Development of an Integrated Mechanistic Source Term Assessment Capability for Lead- and Sodium-Cooled Fast Reactors

Final CRADA Report

Nuclear Science and Engineering

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Non Proprietary Final CRADA Report

For the Office of Scientific and Technical Information (OSTI)

CRADA Number: A17150

CRADA Title: Development of an Integrated Mechanistic Source Term capability for Lead- and Sodium-Cooled Fast Reactors

CRADA Start Date 1/5/2018 – **End Date** 7/5/2019

DOE Program or Other Government Support

Program office: DOE, Office of Nuclear Energy

Program manager name:

Program manager phone or email:

Participant(s)

Participant 1 name: Fauske & Associates, LLC

Complete address: 16W070 83rd Street, Burr Ridge, IL 60527

Participant 2 name:

Complete address:

Participant 3 name:

Complete address:

Argonne National Laboratory

Argonne PI(s): Tanju Sofu

Funding Table

To add rows, right-click in bottom row and select "Insert" "rows above".

	Planned Funding	Actual Funding	In-Kind
Government	400,000\$	400,000\$	
Fauske & Associates	100,000\$	100,000\$	\$
	\$	\$	\$
	\$	\$	\$
Total	500,000\$	500,000\$	\$

Nature of Work

Describe the research (summary of Scope of Work and principal objectives of the CRADA):

The scope of this project entails development of mechanistic source term assessment capabilities to inform and guide plant design activities and to support license applications. The main objective of this project is to couple ANL's SAS4A safety analysis code with the FAI's FATE facility modeling code for mechanistic source term assessments in liquid-metal-cooled fast reactors. The end product is a coupled SAS4A-FATE code system for integrated mechanistic source term assessments and Level-2 PRA as the precursor for off-site dispersion analyses (Level-3 PRA). The SAS4A code is used to model system transients with fuel failures and its results are linked to the FATE code to predict radionuclide transport through the reactor vessel cover gas and containment system. The integrated code package offers a state-of-the-art radionuclide tracking capability for a spectrum of accident scenarios consistent with the plant dynamic response including the reactivity feedback to support Probabilistic Risk Assessments (PRA).

DOE mission area(s):

Energy and Environmental Science and Technology

Conclusions drawn from this CRADA; include any major accomplishments:

The goal of this CRADA was achieved. FATE is coupled to SAS4A/SASSYS-1 and a radionuclide release module was written to use the fuel failure results from SAS4A/SASSYS-1 to release radionuclides to the coolant and to the cover gas region. Once released, FATE tracks transport and retention of radionuclides in the primary vessel and in the containment. The model of the Demonstration Lead-cooled Fast Reactor that Westinghouse proposed as part of the US DOE Advanced Demonstration and Test Reactor Options (ADTRO) study was created using SAS4A-FATE. A sample calculation was performed to demonstrate the mechanistic source term analysis capability of the integrated tool to model the release and transport of radionuclides following an unprotected transient overpower (UTOP) scenario in a generic lead-cooled fast reactor.

Thermo-physical properties of liquid lead in SAS4A and FATE were assessed. FATE was coupled to SAS4A/SASSYS-1 as a plug-in that gets dynamically loaded at run time. FATE is invoked only when ex-vessel modeling, such as containment, is needed. A new routine called FATE-WRAPPER was created in SAS4A/SASSYS-1, where program control handover and data exchange between the two codes occur. SAS4A/SASSYS-1 calls FATE at the end of each time step before the single-time-step integration is performed.

A radionuclide release module (RRM) was developed to link the SAS4A/SASSYS-1 and FATE codes. Specifically the RRM determines and distributes the radionuclides released from the failed fuel pins between the primary coolant and the cover gas region. With the initial inventory of radionuclides in the fuel matrix and in the gap provided as user input, and the fuel temperatures and extent of fuel pin failure calculated by SAS4A/SASSYS-1, the RRM determines the additional radionuclide releases from the fuel matrix during the accident when the fuel overheats and, when the fuel pin fails, discharge rates to the coolant and to the cover gas region. For example, noble gases are assumed to be released directly to the cover gas region. Radionuclides that are discharged from the failed fuel pin but do not reach the cover gas region become part of the primary coolant. Experimental investigations will ultimately be required to measure the decontamination factors in liquid lead for radionuclides of interest, and such factors incorporated into the RRM.

The RRM was constructed based on the ORNL-Booth diffusion release model, which uses the classical single-atom diffusion equations and diffusion coefficients that are supported by experimental data for oxide fuels. The RRM was validated against experimental data from the Vertical Induction Tests (Oak Ridge National Laboratory), VERCORS Tests (Grenoble Nuclear Centre), and Phebus FPT-1 Tests (Institut de RadioProtection et de Surete Nucleaire).

A SAS4A/SASSYS-1 model was developed for the Demonstration Lead-cooled Fast Reactor that Westinghouse proposed as part of the ADTRO study. The model represents the coolant choice (i.e. lead), fuel pin and assembly dimensions, primary circuit configuration, number of primary heat exchangers, pumps, and the reactor vessel auxiliary cooling system. The core reactivity feedbacks are modeled on the order-of-magnitude accuracy without the detailed axial and radial distribution in the core and other details such as the design of the core restrain system.

Similarly, a thirteen region FATE model was developed to represent the individual liquid pools and cover gas region in the reactor vessel, annular gap between the reactor vessel and guard vessel, reactor vault auxiliary cooling system, and containment building. In the reactor core, the time-dependent core power is user input and is made consistent with that calculated in the SAS4A/SASSYS-1 model. Transport and deposition of radionuclides in

the containment and ultimate release to the environment is tracked by FATE. Radionuclides retained in the liquid pool are tracked by FATE using the inter-node liquid flow rates calculated by SAS4A/SASSYS-1. Therefore, the SAS4A/SASSYS-1 vessel model is duplicated in FATE. The in-vessel thermal hydraulics condition (such as coolant mass and temperature) calculated by SAS4A/SASSYS-1 is imposed on the FATE model at each time step. The FATE model tracks extra species (i.e., not considered in SAS4A/SASSYS-1 such as radionuclides) in the liquid pool. Evaporation and condensation of these species at the pool surface are determined by FATE based on vapor pressures. To test compatibility of the two codes thermal-hydraulic calculations performed by the two codes for a protected station blackout, in which loss of cooling from primary heat exchangers is followed by reactor trip at the beginning of the transient, are compared. They showed reasonable agreement, confirming validity of code coupling.

An unprotected transient overpower (UTOP) accident was simulated using SAS4A-FATE, demonstrating its capability to predict fuel heatup and failure, release of radionuclides in the fuel, and transport of radionuclides in the primary coolant, cover gas region, and containment. The results also reveal gaps that exist in the methodology. In the current model, chemical forms of radionuclides in lead coolant and pool scrubbing have not been analyzed. Instead, pool scrubbing is considered using user-input decontamination factors. There is a need for experimental data as well as parallel modeling effort to identify chemical forms of the radionuclides in lead coolant and model their retention in the coolant. Another phenomenon that needs to be addressed is enhanced radionuclide evaporation rates due to fog formation at the pool surface.

Technology Transfer-Intellectual Property

Argonne National Laboratory background IP:

SAS4A/SASSYS-1 liquid metal reactor accident analysis code

Participant(s) background IP:

1. FATE facility modeling code
2. Westinghouse Demonstration Lead-cooled Fast Reactor design

Identify any new Subject Inventions as a result of this CRADA:

A Radionuclide Release Module (RRM) was developed in this project. The module determines grouping of radionuclides and their release rates from overheated fuel.

Summary of technology transfer benefits to industry and, if applicable, path forward/anticipated next steps towards commercialization:

The collaboration provides the mechanistic source term analysis capability for licensing purposes to advanced reactor developers. The next step is to address following gaps in the methodology through experiments and analyses:

1. Radionuclide retention capability of a liquid metal pool – this requires experiments and analyses to identify chemical forms of radionuclides and Henry's Law constants in a specific coolant type
2. Fog enhanced evaporation of radionuclides on the liquid metal pool surface – this requires developing appropriate models for the phenomenon and experiments to validate the model

Other information/results (papers, inventions, software, etc.):

Hua, T.Q., Moiseyev, A., Karahan, A., Tentner, A.M., Sofu, T., Lee, S.J., Paik, C.Y., Liao, J., and Ferroni, P., "Development of Mechanistic Source Term Analysis Tool SAS4A-FATE for Lead- and Sodium-Cooled Fast Reactors," Nuclear Technology (2019), <https://doi.org/10.1080/00295450.2019.1598715>.

S.J. Lee, C.Y. Paik, T.Q. Hua, A. Moiseyev, A. Karahan, A.M. Tentner, T. Sofu, J. Liao, and P. Ferroni,
ANL-1060 (06/07/2018)

“Development of the Source Term Analysis Tool SAS4A-FATE for Lead- and Sodium-Cooled Fast Reactors,” SESAME International Workshop, Petten, The Netherlands (2019).

S.J. Lee, C.Y. Paik, T.Q. Hua, A. Moiseyev, A. Karahan, A.M. Tentner, T. Sofu, J. Liao, and P. Ferroni, “SAS4A-FATE Source Term Assessment Capability Applied to the Westinghouse Lead Fast Reactor,” International Congress on Advances in Nuclear Power Plants, Juan Les Pins, France (2019).

Hua, T.Q., Lee, S.J., Liao, J., Moiseyev, A., Ferroni, P., Karahan, A., Paik, C.Y., Tentner, A.M., and Sofu, T., “Development of a Radionuclide Release Module for an Integrated Source Term Analysis Tool for Liquid Metal Cooled Fast Reactors,” the 18th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-18), Portland, OR (2019).

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