. The evaluation resulted in a highly effective rating for the VTR projects implementation of the current QMS. There are no follow-up activities resulting from the audit

In addition to the programmatic audit described above, VTR rolled out the new INL Lessons Learned (LL) module which will simplify the entering, mining, and extraction of LL data. This module also includes a cross-industry LL tool called iShare. iShare puts data from OpEx Share, Institute of Nuclear Power Operations (INPO), the NRC, and other pertinent sources all in one convenient tool. Additionally, managers and other end-users can subscribe to specific categories and types of shared material.

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VTR Executive Director Kemal Pasamehmetoglu presented, "Versatile Test Reactor: Missing Piece of Nuclear Energy Innovation Infrastructure in the U.S." at the Pacific Northwest National Laboratory (PNNL) Nuclear Energy Seminar Series on February 11.

VTR Plant Engineering

Engineering work continues to be focused on design risk mitigation and value engineering studies, technical maturation of key plant components, and nuclear supply chain evaluations and development.

As mentioned in January's report, Congress appropriated $2M in capital funds for FY 2021 which was allocated to plant design to bridge Planning Phase and Execution Phase activities. The capital scope is being reviewed and will be finalized in March.

Two important decisions were made regarding plant design:

- Spent nuclear fuel storage pads will be located outside the VTR protected area fence line. This will allow easier fuel transport, more effective security measures, and more efficient access to fuel processing facilities needed to treat the spent fuel for final storage.
- The M21 Primary Sodium Purification System was designed as an ex-vessel system outside of the head-access area (HAA). This decision reduces project risks associated with low technical maturation level that can cause cost and schedule impacts to the project, utilizes a known design with years of operating experience (OE), and eliminates or improves risks associated with new methodologies for maintenance and operation of the system. This decision was effective immediately and is documented in the minutes of the Special Purpose Review (SPR) held February 26, 2021.

Nuclear Design

Work continued in fuel performance and fabrication, core design, safety analysis and safety basis priority activities. Highlights include preparation of a report on the current assessment of Pu feedstock options, which was supported by Los Alamos National Laboratory (LANL) with options for the use of the PF-4 facility for VTR, and by Savannah River National Laboratory (SRNL) with a summary input on Pu polishing options. TerraPower competed the HT9 Supplier Qualification Plan, drawing on experience from earlier work. In the area of fuel performance, a technical report was prepared on the sensitivity and uncertainty quantification of the fuel performance based on fuel dimensions and reactor operating conditions.

The initial testing campaign was completed for the Pressure drop Experimental Loop for Investigations of Core Assembly in advanced Nuclear reactors (PELICAN), which included commissioning, facility characterization to assesses the operational capabilities of the control and acquisition systems.

In the transient analysis area, additional model development work was started to update the B24 secondary system following GEH's response to a design information request.

Preparations for a Technology Inclusive Content of Applications (TICAP) tabletop exercise in March continued with the completion of the tabletop report, which includes draft SAR chapters and an explanation of how the content of the chapters was developed and selected.

VTR Experiments

Progress continued within the VTR experiments team in development, design, and testing for the main experiment vehicle types. Progress was made in the development of the combined fuel and experiments time-motion study. Several meetings were held to clarify the expectations of how experiments can be moved through the facility, from receipt of experiments to discharge. Results of the study will help guide VTR's preliminary design VTR.

Upcoming Events:

National Academies of Sciences, Engineering and Medicine April Meeting, April 6, 2021

VTR Quarterly Integration Meeting, May 25, 2021 (tentative)

Experiments Integration Virtual Meeting, June 8 - 9, 2021

2021 ANS Virtual Annual Meeting, June 14 - 16, 2021

NURETH-19, 19th International Meeting on Nuclear Reactor Thermal Hydraulics, March 6 - 11, 2022, Brussels, Belgium

IAEA International Conference on Fast Reactors and Related Fuel Cycles (FR21), April 25 – 28, 2022

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Versatile Test Reactor

Technical Highlights

George Malone, Reactor Technical Integration

GE-Hitachi Nuclear Energy, LLC (GEH)/Bechtel National Incorporated (BNI) Design Engineering Support

Completed efforts to plan and schedule Technology Maturation Planning for the period of February 2–28, 2021; additionally, GEH completed efforts to plan and schedule Release 5.2 activities for March 1–April 30, 2021.

Stopped the piping and instrumentation diagram (P&ID), equipment arrangement, and engineering database work in AVEVA based on FY 2021 funding.

Continued work on VTR risk reduction efforts including:

- Analysis-Design Interface discussions. Initiative is aimed to increase of the integration of analysis works performed by Battelle Energy Alliance LLC (BEA) with the design work performed by GEH-BNI- TP. The objective is to ensure needed inputs and outputs are timely scheduled and planned, and interfaces activities and impacts on ongoing work in various organization are well understood. This initiative will streamline work and reduce iterations and rework between analysis and design teams. The short-term goal is to capture key interfaces activities within the L3 Master Control Schedule (MCS) schedule. The long-term goal is to facilitate implementation of a fully integrated schedule among all organizations performing work for the VTR project.
- Developed a tracker to provide the status of BEA comments provided GEH, BNI and TP deliverables to help identify the remaining open items.
- Overall Plant Requirements and Overall Plant Design Specification are in use. Provided BEA a decomposed list of ROGS for input into the Doors Next Generation (DNG) requirement tool. Completed review of first customer change set.
- Continued to update the framework for the Technology Readiness Assessment (TRA) Program and to review this framework with BEA to progress the overall program approach and to support low Technology Readiness Level (TRL) component progression.
- Continued work on Technology Maturation Plan (TMP) samples for the identified four long lead items: B11 Reactor Module, B12 Control Rod Drive Mechanism (CRDM), B21 Electromagnetic (EM) Pump, and F42 In-Vessel Transfer Machine (IVTM).
- Performed additional review of the Critical Technology Elements (CTEs) with BEA in support of the TRA program. Additional CTE were identified and added to the existing list. The additional CTEs were screened to identify the Technology Readiness characteristics that needs to be addressed in the next steps of the TMPs. Completed the TRL-CTE Screening Status Report.
- Began work on subsurface investigation subcontract development.

- Completed M21 Primary Sodium Purification System (PSPS) External Conceptual Design and Risk Assessment.
- Completed SPR of M21 PSPS to select conceptual design for VTR. Option 3, external system outside of the HAA, was selected based on technical maturity, OE, and potential synergies with Natrium as the design is similar to what Natrium will use.
- Continued Argonne National Laboratory (ANL)-GEH interfacing work activities between J11 (Core and Fuel Services) and B11 (Reactor Module System). This effort provides a key framework for collaboration to facilitate critical interfaces definition, coordination, and risk reduction.
- Continued resolution of action items resulting from the SPR of the B24 Heat Rejection System; continued work on Draft B24 Pump Study.
- Continued approach for the time-motion study to address the overall refueling and experimental handling strategies. The proposed approach approved by BEA will incorporate the recommendations from multiple time-motion analysis iterations and reevaluate the conclusions given the fully integrated refueling and experimental handling processes. Given the output of this work, the VTR program and the refueling systems will have a firm technical basis leading into Preliminary Design.

Argonne National Laboratory

Provided technical subject matter expert support for the following technical reviews and meetings:

- VTR Primary Sodium Processing System Risk Assessment,
- Reactor Facility (U11) Conceptual Design Optimization Study,
- VTR M43 Catch Pan and Tank Sizing Study,
- Alternate Ex-Vessel Configuration for the Primary Sodium Processing System,
- VTR Experiment Vehicle Optioneering Study,
- VTR B24 Pump Optioneering Study,
- VTR Cask Concept,
- VTR CRDM Assembly TMP Sample,
- VTR Reactor Module TMP Sample,
- VTR In-Vessel Fuel Transfer Machine TMP Sample,
- VTR ASME Section III, Division 5 High Temperature Reactor Design Strategy,
- EM Pump Power Supply (B21) Optioneering Study Report, and
- VTR IVTM Sodium Wetting Study.

- Participated in the VTR Primary Sodium Processing System Risk Assessment "M21 Assessment" SPR.
- Participated in a meeting with BNI, GEH, and Idaho National Laboratory (INL) to discuss challenges in applying the International Building Code (IBC) to the Reactor Facility and requesting exceptions from the IBC from the Authority Having Jurisdiction (AHJ).

Updated TEV-3727 for new orificing strategy, designed conceptual driver receptacles for three distinct flow rates, designed conceptual shield receptacle for storage of spent driver subassemblies, submitted receptacle drawings to GEH for review, finalized receptacle designs following GEH feedback process, and using reference materials provided by ANL, updated the conceptual design of driver assembly to a final design. Finalized design of shield and reflector subassemblies.

Fast Flux Test Facility (FFTF) Documentation and Data Recovery

Located and retrieved all thermal striping documents for the FFTF and the Clinch River Breeder Reactor (CRBR). The collection comprises 27 FFTF documents, 35 CRBR documents, and several documents showing the interaction between the FFTF and the Prototype Fast Reactor (PFR) at Dounreay regarding thermal striping. These documents, along with a management-level summary report, are being staged for export-controlled information (ECI) review and Information Release. Avoidance of thermal striping is an important design aspect of a sodium fast reactor (SFR) as adjacent sodium streams, with different temperatures and oscillating, when impinging on steel structures will impose high-cycle thermal stresses and eventually high-cycle fatigue will occur. Temperature differences necessary to avoid thermal striping in an operational sense were contained in the technical specification in the Final Safety Analysis Report (FSAR) for the FFTF and more importantly in the Limiting Conditions for Operation (LCOs).

Requested the Appendices to the FFTF FSAR regarding fuel removal and fuel cleaning or washing. These Appendices were not included in Amendment 77 to the FSAR, which is the version currently available. Investigated why the appendices were excluded and found they existed in the form of Engineering Change Notices (ECNs) to Amendment 77. These ECNs were located by the PNNL Finding and Retrieval Team and scanned into electronic format. Appendix G contains FSAR information on the transition from Operating to Defueling Status; Appendix H contains information on Fuel Offloading; and Appendix I contains information on shutdown of Liquid Metal Systems. The FSAR information contained in these three Appendices is informative and applicable to the corresponding fuel unloading and fuel washing activities in the VTR.

The process of assembling an HT-9 fuel assembly from its respective components, such as wire wrap, clad pins, end caps, rails, and ducts is being researched. Identified and located 105 FFTF fabrication procedures regarding this subject. Retrieved and converted these procedures into electronic format. These procedures are applicable to the VTR HT-9 metal fuel assemblies as they are the procedures used to assemble similar HT-9 metal fuel assemblies, i.e., the seven HT-9 Metal Fuel Fabrication (MFF) tests that were assembled by Westinghouse-Hanford and tested in the FFTF. Reviewed the procedures, along with a management-level summary, for ECI and Information Release and placed on the VTR Confluence Site.

VTR Control Rod Mechanical Design Analysis

Calculated the nominal mechanical lifetime of a VTR absorber assembly. The VTR absorber bundle contained 37 pins with HT-9 cladding and with an HT-9 inner duct and outer duct. The calculated value of 760 Effective

Full Power Days (EFPD) should be regarded as a bounding high number for the following reason: VTR dimensions, core physics, and thermohydraulic information was used wherever and wherever available, and where not, FFTF placeholder or default information were used. The information necessary to close these gaps is being exchanged between PNNL and ANL. ANL is providing reaction rate information to PNNL and PNNL is providing mechanical design information, in the form of dimensions and clearances. to ANL. When complete, a reconciled (physics/mechanical) absorber conceptual design will be available for the VTR. Whether this will produce a shorter lifetime, or a longer lifetime has yet to be determined.

Thomas Fanning, Nuclear Technical Integration

Fuel Design and Analysis

Sent the updated orificing strategy to GEH for review, which reflects revised conceptual designs for driver assembly receptacles (i.e., receptacles in the core grid plate) and a shield receptacle that can accommodate storage of spent driver assemblies. Completed a literature review and scoping calculations assessing potential impact of gallium on VTR driver fuel performance. This document also discussed experiments recommended to further investigate gallium thermodynamic and kinetic behaviors and outlined possible additional modeling and simulation work. Revising the document to address feedback and preparing a specific list of experimental data for BISON U-Pu-Zr-Ga models. Completed a technical report on the sensitivity and uncertainty quantification of the fuel performance based on fuel dimensions and reactor operating conditions as well as the uncertainties in model predictions based on uncertainties in the BISON code. Continued work on BISON simulations of reference VTR driver fuel under transient conditions using alternative approaches for porosity closure, sodium infiltration, and fast neutron flux calculations. Continued to assess relative contributions of thermal creep and irradiation creep to cladding damage. The emerging view is that only thermal creep appreciably contributes to HT9 cladding damage at relevant VTR fuel temperatures. The VTR core design memorandum quantifying effects of impurity variation was assessed and determined to be very helpful for a VTR fuel specification and for determining how to best prepare VTR fuel casting charges from available feedstock.

Fuel Manufacturing

Prepared and submitted a report on the current assessment of Pu feedstock options. Los Alamos National Laboratory (LANL) supported preparation of the Pu supply options report and summarized the aspects of pit-derived plutonium. LANL also began developing integrated options for PF-4 programmatic support, including consideration of the implications of a VTR-related mission. Savannah River National Laboratory (SRNL) provided summary input on Pu polishing options. This included completing a majority of supporting analysis and developing a model to project equipment and manpower requirements for each, as input to Pu polishing report. SRNL also completed characterization of non-pit feeds including estimating impurities present and the fraction of each category that was either suitable for pyrochemical processing or for direct VTR use. This work included developing a model to project for each feed type and for three processing options (mainly aqueous, mainly pyrochemical, and completely pyrochemical) the needed process equipment and operating manpower needed as well as estimated waste produced and resultant radiation exposure. TerraPower competed the HT9 Supplier Qualification Plan, drawing on experience from earlier work. Continued design and procurement preparation efforts for fuel fabrication equipment including casting equipment and rod loading equipment. The contract document for procurement of a prototype casting furnace and safety enclosure was approved for release. Completed procurement for vendor design services of the prototype casting furnace and safety enclosure and submitted for bid. TerraPower completed a Functional and Operations Requirements document for the Rod Loading Equipment by incorporating review comments.

Core Design

The Pressure drop Experimental Loop for Investigations of Core Assembly in advanced Nuclear reactors (PELICAN) is an experiment initiated to support analysis of the hydraulic behaviors within the primary heat transport system (PHTS) of the VTR. The design and measurement capabilities of PELICAN allow empirical measurements of flow rate and pressure drop across prototypic axial reflectors, fuel, and plena segments of a full-scale fuel assembly to complement and validate the ongoing VTR design efforts. PELICAN has been designed to represent prototypic hydraulic conditions within the VTR assemblies using pressurized water to match the viscosity and Reynolds numbers of sodium flow within a VTR. A 50 HP pump is used to drive flowrates up to 40 kg/s and ensure testing conditions can achieve representative flow and pressure drop conditions. Completed an initial testing campaign which included commissioning, facility characterization to assesses the operational capabilities of the control and acquisition systems, and the first phase of experimental testing. For this first phase, orifice plates provided simple test articles for which a known and measurable pressure drop can be generated as a function of flowrate. The use of orifice plates provides both a means of validating the experimental facility, and a simple geometry from which comparative flow simulations can be performed. An example of the pressure drop as a function of flow rate along with theoretical predictions is shown for one such orifice plate in Figure 1. Shown inset is a cross-section of a portion of the PELICAN loop with the orifice plate highlighted in green and the direction of the flow shown with the blue arrow. In the coming months, the experimental testing phase will transition from orifice plates to prototypic test articles. The first such article is an upper reflector which was machined from a single piece of aluminum, shown in Figure 2. This design was selected to serve as a representative model of the main features found in the full-scale component.

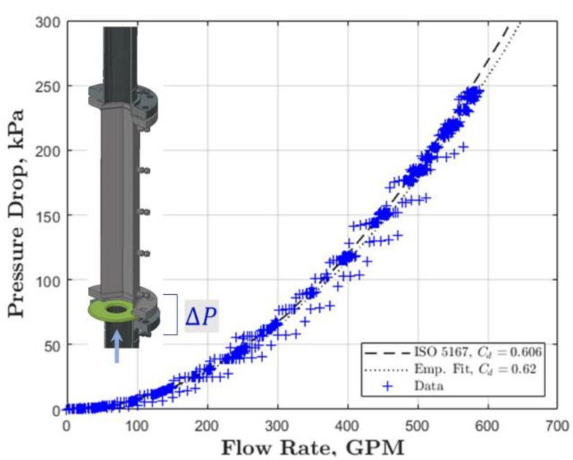

Figure 1. PELICAN generated ΔP vs Q data from first phase testing with orifice plates.

Figure 2. CNC machine test article slated for installation in PELICAN.

Safety Analysis

Completed the first revision of a detailed report that describes all SAS4A/SASSYS-1 model parameters used for safety analysis of the VTR. This will be a living document that is updated as the VTR SAS4A/SASSYS-1 model continues to evolve along with the VTR design. Completed model updates for flow through the thermal insulation cavity (TIC) based on additional information provided by GEH. Additional model development work has begun to update the B24 secondary system following GEH's response to a design information request. As model updates are implemented, the full suite of transient scenarios is repeated to assess the impact of each update on safety margins and peak temperatures. Progress has been reported previously on the computational fluid dynamics (CFD) modeling effort of the upper plenum, using prescribed transient inlet boundary conditions. From this effort, the influence of thermal stratification on the protected station blackout is expected to be significant, and an estimate has been made of the computational expense associated with coupled simulations. Based on this estimate, the decision was made to pursue some simplifications to the CFD model that can reduce the expense of the calculations. In the next phase, the dynamically coupled SAS-CFD simulations will be pursued.

SAS Verification & Validation (V&V)

Drafted SAS/CFD coupling interface requirements specific to the VTR and are under review. Continued verification of the radial core expansion models. Identified issues with the MCTF validation model as the result of an imbalance between flow elements due to incorrect element nodalization in the model.

Safety Basis

Provided a briefing on the VTR Conceptual Safety Design Report (CSDR) to representatives of the Shoshone-Bannock Tribes.

Sodium Fire Hazard Analysis and Software V&V

Improved sodium fire modeling capabilities by developing a compartment atmosphere flow model to better predict oxygen migration to the surface of a sodium pool and sodium drop surfaces. The model changes result in better agreement between simulation results and experimental data because of improvements in predicted sodium burning rates. Continued to make improvements in a flame-sheet burning model which should also improve predicted sodium burning rates.

Probabilistic Risk Assessment (PRA)

Continued preparing for a Technology Inclusive Content of Applications (TICAP) tabletop exercise. Finalized the tabletop report, which includes draft SAR chapters and an explanation of how the content of the chapters was developed and selected, in preparation for the formal tabletop exercise in March. Completed transfer of the PRA model and supporting information from GEH to the laboratory team. Information will be integrated into the new VTR PRA repository to improve multi-organization collaboration and enhance quality assurance (QA) and version control protocols. Software V&V activities are nearing completion for the Simplified Radionuclide Transport (SRT) code Version 2.0, which is being used for mechanistic source term calculations. Centered work on development of the SRT model validation documents, which examines the assumptions and calculations for each phenomena model in the code and provides validation information and analyses.

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Kevan Weaver, Experiments Technical Integration

Selected key accomplishments within the four experiment vehicle types and support areas are included below.

ELTA – Sodium Cartridge Loop Development
Technical Lead: Mitch Farmer, ANL
Partners: University of Wisconsin, Purdue, Framatome

- Continued to work with on the sodium purification and monitoring system to be deployed in the sodium fast reactor (SFR) cartridge loop. Adopted the approach of using a hot trapping technique for purity control and a vanadium wire technique for average purity monitoring over the irradiation cycle for the first-generation sodium cartridge. Framatome and the University of Wisconsin held periodic meetings to inform cartridge loop design development.

- Continued progress on CFD simulations for the SFR cartridge performance. This included refinement of the meshing scheme, with an initial focus on the core simulator region for the scaled-up 7-pin water cartridge hydrodynamic similarity study, and the scaled 3-pin water cartridge core for the thermal similarity study. Completed additional work on meshing the entire 7-pin sodium cartridge loop, including additional flow regions such as the pump region, the downcomer region, and the reactor coolant region.

- Completed the overall scaled 7-pin water cartridge test loop design, including the test section, inlet, outlet, and support structure. Most of the acrylic materials that will be used for the facility have been purchased and delivered. Stainless steel tubes to simulate the fuel rods remain to be purchased. Other components, such as flow instrumentation, pressure transducer, and the data acquisition system are under investigation.

- Continued work on operational testing of a proposed SFR cartridge impeller design in a small PVC water loop in Bldg. 206 at ANL. Initial scoping measurements indicate that the current impeller design will be able to meet the SFR cartridge loop pumping requirements using a three-stack configuration. Installed additional instruments (i.e., shaft torque meter) to provide data needed to support development of the complimentary magnetic pump coupler.

- Completed fabrication of the prototype after installation of custom SmCo magnets into the coupler mechanism by the magnet manufacturer. Completed the test rig to verify operational attributes for the coupler and received safety approval to conduct testing.

- Developed specifications for a sodium loop test stand to verify operational attributes of the SFR cartridge out of pile and are being reviewed.

- Continued to provide technical input and support to assemble a model for the current pre-conceptual design for the SFR cartridge loop to carry out design support and preliminary safety analysis calculations for selected reactor transients as predicted with the SASSYS code. Working on additional model validation using information from the Oak Ridge National Laboratory (ORNL) test stand as the basis.

ELTA – Lead/Lead Bismuth Cartridge Loop Development
Technical Lead: Cetin Unal, LANL
Partners: University of New Mexico, Westinghouse

- Completed water pump loop construction and passed the leak test. Measurement sensors were wired up with the data acquisition system and passed data reading test at the initial run. Measurements in the current loop are flow meter, temperature, and three pressure sensors with time a interval of 1 second (shown in Figure 3). First test produces initial data for a centrifugal pump. First run data is being analyzed and will be tested again after additional flow meter calibration. For the ELTA-CL application, the axial pump is more desirable. We will test the two-stage axial pump after calibrations are updated from the first test. Additionally, more improvements in the water pump loop is being considered: 1) the ball valve in the current loop will be replaced with a gate valve for effective pump curve generation, 2) strobe light sensor will be employed for RPM measurement of the motor.

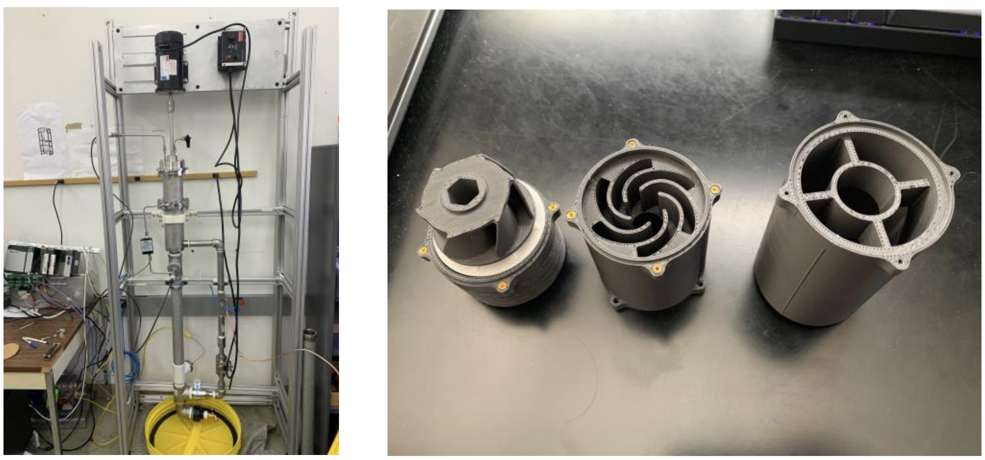

Figure 3. Water pump test loop with all measurement sensor installed (left), and a stage view of a 3D printed axial pump (right)

- Pb melting at the transition from standby mode to nominal operating condition is a critical issue for the ELTA-CL operation perspective. Investigated transient thermal behavior of the ELTA-CL from standby (200°C of Na) mode to nominal operating (350°C of Na) mode. Developed a 3D transient ELTA-CL conduction model to gain some understanding of a characteristic time scale for the Pb melting and associated heat transfer mechanism within the ELTA-CL at the start-up scenario. Initial models were set up for monitoring the Pb temperature change in the loop at the given VTR sodium temperature with no phase change consideration. The result will be used as an input for the cartridge heater design. Additionally, phase change Pb melting simulation is being investigated to understand a more realistic Pb melting time scale.

- Normal operation of the Pb-loop requires the melt tank to be pressurized all the time and the operating pressure window is very small (~43psi). Lower pressure cuts the circulation and higher pressure makes the molten Pb reach the ports of the expansion tank and clog. Therefore, pressure monitoring is needed 24/7 for month-long tests. Decided to isolate the melt tank from the loop using a remote-controlled valve between the melt tank and the loop. Once the Pb is pushed up into the loop, the valve will be closed, and the melt tank will not need to be pressurized. Designed and ordered the valve last summer and installed the valve in February. It is controlled by a pneumatic actuator with a normally closed feature. Made electrical and pneumatic connections and tested via manually introducing power. Radiant heaters were placed back after the leaks were sealed with coupling. Replaced four radiant heaters damaged by the leak at welds.

ELTA – Molten Salt Cartridge Loop Development
Technical Lead: Joel McDuffee, ORNL
Partners: University of Utah, University of Idaho, MIT, TerraPower

- Annular Flow Characterization
    - Completed Modelica updates to match RELAP and TerraPower report validation (i.e., modified sodium outlet temperature to be fixed at 500°C by modifying sodium mass flow rate).
    - Performed a sodium heat transfer correlation study to determine effect on overall annular MSR-EV design performance.
    - Integrated into thermosyphon modeling and design.
    - Developed TRACE models in SNAP of previous thermosyphon designs.
    - Streamlined a process for extracting experimental Thermosyphon Test Loop data, plugging initial conditions into TRACE model templates, running all test cases, and plotting comparisons of experimental and simulated results.
- Pressure and Corrosion Sensor Development
    - Prepared a paper for the 12th Nuclear Plant Instrumentation, Control and Human-Machine Interface Technologies (NPIC & HMIT) conference presenting results of the thermal testing of the aluminum sensor which is being reviewed.
    - Received parts to finish assembly and begin testing the nickel and stainless steel sensors in a molten salt.

RTA – Rabbit Capability Development
Technical Lead: David Wootan, PNNL
Partners: Texas A&M University

- Began construction of the shuttle transfer system mockup for the Rabbit. This mockup will evaluate the performance of the hardware for the Rabbit, prior to installing the actual hardware in the reactor pool. Some delays have been encountered with respect to the data acquisition system, although the overall intent to install shuttle transfer system in the reactor pool remains.
- Investigating the extent to which the performance of the Rabbit may be sensitive to and affected by adjacent experiments. The calculations have required more time than anticipated due to some very small differences in some parameters. Nonetheless, the sensitivity analysis remains underway with an MCNP model containing the Rabbit that has been shown to be consistent with the current VTR configuration.

- Initiated an extensive literature review on flow-reducing components. Initial findings suggest that the flow physics for the Rabbit are well understood and such components can be designed and implemented with a high degree of confidence.

Support Area – Instrumentation and Controls
Technical Lead: Sacit Cetiner, ORNL
Partners: ACU, Georgia Tech, MIT, University of Pittsburgh, Cosylab

- Eddy Current Flow Meter (ECFM)
  - Work on the ECFM continued focusing more on maturing the design for concept demonstration.
  - ORNL and Westinghouse (via the ELTA-CL for Pb) are working collaboratively on advancing the technology for deployment in experiments.
  - Completed the bypass loop design which allows testing of multiple ECFMs in parallel providing insight into the effect of pipe diameter and pipe wall thickness. The effect of these parameters on the signal output will be tested, and the results will be used as validation for multi-parameter sensitivity studies in the finite element model (FEM) simulations.
  - The loop contains an independent flow control valve and a V-cone flow meter for calibration and validation of ECFMs.
  - The loop will be installed on the Transient Test Facility (TTF)—a mercury loop for component testing for SNS at ORNL.
  - Developed a concept that deploys an ECFM as an embedded sensor on the structural shroud element. However, there are important questions that have not yet been properly addressed whether this concept would be viable. The FEM models (COMSOL) are being modified to predict the electrical output from such a configuration. The concept will then be tested in the mercury loop by slightly modifying the bypass loop to accommodate the flow geometry as developed for the ELTA-CL for Pb.
  - Preparing an experimental setup for testing the ECFM sensor at high-temperature operation (separate effects; just temperature, no fluid or radiation effects). The flow of conductive liquid is being emulated with a solid conductor. The design parameters and operation conditions will be modeled in COMSOL, and sensor design will be optimized. The facility will provide important insight for confirming that the material properties (particularly electromagnetics parameters) used in the models are accurate.
- Fast-Spectrum Self-Powered Neutron Detector
  - Executed the tantalum emitter simulations and completed design parameter optimizations.
  - The insulator optimization is still ongoing, and on track to be finished within the next period of performance.