INL/EXT-19-55872-Revision-0

FY2019 August Status Report for the Versatile Test Reactor

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operated by Battelle Energy Alliance

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Idaho National Laboratory
Idaho Falls, Idaho 83415

http://www.inl.gov

Prepared for the
U.S. Department of Energy

Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517

Versatile Test Reactor

John Bumgardner, Program Overview

Versatile Test Reactor (VTR) outreach during August was focused on making final preparations for the web-based public scoping meetings held August 28 – 29, 2019. These meetings are a required activity as a result of the Notice of Intent to prepare an environmental impact statement (EIS) for the VTR. INL Communications and VTR staff assisted the Department of Energy (DOE) in production of four videos shown during the scoping meetings. The videos are as follows: a video giving an overview of the National Environmental Policy Act (NEPA) and the EIS process; a video titled "The Difference Between Power and Test Reactors," which explains the difference between reactors that produce electricity and reactors used for scientific and engineering studies; a video titled "Versatile Test Reactor and Investing in the Future of Nuclear Energy," which outlines the role that a fast neutron spectrum test reactor will provide in the development of advanced nuclear energy technologies; lastly, a video titled "Virtual Tour of a Versatile Test Reactor," which provides a fly-through of the proposed VTR and an accompanying narrative. Through the public scoping meetings, several comments were submitted to DOE-ID which will be addressed. The comment period will remain open until September 4, 2019. The videos detailed above can be viewed on DOE's VTR web page at https://www.energy.gov/ne/nuclear-reactor-technologies/versatile-test-reactor.

To further underscore the need for nuclear energy and the VTR, a Fact Sheet was produced and issued. The entire Fact Sheet can be viewed on the VTR SharePoint site under Program Information or copies can be requested from the VTR program office.

Kemal Pasamehmetoglu, VTR Executive Director, continued outreach efforts giving a VTR overview presentation during the "Technology and Innovation" panel session at the 2019 ANS Utility Working Conference.

WHY VTR?

The United States has long been a leader in the development of nuclear technologies. Currently, there is no fast neutron testing capability in the United States to support advanced reactor research and development. The only capability for fast neutron testing is the Bor-60 reactor in the Russian Federation. The VTR will leverage existing U.S. government and industry investments in nuclear reactors to accelerate its design and construction process, using proven nuclear reactor technology to create a world-class scientific infrastructure.

MEDIA CONTACT

Kortny Rolston-Duce
kortny.rolston-duce@inl.gov

TECHNICAL CONTACT

Kemal Pasamehmetoglu,
Executive Director
kemal.pasamehmetoglu@inl.gov

WHAT IS THE PROBLEM?

The U.S. lacks a scientific facility to provide fast neutron testing, required for rapid and accurate new material and nuclear fuel research and development. Many U.S. companies are working on technologies to make the next generation of reactors and the existing reactors more economically competitive and reliable. New reactors and support of existing reactors require continuing research and development of new materials and nuclear fuels.

New Materials and Nuclear Fuels Development
Development efforts typically start with materials research, progress through refining and testing the material, and finally perform rigorous safety testing to qualify the new material or nuclear fuel for nuclear reactor use. Fast neutrons are used for all phases of this development.

Fast Neutrons for Testing
Fast neutrons have a higher energy level than slow (thermal) neutrons; therefore, they interact differently with the material exposed to these neutrons. This energy level is necessary for developing and providing accelerated testing of materials, fuels, and instrumentation for use in the existing fleet and advanced reactor concepts.

Impact of Testing Gap
The impact of not supporting nuclear technology development includes:

- Losing global nuclear technology leadership.
- Diminished ability to compete in estimated $1 trillion global market.
- Substantially reduced global influence in nuclear safety, security, and nonproliferation policies.

WHAT IS THE SOLUTION?
VTR – A Flexible Resource for Generations

The VTR will use existing proven nuclear reactor technology to provide fast neutrons, and a capability to rapidly insert, conduct, and remove state-of-the-art experiments. Future innovations in experimental capability can be used in the VTR without modification to the facility.

The VTR will provide support for progress in multiple important science and technology areas including:

- Testing and qualification of advanced reactor fuels.
- Testing and qualification of innovative structural materials.
- Testing of innovative components and instruments.
- Validation of advanced modeling and simulation tools, and the versatility to support future technical missions.

By using proven technology, the VTR will leverage existing reactor designs and operating experience to reduce the risk, cost, and time for design and construction.

The VTR development efforts are using the best available resources from the DOE laboratories, industry, and universities to expedite the reactor design and construction, and to develop the scientific infrastructure for a powerful testing capability sustained over many decades.

The VTR specific reactor technology and location are being determined by the U.S. Department of Energy in accordance with capital acquisition and NEPA processes.

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Other News

FY 2020 budgets and work packages are being developed. Funding is anticipated to be steady for FY 2020, which will cause some programmatic challenges, as the program has grown significantly. A second design review was held at GE-Hitachi (GEH) that revealed appropriate and efficient design progress. Additional details from this meeting will be provided by the Technical Integration Principals.

Progress continues on the 30+ deliverables for Critical Decision (CD)-1, with a focus on the Conceptual Design Report (CDR), Conceptual Safety Design Report (CSDR), cost estimate, and schedule.

Pat Schweiger, Reactor Technical Integration Accomplishments

VTR Sodium Fire Analyses

A new approach was formulated for mitigation of sodium fires from postulated bounding and conservative sodium releases inside the secondary sodium drain tank compartment or pipe chase compartment for each secondary sodium loop. The approach collects the released sodium inside a collection tank installed underneath the drain tank compartment for the loop, reducing the potential sodium burning rate. Recovery from the postulated bounding and conservative sodium release event is possible by heating the tank to melt the solidified sodium under an inert cover gas and then pumping out the molten sodium. Further discussions will be held with the GEH-Bechtel design team to complete the conceptual system design.

System Design Description (SDD) Reviews

Input was provided in support of, and reviews were conducted for, multiple SDDs as submitted by GEH. Idaho National Laboratory (INL) and GEH established an SDD tracker that indicates when updates or completion of specific SDD revisions will become available. This tracker will assist in completion of the CDR as well as the modeling and safety analysis for the conceptual design phase.

Design Engineering (GEH) Support

The second comprehensive design review was held August 19 - 22, 2019, in Wilmington, NC, to review the conceptual design work completed by the GEH design team. A description and summary of the major decisions made during the meeting are being compiled and will be incorporated into the outcome report, a milestone due September 18, 2019. In addition, the results of the design review will be incorporated into the CDR being prepared for CD-1. This design review included a quality assurance (QA) assessment from the design review participants.

Pat Schweiger visited the MELOX fuel fabrication and George Besse II uranium enrichment plants on August 13 - 14, 2019. These facilities provide fuel fabrication services for commercial light-water reactors and enriched uranium for the French nuclear power industry and other customers throughout the world. The visit clarified the capabilities and potential limitations associated with obtaining above 5% enriched uranium and the capabilities available to support fabrication of VTR fuel.

The 3D model of the VTR continues to be updated every two weeks to show the latest plant configuration based on on-going conceptual design activities. Significant progress was made in the plant design, electrical, and general plant areas. An updated 3D fly-through video was prepared to show these additional details and was shown at the design review meeting. The 3D modeling effort continues to be effectively used to communicate complex plant configurations and enables better integration between a geographically-disperse team.

Fast Flux Test Facility (FFTF) Documentation and Data Recovery

A total of 232 Royal Industries' control rod drawings (various sizes, up to and including Size J) used in the FFTF have been located, retrieved, converted to electronic format, and loaded to the VTR SharePoint Site. Subsequently, about 70 larger drawings were difficult read due to the incorporation of Export Control markings on the drawings, which reduced the drawings from five megabytes to two megabytes. A corrected set of drawings will be loaded the VTR SharePoint Site in September.

GEH requested drawings related to the Control Rod Disconnect Driveline (CRDD). Twenty-two (22) Westinghouse Hanford drawings with the designation H-4-XXXXX have been located and are being retrieved.

The procedures used for filling the FFTF reactor systems with sodium have been designated as Export Controlled by Pacific Northwest National Laboratory (PNNL). These procedures will be loaded to the VTR SharePoint Site.

VTR Control Rod Mechanical Design Analysis

The preliminary conceptual mechanical design of an absorber bundle is nearing completion. The control rod mechanical design analysis computer code CNRD2, which was used to design FFTF absorber bundles, was used to perform this analysis. This code calculates helium production/retention in the B4C pellet, gas pressure in the upper and lower gas plenums, cladding stress and strain, B4C pellet/cladding mechanical interaction and associated strain, and absorber bundle looseness/tightness due to irradiation swelling. A 61-pin bundle was used for this preliminary assessment although the results can be recalculated for other pin bundle designs. A milestone report is being drafted and is scheduled to be completed and delivered by September 30, 2019.

Rabbit Design for the VTR

The preliminary design for the VTR Rabbit is a collaborative effort between PNNL and Texas A&M University and is based on the conceptual design for the FFTF Rabbit. A work scope has been developed for FY 2020 between the Technical Point of Contact (TPOC) at PNNL and the Principal Investigator (PI) at Texas A&M University. The work scope includes: developing the overall design for the Rabbit including the interface with the VTR core and the Experiment Hall, developing key methods and connections for installing and removing the Rabbit System through the reactor vessel head, assessing neutron activation, gamma and neutron streaming, required shielding, and assessing the impact of the core on Rabbit safety and conversely the impact of the Rabbit on core safety.

Fuels

Worked with INL Bison developers to resolve questions about material models and also contribute Oak Ridge National Laboratory (ORNL) modifications to U-Pu-Zr fission gas release model back to the main Bison source code.

Plant Design and Engineering

Completed fabrication of two new eddy current flow meter (ECFM) probes for testing at the ORNL Spallation Neutron Source (SNS) liquid mercury loop.

Added two Stanford Research low-noise amplifiers and current probes to the data acquisition system for independent signal measurement. This will enable direct digital lock-in signal processing capabilities improving signal sensitivity.

Experiment Development

A second prototype sensor was fabricated out of aluminum to test the sensor response to pressure prior to corroding any of the diaphragm thickness (destructive testing). The sensor data acquisition and processing software is functional and only requires a few small modifications before active sensor interrogation will begin. Various chemicals are being considered for corroding the initial aluminum prototype.

The GEANT4 model for the fast-spectrum SPND design study was finalized.

Jordi Roglans-Ribas, Nuclear Technical Integration Accomplishments

Fuel

Conceptual design of the developmental testing casting furnace, slug processing equipment, and enclosing gloveboxes continued. Functional and operational requirements and procurement specifications for the developmental testing casting furnace, slug processing equipment, and enclosing gloveboxes are ready for release. A revised procurement strategy is needed given the VTR budget uncertainty going into FY 2020. It is likely that the contracted scope will be further divided to allow progress on some key aspects of design within available funding.

A draft fuel performance design basis document to provide criteria for use of U-Pu-Zr in HT9 cladding to 10 atom percent peak burnup remains in process. Safety analysts have requested that additional time-at-temperature limits for fuel cladding interdiffusion be incorporated.

Work defining the discharged fuel management concept identified in the document titled "*VTR Fuel Facility Plan*," (INL-LTD-19-54001) continued to consider prototypic site options for cask transfer from interim storage inside the VTR fence to the Fuel Conditioning Facility. The existing HFEF-5 cask design is being evaluated to determine whether, and how much additional heat rejection capability would be needed to transfer a VTR fuel assembly of spent fuel rods (i.e., 217 rods).

Core Design

Activities focused on the various uncertainties potentially affecting the VTR led to identifying and consolidating a list of over 300 uncertainties and categorizing them based on their type and means of assessing them. Dependence trees are being developed for key performance characteristics, such as fuel and cladding temperatures, to identify key uncertainties contributing to the characteristics of interest. High-impact uncertainties will be the focus of follow-up activities and will be accurately quantified through the use of modeling and simulation tools, experiments results, or engineering judgement.

Safety Analysis

An initial draft for the end of FY 2019 deliverable report has been written that documents the SAS4A/SASSYS-1 models, protected and unprotected safety analyses, a number of other transient scenarios, and select special assessments performed throughout the year. Although incomplete, the preliminary draft has been shared with the Safety Basis team to ensure consistency with the CSDR.

The existing SAS models will be used to begin performing sensitivity analyses so input parameters that have significant impact on limiting margins can be identified. This evaluation will help prioritize future uncertainty quantification efforts.

Safety Basis and Probabilistic Risk Assessment (PRA)

Efforts focused on resolving questions and issues associated with design improvements and development and finalization of documents necessary to support the CSDR.

PRA resolutions also focused on working toward a risk-informed design for the Reactor Protection System. Close coordination with the design team yielded a very robust design with a very low failure rate.

Activities to finalize PRA documents involve the development of calculations and technical evaluations in support of transitioning scoping memos into referenceable technical documents as well as performing checking and model finalizations for the updated PRA activities as a result of design information updates from the previous version of the PRA. Data on sodium piping failures at previous sodium fast reactors and sodium test loops has been reviewed for applicability to the VTR. Findings will be summarized in a Technical Evaluation (TEV). The information will also be used as the basis to develop event trees to represent sodium piping failures for three areas of the facility.

Alternate Safe Harbor Approach Prepared

An alternate safe harbor approach was prepared and internally reviewed and discussed with DOE Safety Basis personnel. This alternative methodology provides a clear mechanism to demonstrate DOE approval of the selected guiding safety basis methodology and provides clarity on how the project approach to development of the safety basis meets the requirements of 10 CFR 830 Subpart B associated with safety basis documents.

Initial NEPA Information Response Prepared

An initial document structure and content outline was prepared to facilitate timely support of the data needs request, once formally issued. An initial draft copy was provided that created a structure to enable capturing inputs in a consolidated manner to support the EIS schedule as well as provide a mechanism for tracking available data and what still needs to be collected.

Software QA activities

The first phase of the MCC-3 and DIF3D verifications are nearing completion. Significant progress has been made in leveraging ZPPR-15 loadings to validate neutronics codes used for VTR. Impact on the core performance characteristic, control rod worth, and reactivity coefficients is being assessed for single representative fuel experiments fueled with oxide, metallic, nitride, and carbide fuels.

The team continues to develop verification and validation cases for the Reactor Vessel Auxiliary Cooling System (RVACS) modeling capability in SAS4A/SASSYS-1. Validation is based on past Natural convection Shutdown heat removal Test Facility (NSTF) testing conducted at Argonne in the 1980s. The configuration of NSTF at that time was consistent with the PRISM Mod-A vessel and guard vessel geometries.

Los Alamos National Laboratory (LANL) discovered some inconsistencies in TRACE-NA relating to junctions. For the pump coast down studies, as they are all single phase, LANL will continue with the Nuclear Regulatory Commission reference version 5.0Patch 5. LANL staff are assessing results with the new code version for the pump coast down transient.

VTR Conceptual Design Report

Many new sections were added to the draft VTR CDR. Several meetings were held to gather additional information on systems and design decisions in the current conceptual VTR design (i.e., fuel handling, instrumented assemblies, Rabbit test loop, cartridge closed loops, handling of the test vehicles at the VTR).

Kevan Weaver, Experiments Technical Integration Accomplishments

The Experiments team within the VTR program includes nine "functional" areas aligned with the experiment vehicle types, along with related capabilities and initiatives, anticipated to be utilized within the VTR. Each area is led by a national laboratory technical expert and is supported by other national laboratory personnel, university partners, and industry partners.

The Level 3 milestone titled "*Versatile Test Reactor Experiment Vehicle Development: Integrated Status Report of FY-18 Awards Concept Development,"* INL/INT-19-55188, Rev. 0, dated July 2019, was completed. This report provides a summary of the work performed through the last year (beginning in FY 2018 and ending in FY 2019) to support the design of the test vehicles, and thus the infrastructure in the VTR, needed to support projected future irradiation testing.

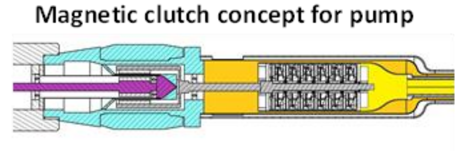
The Level 3 milestone document titled "*VTR Draft Scope and Cost Estimate for Experiment Capability,"* INL/LTD-19-55593, Rev. 0, dated August 29, 2019, was completed. This document provides the necessary experiment capability infrastructure requirements with associated scope definition and estimated costs based on expert judgment and current scope assumptions. This document will feed ongoing formal estimating as part of the CD-1 cost estimate currently under development.

A few key accomplishments within several of the functional areas are included below.

Sodium Capability Development
Technical Lead: Mitch Farmer, ANL
Partners: University of Wisconsin, Purdue, Framatome

- Working with industrial partner Framatome, completed a testing requirement report for the sodium fast reactor area to guide the VTR experiment planning process.

Magnetic clutch concept for pump



- Completed the development, documentation, and validation of an engineering design tool to support safety assessments for cartridge loops was completed.

Early Concept: detachable under-sodium instrument connector

- Completed the development of a compact, stageable, high head cartridge pump design based on a water well pump concept.
- Completed the design for, and are assembling, a small water test loop to verify predicted performance using water. If successful, a plan for under-Na testing in Mechanisms Engineering Test Loop Facility (METL) will be developed.
- Continued development of a concept for a detachable, hermetically-sealed, magnetic clutch pump drive.
- Completed the development, fabrication, and testing of a detachable-under sodium instrument harness was completed.

Lead Capability Development
Technical Lead: Cetin Unal, LANL
Partners: University of New Mexico, Westinghouse

- Building a UNM material test loop operating with lead up to 700ºC. Close to start of operation.
- Determined the desired design capabilities and goals of the VTR lead cartridge.
  - Drafted cartridge testing requirements.
  - Developed a preliminary test matrix for out-of-pile testing at UNM during first 3 years.
  - Plan to corrosion test MA 956 w/Weld, 316 w/Weld, APMT, C26M, FeCrSi, HT9, D9, SiC.
  - Targeted conditions - 500 C, 3 m/s (1st yr), 700 C ,1 m/s (2nd yr), 700 C, 3 m/s (3rd yr).
- Acquired initial requirements for fuels and materials testing from the industry partner.
- Expecting to fully execute the contract with Westinghouse by mid-September.

Molten Salt Capability Development
Technical Lead: Joel McDuffee, ORNL
Partners: University of Utah, University of Idaho, MIT, TerraPower

Thermoshyphon Test Loop (TSTL)

- Completed conceptual design of a static salt experiment, with a variety of additional conceptual designs planned to support a range of molten salt experiments.
- Completed restart of the thermosyphon test loop (TSTL) at ORNL, which has similar geometry to an annular flow molten salt experiment in VTR.
  - Existing facility will be used to gather flow data to support cartridge loop modeling effort for both molten salt and sodium.
- Fabricated an aluminum prototype sensor and will use the sensor to test response with active pressurization before proceeding to destructive corrosion testing. Corrosion is the primary concern for molten salt irradiation testing.

Material Capability Development
Technical Lead: Tarik Saleh, LANL
Partners: Oregon State University, Purdue, EPRI

- Enabled gaseous testing capabilities at OSU through design and procurement of a custom autoclave capable of mating with the traditional electronic actuator and housing the compact, bellows-loaded device.
    - The autoclave is capable of pressures up to 2900 psi and temperatures up to 1200°F (649°C), enabling testing in high temperature/high pressure gaseous environments. This test apparatus is currently being assembled, with crack growth testing expect to commence shortly thereafter.
- Down-selected potential nondestructive testing (NDT) technologies at OSU for monitoring crack growth in various advanced environments and focused on adapting potential drop and acoustic emission monitoring systems for use in cartridge loop environments.
- Issued the Purdue contract with a focus on establishing a mechanical property testing capsule that incorporates novel sensors to measure in-pile creep, ultimate tensile strength, yield strength, temperature, local power, and neutron flux.

Upcoming September Events

September 10 – 11, 2019, VTR Workshop with Japan, SpringHill Suites Idaho Falls
September 11 – 12, 2019, VTR Quarterly Integration Meeting, SpringHill Suites Idaho Falls