APPLICABILITY ENHANCEMENT OF SAS4A/SASSYS-1 COMPUTER CODE TO LEAD FAST REACTOR SYSTEMS

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Abstract

The demonstration of applicability of safety analysis computer code to a nuclear reactor system is of importance for the licensing of the reactor. The applicability includes the capability of the computer code to addresses all important phenomena expected in the reactor and the verification and validation of the computer code for these phenomena. The SAS4A/SASSYS-1 system level computer code is the safety analysis computer code for the Westinghouse lead fast reactor (LFR) and demonstrating its applicability is an important task in the LFR development.

The SAS4A/SASSYS-1 code is capable to simulate anticipated operational occurrences, design basis accidents and beyond design basis accidents of liquid metal fast reactors, and the major strengths of SAS4A/SASSYS-1 include a complete reactivity feedback model, a comprehensive fuel rod performance model, and extensive applications to liquid metal reactors. Over the past few years, the SAS4A/SASSYS-1 code was further improved as a part of various United State Department of Energy programs to extend its applicability to LFR systems. The improvements include the capability of mechanistic source term analysis, oxide fuel modeling enhancements, the capability of simulating the passive heat removal system through coupling with the GOTHIC code, the capability to simulate primary heat exchangers, and the enhancement of computer code verification and validation base specific to LFR systems. The validation is expected to be further enhanced with the LFR testing program, which includes experiments in United Kingdom Advanced Modular Reactor phase 2 program, LFR integral effects test facility, and International Atomic Energy Agency collaborations.

1. INTRODUCTION

The Westinghouse Lead Fast Reactor (LFR) (Figure 1) is a 450 MWe class, lead-cooled, fast neutron spectrum, pool-type reactor being developed by Westinghouse Electric Company in collaboration with domestic and international organizations [1]. The plant leverages the inherent favorable properties of liquid lead as a coolant and utilizes passive safety systems for high reliability and public safety.

As a key part of the Westinghouse LFR program, safety analysis of the LFR provides evidence to support the LFR safety case as well as providing feedback to improve the safety system design of the LFR. Though the safety analysis methodology is under-development, the safety analysis is envisioned to be a framework of various analyses utilizing a suite of computer codes characterized by multiple resolution levels, ranging from system-level analysis, component-level analysis and high fidelity analysis, and multiple disciplines including reactor physics, thermal hydraulics, radionuclide transportation, and materials chemical interactions, to support the reactor design and licensing activities. Among the suite of computer codes, SAS4A/SASSYS-1 (SAS) is the system level safety analysis computer code for analyzing licensing basis events.

The SAS4A/SASSYS-1 safety analysis code, developed and maintained by Argonne National Laboratory (ANL), provides the transient simulation of anticipated operational occurrences (AOO), design basis accidents (DBA), and beyond design basis accident (BDBA) conditions in liquid-metal cooled fast reactors. The code maintains unique capabilities to account for inherent fast spectrum feedback effects and passive safety features, key elements of the sodium-cooled fast reactor (SFR) and LFR safety basis. The software facilitates assessment of key safety basis metrics, including margins for structural thermal limits, metallic fuel failure, sodium boiling, and fission product release.

The Westinghouse LFR program specifically selected SAS4A/SASSYS-1 (SAS) computer code for the LFR safety analysis and as a framework for safety analysis methodology based on a few important considerations.

- SAS4A/SASSYS-1 is a mature code with the development in ANL for years and has been applied to liquid metal fast reactors. It is capable to model important phenomena in liquid metal fast reactor such as fuel rod performance, reactivity feedback in the core, reactor vessel auxiliary cooling, etc.

— SAS4A/SASSYS-1 has a long history of being applied to SFR design and licensing. It was used to support licensing SFR with the United States Nuclear Regulatory Commission.
— SAS4A/SASSYS-1 provides a good basis of quality assurance, such as fundamental verification & validation (V&V) database, and American Society of Mechanical Engineers NQA-1 qualification.
— ANL has established a long term and good collaboration with the Westinghouse LFR program.

Therefore, adoption of the code is the most effective way to satisfy the basic requirements for the development and assessment of a safety analysis evaluation model for the Westinghouse LFR program.

Computer code applicability is a key step in the safety analysis evaluation model development and an vital part of licensing readiness. It increases the confidence of applying the computer code to the specific safety analysis and determines the bias and the uncertainty of the evaluation model. The applicability includes the capability of the computer code to addresses all important phenomena expected in the reactor and the verification and validation of computer code for these phenomena. This paper seeks to demonstrate the applicability of SAS4A/SASSYS-1 to the LFR by presenting the capability enhancement of the computer code and verification and validation of the code to the LFR design. The paper also presents the applicability of GOTHIC computer code, which is coupled with the SAS code to perform the safety analysis of the LFR, specifically modeling passive heat removal system (PHRS) of the LFR.

Section 2 provides an overview of the SAS4A/SASSYS-1 computer code and the GOTHIC computer code. Section 3 lists the current V&V of the SAS code and the GOTHIC code for the LFR, such as lead verification database, benchmarking against testing data. Section 4 describes plans for additional V&V for the LFR system. Section 5 is the conclusion.

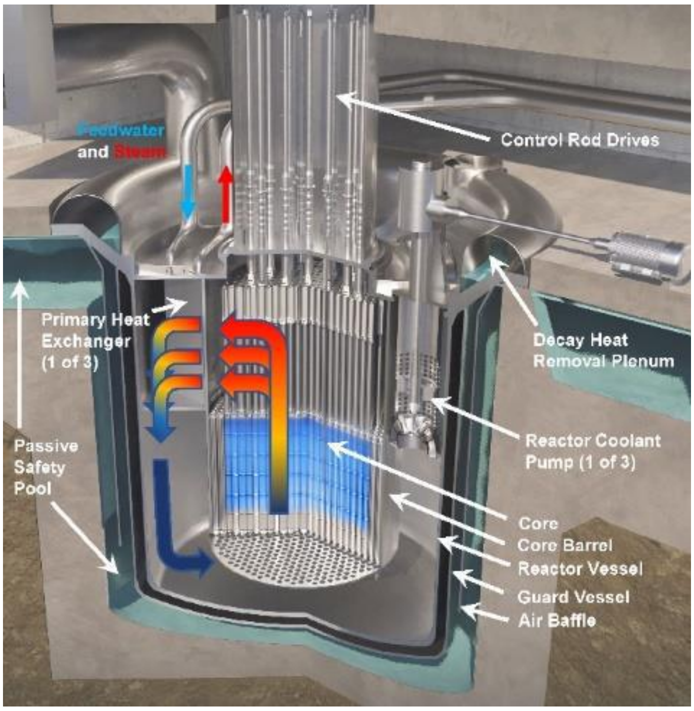

FIG. 1. Schematic of representative system layout of the Westinghouse LFR.

2. SAS4A/SASSYS-1 AND GOTHIC COMPUTER CODES OVERVIEW

2.1. SAS4A/SASSYS-1 computer code

The SAS4A/SASSYS-1 computer code was developed by ANL initially to perform safety analysis of SFR and was subsequently provided with the properties of lead coolant to extend its applicability to LFR systems. The SAS code is capable to simulate AOO, DBA and BDBA of liquid metal fast reactors. The V&V of this code covers fundamental V&V test suite for SFRs and includes validation against experimental data from Experimental Breeder Reactor (EBR-II) shutdown heat removal tests, and Fast Flux Test Facility (FFTF) loss of flow without scram tests. The SAS computer code was licensed to Westinghouse in 2017 to support the LFR program.

The major strengths of SAS4A/SASSYS-1 include a complete reactivity feedback model, a comprehensive fuel rod performance model, and extensive applications to liquid metal reactors demonstrated as part of multiple domestic and international programs. The SAS code also includes a reactor vessel air cooling system (RVACS) model, which was originally used for the design of the Westinghouse LFR to simulate the performance of the PHRS, but only when operating in the air-cooling mode. The SAS RVACS model does not include water-cooling analysis capabilities.

Westinghouse has been working with ANL to improve the capability of the SAS4A/SASSYS-1 code to model the Westinghouse LFR system [2]. Major achievements include the development of a mechanistic source term analysis capability, an improved oxide fuel performance model, code coupling between SAS and GOTHIC to address the PHRS model (SAS-GOTHIC, see Section 2.1.2), the capability of modeling the LFR primary heat exchangers, and an enhanced V&V matrix including integral effects tests. Key developments have been made as part of four projects, which are listed below:

- Gateway for Accelerated Innovation in Nuclear (GAIN) project: “Development of an Integrated Mechanistic Source Term Assessment Capability for Lead and Sodium-Cooled Fast Reactors”
- Technology Commercialization Fund (TCF) project: “Joint Development of SAS4A Code in Application to Oxide-fueled LFR Severe Accident Analysis”
- TCF project: “Qualification of System-Level Advanced Reactor Safety Analysis Software for Lead Systems”
- TCF project: “Capability Enhancement for System-level Thermal Hydraulic Modeling of Lead Fast Reactors”

Each enhancement of capability of SAS4A/SASSYS-1, such as improvement of lead/LBE properties, capability of mechanistic source term analysis, oxide fuel model, SAS-GOTHIC coupling, and primary heat exchanger model, are presented in the following subsections.

2.1.1. Improvement of Lead/LBE properties

The lead and lead-bithmus-eutectic (LBE) properties were originally incorporated into the SAS code in 1980’s. A review of the lead properties library indicated that the lead and LBE property libraries in SAS4A/SASSYS-1 were not consistent with the industry recognized OECD lead and LBE database [3]. Therefore, the properties from OECD database were studied and the suitable correlations were identified. These correlations were then converted to the format that matches the built-in formulations inside the SAS code. The error introduced by this approach was considered negligible. The following properties of lead and LBE were updated in the SAS code to be consistent with the OECD lead and LBE database:

- critical temperature
- heat of vaporization
- saturation pressure
- saturation temperature
- liquid density
- vapor density
- liquid specific heat
- vapor specific heat
- adiabatic compressibility
- thermal expansion
- viscosity

2.1.2. Development of mechanistic source term analysis capability

A mechanistic source term capability for LFRs and SFRs was developed by coupling the SAS code with Fauske & Associates’ (FAI) FATE facility modeling computer code [4]. A Radionuclide Release Module (RRM) was developed as a part of the code coupling effort to model radionuclide releases by overheated fuel and their retention in the lead coolant. A representative LFR was modeled with each code. The RRM was validated by comparing predicted radionuclide release fractions against experiments, for UO2 and MOX fuel. An unprotected transient overpower (UTOP) accident was simulated using the coupled SAS4A-FATE code, demonstrating its capability to predict fuel heatup and failure, release of radionuclides in the fuel rod, and transport of radionuclides in the primary coolant, in the cover gas region, and in the containment. The coupling technique, benchmarking and demonstration results were reported in [5][6][7].

2.1.3. Enhancement of oxide fuel model

This enhancement was aimed to improve the oxide fuel models for both UO2 and MOX fuels for the LFR system. The main achievement was the development of new SAS modules for oxide fuel performance in lead coolant [8]. This included improving physical models within the oxide fuel performance module in SAS, implementation of 15-15Ti cladding mechanical and creep rupture properties, and development of a stochastic cladding failure propagation model [9] specifically for LFR. The improved physical models for the oxide fuel module include fission gas model, fuel chemistry model, and fuel thermal conductivity and gap conductance model. New models such as those simulating the fuel dimensional changes, fuel cracking, fuel hot pressing, and fuel-cladding mechanical interaction were also incorporated. These models enable realistic analysis of fuel behavior including, if occurring during a postulated event, fuel-coolant interaction.

2.1.4. Development of SAS-GOTHIC coupling capability

This development aimed at improving SAS modeling capabilities for the LFR PHRS. One of the key developments made in the project stemmed from the lack of a modeling capability to address the water-cooling mode of the PHRS within SAS, as SAS only captures the air-cooling mode. This is because, in an accident scenario, the PHRS water surrounding the guard vessel provides early and long-term cooling capability for the reactor, whereas air as the infinite heat sink removes decay heat for extended long-term cooling. The modeling capability requires the safety analysis computer code to be able to model both water cooling and air cooling. Therefore, in consideration of its extended use for the safety analysis of PWR containment systems, GOTHIC (Section 2.2) [10] was selected to fill the SAS modeling gap and to address the entirety of the PHRS cooling modes for the Westinghouse LFR. The coupled SAS-GOTHIC code demonstrated its capability for modeling water cooling, air cooling and the transition from water cooling to air cooling during accidents.

2.1.5. Development of primary heat exchanger model

Another area of SAS development pertained to modeling the Westinghouse LFR’s primary heat exchangers, in which the lead coolant in the primary side and the coolant in the power conversion system are thermally coupled. Because of its advantage of compactness, the Westinghouse LFR design adopted hybrid microchannel-type heat exchangers, which are diffusion-bonded heat exchangers characterized by a very large heat transfer area-to-volume ratio and high reliability in accommodating extremely high differential pressures. Since SAS is limited to modeling the heat exchangers as being coupled with intermediate loop, a new heat exchanger model that addresses the LFR design was successfully developed and demonstrated.

2.2. GOTHIC computer code

GOTHIC is a general-purpose thermal-hydraulics software package for design, licensing, safety and operating analysis of nuclear power plant containments, confinement buildings and system components. Applications may include pressure and temperature determination, equipment qualification profiles and inadvertent system initiation, and degradation or failure of engineered safety features. GOTHIC can be used for a wide variety of plant operations support issues involving single and multiphase heat transfer and fluid flow provided that the application is consistent with the underlying physical basis and assumptions and the code validation basis. A two-dimensional or three-dimensional model can be created with the actual plant geometry to calculate steady state or transient operating conditions. The code contains multiple options for heat transfer including two-phase and boiling correlations.

In the Westinghouse LFR, removal of decay heat during accident conditions is performed through the PHRS, which includes the annular water pool outside of the guard vessel (GV). To analyse the performance of the PHRS during a postulated accident scenario, the GOTHIC computer code is coupled to the SAS code since the latter lacks the capability to model the thermal hydraulics of water in the PHRS. The GOTHIC computer code was therefore selected for the portion of the PHRS located outside of the RV as it can model noncondensable gases, steam, water droplets and/or liquid water. Furthermore, GOTHIC includes the capability to run simultaneously with other codes and to communicate information back and forth as the calculations proceed. Westinghouse and ANL collaborated to enable the SAS4A/SASSYS-1 code coupling with GOTHIC and to ensure the data was being transferred appropriately between the two codes[11]. Together, SAS-GOTHIC can perform safety analysis for the Westinghouse LFR.

3. COMPUTER CODE VERIFICATION AND VALIDATION FOR LFR

Following the EMDAP in Regulatory Guide 1.203 [12], the safety analysis computer code, after the development phase, will pass through V&V of the computer code. Verification is the process of ensuring that the products and processes of each major activity of the life cycle meet the standards for the products and the objectives of that major activity. The verification of computer code usually refers to comparison against idealized problem or analytical solution to justify the functionality of code. Validation is the process of demonstrating that the as-built software meets its requirements. Validation is usually performed via benchmarking against experimental results or a system performance. The purpose of assessment is to validate the models in the safety analysis code and to determine the bias and uncertainty of these models.

In the validation, the performance of the computer code is assessed against the ranked phenomena in phenomena identification and ranking table (PIRT) for accident scenario. The LFR PIRT [13] identified the phenomena that are important to the safety of the plant and determines what the state of knowledge is for each phenomenon. The LFR testing program was established to address the knowledge gaps and the safety analysis computer codes were developed to address safety significant phenomena and provide validations.

In this section, the assessment matrix of SAS4A/SASSYS-1 and GOTHIC is presented. The SAS verification database and its expansion to the LFR system is presented in section 3.1. The SAS validation against a matrix of separate effects tests (SETs) and integral effects tests (IETs) are provided in section 3.2 and section 3.3 respectively. The V&V of GOTHIC is provided in section 3.4. The additional planed V&V is presented in Section 4.

3.1. Verification of SAS for Liquid Metal Reactors (SFR and LFR)

The SAS4A/SASSYS-1 V&V Test Suite currently contains over 300 test cases[14]. These tests incorporate verification, validation and training input models for various components and system configuration. Verification test cases were developed using the methodology presented in [14]. Analytic solutions were derived and comparisons were made between the SAS4A/SASSYS-1 predictions and the analytical solutions. The majority of the verification cases, which were summarized in Reference [14], utilize sodium as the reactor coolant and are based on facility layouts that are characteristic of an SFR. At a more granular level, many of the verification test cases can be considered coolant agnostic. These test cases were categorized into 6 groups:

1) Simple Steady-State Cases: Case 1 is the SFR fuel channel test case, Case 1.2 - Case 1.9 verify that SAS4A/SASSYS-1 correctly captures additional complexity that can be built on top of a base model.
2) Simple Transient Cases: Case 2.1 - Case 2.4 verify that the transient solver routines correctly predict the base model response to a zero transient, or simple change in the boundary conditions.
3) Material Properties Cases: Case 3.1- Case 3.6 verify the sodium properties in SAS4A/SASSYS-1.
4) Core Power Cases: Case 4.1 - Case 4.22 verify the core power models.
5) Heat Removal Systems Cases: Case 5.1-Case 5.16 verify capability of heat removal components. Case 5.17, Case 5.18, and Case 5.23 – Case 5.26 verify that SAS4A/SASSYS-1 correctly captures heat transfer between components. Case 5.19 - Case 5.22 verify capability of natural circulation.
6) Control System Cases: Case 6.1 – Case 6.4 verify the control system logic and its ability to read measured signals.

Most of test cases are equally applicable to both the SFR and the LFR and they are retained in the LFR V&V basis. In order to extend the coverage of the SAS4A/SASSYS-1 verification test suite into LFR design space, several of the SFR specific test cases were recreated for lead as a coolant and facility layouts that are more representative of an LFR. In order to reduce the number of new test cases, the SFR specific test cases were further analyzed to identify overlapping characteristics. As a result of this analysis, seven new test cases were created for LFR. These test cases are summarized in Table 1. These additional verification test cases for LFR generated acceptable results and enhanced applicability of SAS4A/SASSYS-1 to LFR.

TABLE 1. Additional SAS4A/SASSYS-1 LFR-Based Verification Test Cases

<table>
<tr><th>Case</th><th>Description</th><th>Category</th></tr>
<tr><td>1.10</td><td>Base LFR Fuel Channel</td><td>Simple Steady State Cases</td></tr>
<tr><td>3.8</td><td>LFR Temperature-Dependent Coolant Density</td><td rowspan="4">Material Property Cases</td></tr>
<tr><td>3.9</td><td>LFR Temperature-Dependent Coolant Heat Capacity</td></tr>
<tr><td>3.10</td><td>LFR Temperature-Dependent Coolant Thermal Conductivity</td></tr>
<tr><td>3.11</td><td>Temperature-Dependent Built-In Lead Properties</td></tr>
<tr><td>5.28</td><td>LFR Equilibrium Temperature Distribution</td><td rowspan="2">Heat Removal System Cases</td></tr>
<tr><td>5.29</td><td>LFR Equilibrium Pressure Distribution</td></tr>
</table>

3.2. Validation Cases of SAS for Liquid Metal Reactors

Over 40 years, the SAS4A/SASSYS-1 code has been benchmarked against extensive amount of experimental data. The TREAT fuel failure benchmarking for SFR were performed for early versions of SAS4A/SASSYS-1 and these validations increased confidence of applying SAS to the metal fuel and oxide fuel.

The Reactor Vessel Auxiliary Cooling System (RVACS) is a built-in component in SAS4A/SASSYS-1 to model the cooling of reactor vessel wall using natural circulation of air during normal operation or accidents. Radiative and convective heat transfer are expected between each of the structures in the geometry of RVACS component, and the mass flow rate of the air is determined by balancing the buoyancy pressure head with the frictional pressure losses. The validation efforts showed good agreement between the SAS4A/SASSYS-1 detailed RVACS predictions [17] and a set of 1980s Natural Convection Shutdown Heat Removal Test Facility (NSTF) experimental data [18].

Numerous system-level benchmark analyses against experiments of sodium reactors were performed using SAS4A/SASSYS-1. Completed integral assessments include the EBR-II Shutdown Heat Removal Tests (SHRT) [19] and the Phénix Natural Circulation (NC) Test [20], both of which were an International Atomic Energy Agency (IAEA) Coordinated Research Project (CRP). Ongoing benchmark activities include analyses of the FFTF loss of flow without scram (LOFWOS) test.

The EBR-II SHRT benchmark activity focused on analysis of the SHRT-17 and SHRT-45R tests, which were protected and unprotected (respectively) full power loss of flow transients. The objective of SHRT-17 was to demonstrate the effectiveness of natural circulation in the reactor, while the objective of SHRT-45R was to demonstrate the ability of passive reactivity feedback to reduce the reactor power to decay heat power levels. As such, key phenomena exhibited during these tests include flow coastdown behavior, development of natural circulation flow regimes, thermal stratification in pool volumes and the Z-pipe, and inherent reactivity feedback mechanisms (primarily fuel/core expansion, Doppler, and coolant feedback effects). Beyond the CRP activity, other unprotected loss of flow and loss of heat sink tests in the SHRT series have also been analyzed using SAS4A/SASSYS-1. This includes SHRT-45, which had identical experimental conditions as SHRT-45R; SHRT-43R, a loss of flow initiated at full flow and 70% power; BOP-301, a loss of heat sink at full flow and half power; and BOP-302R, a loss of heat sink at full flow and full power.

The Phénix NC Benchmark Test, conducted as a part of the End-of-Life Test Campaign, examined a protected loss of heat sink transient from 35% power and 70% flow conditions with the objective of demonstrating the effectiveness of natural circulation in the primary system and the effect of dynamic heat rejection via secondary cooling systems. Approximately three hours following pump trips in the primary and secondary systems, heat rejection of the total system was augmented by air cooling of the steam generators, the effect of which was evident in primary and secondary system temperature conditions. Similar to the loss of flow tests in EBR-II, key phenomena exhibited during these tests included flow coastdown behavior, development of natural circulation flow regimes, and thermal stratification in sodium pool volumes. Beyond the EBR-II SHRT benchmark tests, the NC test also introduced the effects of dynamic heat rejection during the transient.

Although these benchmarking cases were performed for the testing of SFRs, the validation status can be extended to the LFR because of common phenomena of the natural circulation of coolant, coast down, coolant pool flow behavior, core kinetics, and reactivity feedback in core.

3.3. Validation Cases of SAS for LFR Fuel

The revised SAS modules for oxide fuel performance in lead coolant was developed to support the LFR systems (section 2.1.3). Extensive validations against the measured data from fuel irradiation experiments were performed to justify the models [8]. Benchmark results include normal operation fission gas release, separate effect transient fission gas release, and CABRI rapid and slow TOP tests. These cases are discussed below.

<u>Normal Operation Fission Gas Release:</u> Normal operation behaviors of RIG1, RIG2, SCARABIX, VIGGEN-4, MK-2, and LVD fuel pins were simulated with the SAS4A/SASSYS-1 code. Table 2 gives the test matrix. The testing fuel pins have a wide range of fuel burnup. SCARABIX and LVD pins have annular configuration and others have solid cylindrical geometry. Among test pins, LVD pin operated at PFR up to 23 at% without failure and without any significant clad strain, indicating the favorable performance of low smear density annular fuel designs. This benchmark results of cumulative fission gas release data at the end of life between the simulation results and experimental data showed that the accuracy is within ± 6% average difference.

TABLE 2. TEST MATRIX FOR STEADY STATE FISSION GAS RELEASE

Test name	Reactor	Fuel pellet	Clad	Peak Burnup(at%)
RIG 1	PHENIX	Solid MOX	SS316	1
RIG 2	PFR	Solid MOX	SS316	2.9
SCARABIX	SUPER PHENIX	Annular MOX	15-15Ti	6.4
VIGGEN-4	PHENIX	Solid MOX	15-15Ti	11.8
MK-2	JOYO	Solid MOX	SS316	14.4
LVD	PFR	Annular MOX	PE16	23.1

<u>Fast Transient Fission Gas Release:</u> Response of rapid heating of irradiated MOX fuel was tested at Hanford Engineering and Development Laboratory (HEDL). The MOX fuel samples were irradiated at EBR-II, then the sample in capsule was electrically heated in the laboratory and temperature and pressure transient within 20 seconds were recorded. The MOX fuel sample with 25 wt% Plutonium irradiated at EBR-II with 17.7 kW/m peak linear heat rate and 2.7 at% peak burnup was used in FGR9 experiment. In the FGR15 experiments, MOX fuel sample with 25 wt% Plutonium irradiated at EBR-II with 44.5 kW/m peak linear heat rate and 4.75 at% peak burnup were used. The comparisons of measured and predicted fission gas release produced good results with reasonable accuracy.

<u>Transient Fuel Failure:</u> The CABRI-2 experimental program with 15-15Ti cladding were used to benchmark the oxide fuel models in SAS4A/SASYS-1. Table 3 gives the test matrix. During the tests, the power ramped from typical operating conditions to 80 kW/m to 125 kW/m peak linear heat rate. Test pins were subjected to excessive fuel melting and some pins experienced cladding failure. The clad failure mechanism for these slow TOP transients was thermal creep rupture. The benchmark is intended to demonstrate that the SAS4A/SASYS-1 code can well predict 15-15Ti clad failure time and location, axial distribution of the molten fuel fraction and transient fission gas release behavior. In general, the simulation well predicted the observed behavior, such as the molten fuel radius (or radial location of the melt front) and central hole boundary at the end of the transient for

E12, PF1 and MF2 tests, respectively. The melt-front predictions remain at the conservative side for E12 and PF1 experiments and in great agreement for the MF2 test data. Small mismatches in the melt-front towards the bottom and top ends could be explained by in-pin molten fuel relocation. This mechanism was ignored in SAS4A/SASYS-1 simulations. The predicted and measured transient fission gas release fraction for E12 test is 46% and 38% respectively. Considering the large uncertainty in transient fission gas release in the test, the agreement was found reasonable.

TABLE 3. TEST MATRIX FOR SLOW TRANSIENT

Test name	Reactor	Fuel pellet	Accident	Clad Failure
AI3	PHENIX	Solid MOX	Rapid TOP	Yes
BI3	PHENIX	Solid MOX	LOF+Rapid TOP	Yes
E12	PHENIX	Solid MOX	Slow TOP	Yes
BCF1	PHENIX	Solid MOX	Slow TOP	Yes
PF1	SUPER PHENIX	Annular MOX	Slow TOP	No
MF2	SUPER PHENIX	Annular MOX	Slow TOP	No

3.4. Validation against Integral Effects Test for LFR (CIRCE-HERO Benchmarking)

The lack of validation against IETs in the V&V matrix of SAS4A/SASSYSY-1 was addressed by introducing the benchmarking against the CIRCE test. A survey of the existing lead and LBE test facilities were performed in 2019 and the survey indicated that the suitable experiments in terms of scaling, testing conditions, and facility layout similarity to the Westinghouse LFR and with available data are those performed with the CIRCE facility [18] at ENEA (Italy). The CIRCE-HERO experimental campaign [15][16] is a large scale IET that provides highly representative validation data on the establishment of natural circulation in a lead-based reactor system. The primary working fluid is LBE and the secondary working fluid in HERO is water/steam. A number of different experiments including Protected Loss of Flow (PLOF) and Protected Loss of Heat sink (PLOH) accidents were performed. The test data are suitable to benchmark safety analysis computer codes.

Though the CIRCE facility was not directly scaled down from the Westinghouse LFR, the reactor layouts are consistent, and the volume scale of CIRCE is sufficiently large for the LFR validation. Therefore, the testing data from the CIRCE testing is applicable to assess the SAS code for the LFR safety analysis. The CIRCE benchmarking increased applicability of the SAS code to the LFR systems.

The CIRCE-HERO layout is shown in Figure 2 together with the SAS model. The SAS model consists of a set of Compressible Volumes (CV) and Segments (S) representing the fuel pin simulator, riser and steam generator. The LBE pool is divided into two compressible volumes CV1, the inlet plenum, and CV4, the upper cold pool. The fuel bundle simulator is represented using a core channel which contains three lower reflectors, an active fuel region and one upper reflector. Among three CIRCE-HERO tests, loss of flow case, Test 3, was modeled and benchmarked because of its core flow behavior and partial flow in the secondary side to provide heat removal to maintain natural circulation.

In general, the nature circulation flow in the CIRCE loss of flow case was predicted reasonably well, while the transient from the force flow to nature circulation needed improvements [21]. Future improvement is expected to focus on modeling the steam generator explicitly and addressing thermal stratification in the upper and lower cold pools. In general, the benchmarking addresses the lack of IET in the SAS validation matrix.

3.5. Verification and Validation of GOTHIC

The GOTHIC code qualification report by EPRI encompasses phenomena which support the LFR PHRS analyses and a multitude of other applications [10]. Qualification is established through the comparison of GOTHIC predictions to solutions of analytic problems and to experimental data for containment, and related applications. The V&V tests can be classified as analytic, separate effects or combined effects. The analytic tests compare the code results to generally accepted analytical solutions to the governing physical laws or to a correlation fit to experimental data. The separate effects tests have primary focus on a single phenomenon with other phenomena possibly playing minor roles. The combined effects tests measure selected system parameters in tests where multiple phenomena play significant roles, possibly at different times in a transient test. Some of the validation tests also represent comparisons with multiple test runs in an experimental program designed to study various aspects of a particular problem.

The qualification report describes 83 separate experiments and test problems. Some of the applicable phenomena to PHRS analysis are listed as follows:

- Pressure Drop (Single Phase, Bubbly Flow, Film-Drop Flow)
  - FLECHT SEASET Natural Circulation Tests
- Thermal Convection (Natural, Forced, and Mixed)
- Thermal Conduction in Solids
- Thermal Diffusion (Vapor and Liquid)
- Thermal Radiation
  - Conductor Surface-to-Surface Radiation, analytic solution
- Condensation on Walls
- Liquid Hold Up in Vertical Flow
- Boiling Heat Transfer
- Pool Boiling
- Pool Heat Transfer
- Pool Surface Evaporation
- Natural Circulation

The qualification report includes a test matrix to map the important phenomena to different experiments and analytical problems. There are also forty standard problems executed with each new code release to ensure the code continues to adequately predict results while using basic physical modeling capabilities and some component functionality.

Additional GOTHIC validation against the LFR separate effect test on the PHRS thermal hydraulics is ongoing as discussed in the next section.

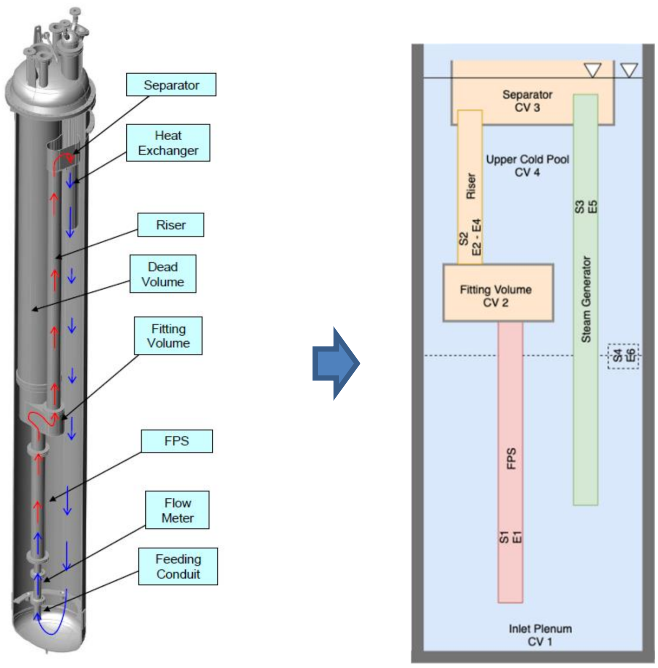

FIG. 2. CIRCE facility and the SAS model.

4. PLANNED VERIFICATION AND VALIDATION CASES

The LFR testing program aims to fill the technology gaps in the key materials, components or systems of the LFR plant. It provides demonstrations of the LFR engineering that reduces the uncertainty of the LFR design and provides experimental data for the V&V of the safety analysis computer codes. With the support of LFR PIRT, the phenomena important to plant safety with insufficient state of knowledge were identified and LFR testing plan was developed to fill these gaps. The LFR testing program provides highly LFR-applicable testing data for the validation of SAS4A/SASSYS-1 and GOTHIC.

Since 2018, Westinghouse has been participating in the Advanced Modular Reactor (AMR) program funded by the UK Department for Business, Energy & Industrial Strategy (BEIS). As a part of AMR Phase 2 program, eight state-of-the-art test facilities were designed and installed to support the development of the Westinghouse LFR. Among them, passive heat removal facility (PHRF) and versatile loop facility (VLF), are major thermal hydraulic testing facilities, and the experimental data will to be used for the validation of SAS4A/SASSYS-1 and GOTHIC.

The purpose of PHRF is to assess the thermal hydraulic performance of the PHRS, and provide thermal hydraulic data for the validation of modeling and simulation tools to model PHRS. The facility was designed following the principle of power-to-volume scaling method. The power density is kept constant, the volumes were reduced according to the thermal power, the heights were prototypical, and the operating pressure was prototypical. This scaling approach correctly preserved various physical phenomena, natural circulation and mass transport in general, flow patterns, type of heat transfer, and time scale. The 3D model of the 500kW PHRF at the Ansaldo UK site is illustrated in Figure 3. The data from PHRF will be used to validate the GOTHIC code in the prediction of the PHRS performance.

The VLF (Figure 3) is intended to test thermal hydraulic performance of fuel rods mockup and the subscale primary heat exchanger in a loop layout. The fuel rods mockup is a 19-rod, grid spacer-supported, 1.3 m active length, electrically heated rod bundle capable to reach linear powers up to 250 W/cm. This bundle mockup is aluminized to withstand operation up to 650°C. The diffusion-bonded printed-circuit heat exchanger in VLF is fed by the two operating fluids, i.e., liquid lead and supercritical water, prototypically reproducing channel configurations on both primary and secondary side. The VLF test facility is installed at the same site with the PHRF. The tests provide relevant data for the validation of the SAS4A/SASSYS-1 for the LFR prototypical fuel assembly and the primary heat exchanger, and force flow and natural circulation flow in a lead loop layout as well as the transition from forced flow to natural circulation flow.

Beside VLF, by leveraging IAEA CRP, "Benchmark of Transition from Forced to Natural Circulation Experiment with Heavy Liquid Metal Loop" [22], the SAS4A/SASSYS-1 is expected to be benchmarked against the NACIE testing by ENEA on loss of flow transient with unblocked fuel bundle and partially powered fuel bundle.

The IET facility for the Westinghouse LFR is planned to support licensing the LFR and computer code V&V. The facility is expected to be a large-scale pool type testing facility with electrically heated rod bundle, printed-circuit heat exchanger, mechanical pumps, secondary loop, and a scaled PHRS. It will be operated at prototypical LFR temperature and pressure. With proper scaling, the facility could duplicate thermal hydraulic behavior expected during accidents in LFR and interaction of thermal hydraulic behvaiors among components inside the LFR. The experimental data will be used to benchmark the coupled SAS-GOTHIC computer code to enhance its applicability to the safety analysis of LFR.

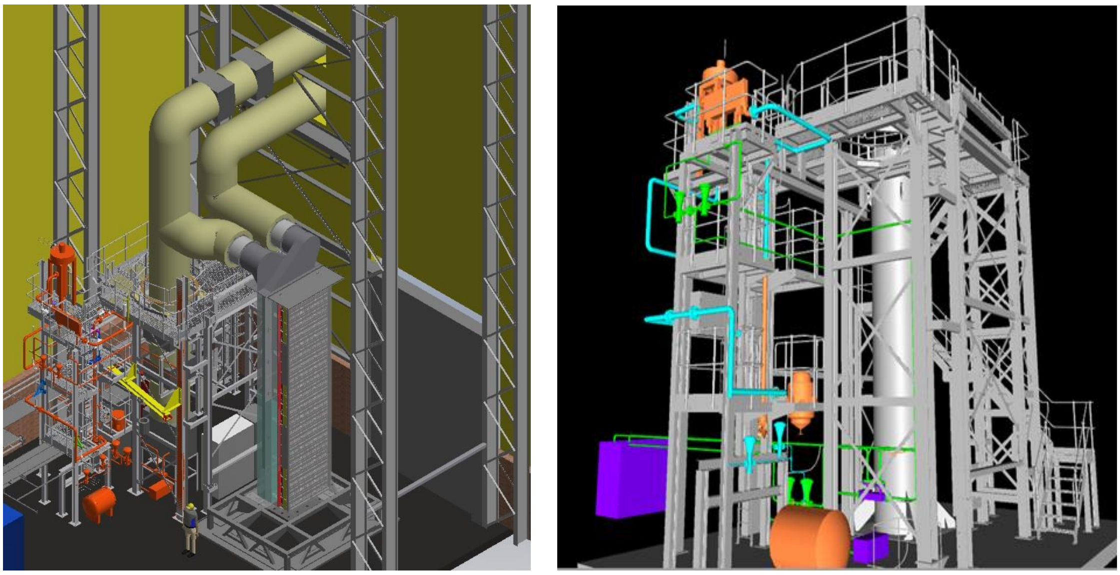

FIG. 3. 3D rendition of passive heat removal facility (left) and versatile lead loop facility (right).

5. CONCLUSIONS

Westinghouse continues to develop its next generation of high-capacity nuclear power plants based on LFR technology. As a key part of development of safety analysis for LFR, the capability of SAS4A/SASSYS-1 is further increased to address lead/LBE properties, the mechanistic source term analysis, oxide fuel modeling enhancements, the capability to simulate the PHRS through coupling with GOTHIC, and modeling hybrid microchannel-type primary heat exchanger. The applicability of SAS4A/SASSYS-1 to the LFR system is greatly enhanced by expanding the verification cases to the LFR system, development of validation cases for the LFR fuel, validation against the CIRCE integral effects test. The applicability of GOTHIC to PHRS is reviewed.

Additional V&V cases are planned to address the validation gaps identified in the LFR PIRT by leveraging the AMR phase 2 program, which generate experimental data from the VLF and PHRF facilities, and IAEA CRP on NACIE loss of flow testing. Dedicated IET for LFR is essential for the validation of SAS-GOTHIC code. With these V&V, the applicability of the SAS4A/SASSYS-1/GOTHIC safety analysis computer code to the LFR is further enhanced and its licensing readiness is increased.

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