INL/CON-22-70513-Revision-0

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August 2023

Changing the World's Energy Future

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Prepared for the
U.S. Department of Energy
Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517

Advances in Integral and Separate Effects Experiments for Water-Cooled Small Modular Reactors

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ABSTRACT
This paper focuses on the progress being made in water-cooled small modular reactor (SMR) advanced integral effects and separate effects experiments for reactor licensing. SMRs, considered modular in design, are mostly factory-built and then shipped to the reactor site. Of the several types of SMR designs available, water-cooled SMRs are likely to receive regulatory approval faster than others, as most of the technologies involved (e.g., the fuel and coolant technologies) are matured. However, the unique safety systems of SMRs, which depend on specific SMR design features, require integral and separate effects experiments to achieve reactor licensing. Thus, of the many SMR designs being proposed, only a few have successfully undergone licensing and reached the final development and demonstration stage. Many nuclear vendors and newcomer companies are investing millions of dollars to develop integral and separate effects testing facilities for preparing final safety analysis reports to include in licensing applications. Development and analysis of these experimental facilities is costly and takes about four to five years. The unique challenges involved can be reduced when stakeholders synergistically apply lessons learned, knowing the critical role played by advancements in experiments that support the licensing of SMR safety systems. Identification of knowledge/research gaps with the phenomena identification and ranking table (PIRT) and designing experimental facilities focusing on the phenomena of interest (POI) and figures of merit (FOMs) are pivotal to select the critical path to successful design demonstration and licensing application.

KEYWORDS
Integral effects test (IET), separate effect test (SET), scaling, small modular reactor (SMR)

1. Introduction

According to the International Atomic Energy Agency (IAEA), small modular reactors (SMRs) are nuclear reactors with a power capacity per unit up to 300 megawatt electrical (MWe) [1]. By comparison, gigawatt-size reactors can produce electrical outputs up to 1500 MWe. These SMRs are: (a) smaller in size than conventional nuclear power reactors; (b) modular (e.g., mostly factory-assembled power generating units and transported as a unit to plant site); and (c) actual reactors (e.g., they use nuclear fission to generate heat to electricity generation and process heat applications) [1].

There have been over 80 proposals for SMR designs worldwide, thereby reflecting both their potential and a variety of possible nuclear reactor designs [2]. However, to prove the conceptual design needs supportive computer code analysis and experiments. A new reactor design and development process requires component-level and system-level experiments and analysis [3]. Integral effects test (IET) and separate effects test (SET) facilities are used to perform various reactor system transient and accidental experiments for computer code validation and qualification [4–5]. IET experiments involve the testing of the entire reactor system, while SET experiments focus on individual components or subsystems [6].

An IET facility includes all the major components of a reactor system, such as the reactor core region simulated with electrical heaters, the reactor pressure vessel (RPV), pressurizer (PZR), steam generator (SG), accumulators, and containment. However, SET facilities should support counterpart tests, which require detailed studies of the process phenomena. The most common reactor system SET experiments are: (a) containment SET for steam condensation studies; (b) SG SET for studying SG performance with level swelling, tube rupture, and flow instabilities; (c) reactor core (rod bundle) SET for core thermal-hydraulics, critical heat flux, and departure from nucleate boiling (DNBR) studies; (d) fuel system irradiation experiments with prototypic reactor conditions; and (d) reactor coolant pump (RCP) SET to validate pump performances.

The development of the SET and IET facilities start with phenomena identification and ranking table (PIRT) studies, which identify the phenomena of interest (POI), rank the POIs' importance to the figure of merit, and rank the state of knowledge (SOK) of the POI for the specific reactor design parameters. Then, PIRT studies are needed for a review of the previous reactor system development programs to check the required data availability for assessing the evaluation models (EMs). It is pivotal to ensure data supports the Evaluation Model and Development Assessment Process (EMDAP) by the United States (U.S.) Nuclear Regulatory Commission (NRC) Reg Guide 1.203 [7]. If the data are not available to support EMDAP, then it is required to perform IET and SET experiments to obtain the necessary data to support reactor design. Historically, IET and SET facilities were developed based on scaling to reduce cost and ensure safety. Scaled-down test facility design requires scaling analysis while keeping data scalability from perspectives such as geometry, properties, and phenomena (e.g., event timing, order). The data obtained from the IETs and SETs are considered first category data [8]. Data obtained from the literature or data generated using the standard handbook are considered second category data [8].

The safety of a nuclear reactor is of utmost importance, as its design must take all possible scenarios into account—including normal operation, design-basis accidents (DBAs), and beyond-design-basis accidents (BDBAs). The coupling of thermal, neutronic, and structural systems must be carefully considered when setting design limits to ensure the reactor remains safe and stable under all conditions [9]. The reactor also must be designed with multiple barriers to prevent the release of radioactive materials and general safety criteria must be established to ensure the integrity of these barriers and minimize the risk of harm to people and the environment. However, the IET and SET facilities for thermal-hydraulics (TH) experiments are non-nuclear experiments (i.e., no nuclear/radioactive materials are used). Therefore, it is important to design the IET and SET facilities in a supportive way that matches the conceptual design. Electrical heater rods with a similar geometry size to fuel rods are used in the IET as the heat source. Similarly, a cooling tower or adequate capacity chillers are used as heat sinks to condense steam and return it to the feed water system as part of the closed loop for the reactor cooling water.

Over the past 50 years, several standard IET facilities have been developed worldwide to design and license different reactor designs. These include the SemiScale and Loss of Fluid Test (LOFT) at Idaho National Laboratory (INL), the Advanced Plant Experiment (APEX) at Oregon State University (OSU), the Purdue University Multi-Dimensional Integral Test Assembly (PUMA), and the Full Length Emergency Cooling and Heat Transfer (FLECHT) system in the U.S.; the Rig of Safety Assessment (ROSA)/Large Scale Test Facility (LSTF) and Cylindrical Core Test Facility (CCTF) of a pressurized water reactor (PWR) in Japan; the Simulatore per Esperienze di Sicurezza (SPES) in Italy; the Boucle d'Etudes Thermohydrauliques Système (BETHSY) in France; the Parallel Channel Test Loop (PKL) facility in Germany; the Advanced Thermal-hydraulic Test Loop for Accident Simulation (ATLAS) in Korea; and the Advanced Core-cooling Mechanism Experiment (ACME) in China [3–4, 10–12]. These programs and facilities were developed for targeted commercial light water reactors (LWRs), like PWRs and boiling water reactors (BWRs), to examine reactor safety issues related to plant response during a loss-of-coolant accident (LOCA) and operational transient. Few IET facilities have been developed to target SMRs in recent years. The design of IET facilities differs depending on the reactor heat generation in the fuel, power density, coolant, mode of operation, passive safety system, etc.

2. PIRT and Reactor Accident Progression

A PIRT study is an important element for the EMDAP which is used to ensure that the computer code properly simulates the behavior of the system. Understanding the reactor accident progression is important during the PIRTs, because the reactor accident scenarios are broken down into phases, and assess POIs for each phase.

2.1 PIRT Studies

In the PIRT process, experts gather information about a specific reactor design concept and rank its' importance according to a decision-making objective. A PIRT study identifies the POI and ranks the phenomena with different SOKs and FOMs. This study determines the level of analysis and testing required to demonstrate the safety of a reactor by identifying the potential consequences of different events. PIRT studies typically involve the following steps [13]:

- Motivation: Define the issue driving the need for a PIRT.
- PIRT Objectives: Define the specific objectives of the PIRT.
- Database: Compile and review the background information that captures relevant knowledge.
- Hardware-scenario: Specify plant and components; divide scenario into phases.
- FOM: Select key FOM used to judge importance.
- Phenomena Identification: Identify all plausible phenomena plus definitions.
- Importance Ranking: Assign importance relative to FOM; document the rationale.
- Knowledge Level: Assess the current level of knowledge regarding each phenomenon.
- Document PIRT: Document the effort with sufficient coverage that a knowledgeable reader can understand the process and outcome.

The PIRT studies set experimental data requirements for the high-ranked POI with low/medium SOK, especially for accident scenarios. Some of the most challenging accident scenarios considered are breaks in piping connected to the reactor coolant system (RCS) pressure boundary. These breaks consist of POIs that are tightly coupled together and strongly dictate the accident progression.

2.2 Reactor Accident Progression

Reactor accident progression varies with reactor system design. A few PWR-type reactors, for example, the AP1000 and SMR-160 are equipped with an automatic depressurization system (ADS) to depressurize the RCS following a LOCA to inject cooling water by passive safety systems which differs from pumped injection flow of traditional emergency core-cooling systems (ECCS). The ADS releases coolant from the RCS to containment in the form of steam. Steam condenses in the containment—mostly on the containment wall—and accumulates in the containment [14]. However, the initiation of the ADS also depends on the RCS break size and location. There are several piping breaks in a RCS are possible, such as a main steam line break (MSLB), steam generator tube rupture (SGTR), and direct vessel injection (DVI) line break, and a break in the RCP lines.

Likewise, there are various LOCA scenarios that are based on break size: (1) a small-break LOCA (SBLOCA); (2) a medium-break LOCA (MBLOCA); and (3) a large-break LOCA (LBLOCA). In the early stage of reactor development for Generation I and Generation II reactors, LBLOCAs were prioritized. The SBLOCA, however, proves challenging as the reactor accident advances through several stages. An SBLOCA accident involves subcooled blowdown, saturated natural circulation, ADS depressurization, in-containment refueling water storage tank (IRWST) injection, and long-term cooling. These accident phases can be identified by analyzing the pressure trend versus the time progression of the primary coolant system (PCS), as presented in Figure 1. Developing an IET facility that covers required and targeted accident phases is crucial to reactor licensing applications.

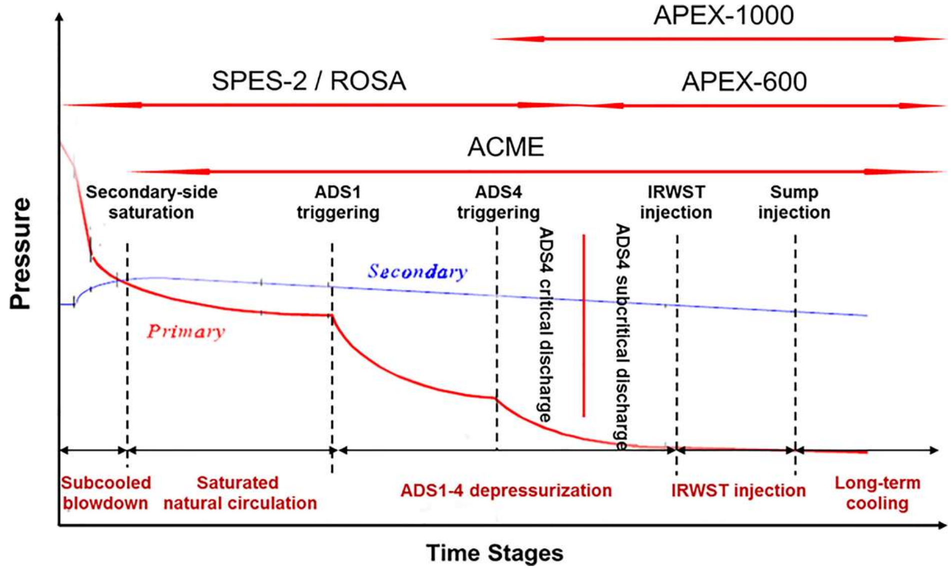

Figure 1. Scoping experiments comparison between selected IET facilities [10].

An SB LOCA event follows a series of events, as shown in Figure 1 [10], which include subcooled blowdown, saturated natural circulation, ADS initiation, IRWST injection, and long-term cooling. It is possible to identify the phases by comparing the pressure trends with the time progression of the PCS. The IET facilities should cover all required and targeted accident phases. Testing plans and test matrixes are required to obtain data from the experimental facilities according to the phases and phase time spans of the reactor accident scenarios [10].

3. Scaling Approaches and Considerations

Scaling analyses are required to develop an IET (i.e., scaled facility) to provide data for computer code assessment and validation, considering whether distortions in high-ranking phenomena are acceptable. It is essential to keep in mind that an IET is not a simulator, but a similar IET with H-ranking phenomena is useful for reactor transient analysis.

The scaling for reactor system thermal hydraulics considers both single- and two-phase conditions, which are important for LWR accident analysis. When considering facility size, there are many factors to think about, such as: (a) available space and scale relations of existing facilities; (b) the need to compensate for shortcomings in existing facilities; and (c) justifiable rationale and the impact on total cost. Scaling parameters include geometric parameters (e.g., length, diameter, area, volume), time, velocity, power, power-volume, and acceleration. Traditionally scaling parameters are expressed using scaling ratios. The ratio of the nondimensional numbers in the scaled-down model and prototype should be close to unity to maintain and preserve phenomenology, such as flow and heat transfer dynamics, as shown in Equation (1):

$$\psi_R = \frac{\psi_{model}}{\psi_{prototype}} = \frac{\psi_m}{\psi_p} = 1 \tag{1}$$

where the subscript R indicates the ratio of the respective nondimensional numbers.

3.1 Scaling Models for Reactor System (Single-Phase and Natural Circulation)

The ideal scaled facility is not realistic or practical. However, it provides the preliminary basis for scaling analysis. It considers: (a) full pressure, prototypic fluid; (b) all materials to be the same as the model and the prototype; and (c) the same geometric ratios between the scaled facility and the prototype. The geometric ratios consist of: (a) length ratio—to match available laboratory space, which determine velocity ratio; (b) area ratio—to match commercially available pipe; and (c) volume ratio—from length and area ratios, which determine power ratio. The scaling analysis of a reactor involve governing equations, models, correlations and nondimensional numbers, as presented in Table 1 [15].

Table 1. The governing models, equations, and variables related to reactor system scaling analysis.

Governing models	Equations and Variables
1-D loop momentum (Each term represents mass flux times velocity or momentum rate of change per unit area or force per unit area)	1-D loop momentum balance equation for 1-Φ natural circulation $$\sum_{i=1}^{N}\left(\frac{l_i}{a_i}\right)\cdot\frac{d\dot{m}}{dt}=\beta g\rho_l(T_H-T_c)L_{th}-\frac{\dot{m}^2}{\rho_l a_c^2}\sum_{i=1}^{N}\left[\frac{1}{2}\left(\frac{fl}{d_h}+K\right)_i\left(\frac{a_c}{a_i}\right)^2\right]$$ where the *i* subscripts refer to the *ith* component and L_{th} is the thermal center length. The core cross-sectional flow area, a_c, is used as the reference flow area.
1-D energy equation (Each term in this equation has dimensions of power)	$$C_{vl}M_{sys}\frac{d(T_M-T_C)}{dt}=\dot{m}C_{pl}(T_H-T_C)-q_{SG}-q_{loss}$$ Rate of change of thermal energy = heat generation and heat losses.
Loop time constant	$$\tau_{loop}=\sum_{i=1}^{N}\frac{l_i}{u_i}=\sum_{i=1}^{N}\tau_i=\frac{M_{sys}}{\dot{m}_o}=\frac{M_{sys}}{\rho_l u_{co}a_c}$$
Loop time constant and reference length number	$$\tau_{loop}=\sum_{i=1}^{N}\frac{l_i}{u_i}=\sum_{i=1}^{N}\tau_i=\frac{M_{sys}}{\dot{m}_o}=\frac{M_{sys}}{\rho_l u_{co}a_c}$$ $$\Pi_{\mathrm{L}}=\sum_{i=1}^{N}\frac{l_i}{l_{ref}}\frac{a_c}{a_i}\text{ , where } l_{ref}=\frac{M_{sys}}{\rho_l a_c}$$
Nondimensional number (Richardson number)	$$Ri=\frac{\beta g(T_H-T_C)_o L_{th}}{u_{co}^2}=\frac{\beta g q_{co}L_{th}}{\rho_l a_c C_{pl}u_{co}^3}$$ These two forms of Ri comes from an energy balance across the core $$q_{co}=\rho_l a_c u_{co}C_{pl}(T_H-T_c)=\dot{m}_{co}C_{pl}(T_H-T_C)$$ $$Ri=\frac{\beta g\Delta TL_c}{u^2}=\frac{\frac{g\beta\Delta TL^3}{v^2}}{\left(\frac{uL}{v}\right)^2}=\frac{Gr}{Re^2}=\frac{u_o^2}{u^2}=\frac{u_o^2}{u_{co}^2}$$ For the reactor system natural circulation, $$u_o=\sqrt{\beta g(T_H-T_C)L_{th}}\ and\ u=u_{co}$$ Therefore, core inlet velocity, $u_{co}=\left(\frac{\beta q_{co}L_{th}g}{\rho_l a_c C_{pl}\Pi_{Fl}}\right)^{1/3}$.

Governing models	Equations and Variables
Loop energy, heat transfer, and heat loss ratios	$\Pi_T = \frac{(T_H - T_C)_o}{(T_M - T_C)_o}$ and $\Pi_{SG} = \frac{q_{SGo}}{\rho_l u_{co} a_c C_{pl} (T_M - T_c)_o}$ $\Pi_{Loss} = \frac{q_{loss,o}}{\rho_l u_{co} a_c C_{pl} (T_M - T_c)_o}$ Where, Π_{SG}, represents the ratio of the other system heat losses to the core heating; Π_{Loss}, $r$epresents the ratio of the heat transfer to the steam generator to the core power heat input.
Diameter, area, and volume scale ratios	$(d_h)_R = (u_{co}^2)_R$, $(a_i)_R = \left(d_h^2\right)_R$ and $(V_i)_R = (a_i l_i)_R$ Reference velocity is the core inlet velocity and reference length is the heated height of the core.
Mass flow rate scale ratio, and power-to-volume ratio	$\dot{m}_R = (u_{co} a_c)_R$ and $(q_{co}/V_i)_R$ Where, we consider fluid property similitude.

The significance of the scaling models are as follows:

- The loop momentum balance equation and loop reference length number provide a primary mechanism of core heat removal during normal operation and certain accident scenarios and also provide a basic for scaling analysis of the IET.
- The loop energy equations provide a rate of change of thermal energy that is equal to the sum of heat generation and heat losses.
- Loop-time constant provides a justification of using non-dimensionalized loop momentum and energy equations.
- Richardson number (Ri) represents the ratio of buoyancy forces to inertial forces, which can be expressed in terms of the Grashof numbers (*Gr*) and Reynolds number (*Re*). Ri can be simplified to the square ratio of characteristic velocity for natural convection (u_o) and local velocity (u).
- Experimental validation could be achieved through direct measurement of the core inlet velocity in the IET.
- The geometric scaling ratios and similarity criteria considered reference velocity and reference lengths. Core inlet velocity and core height (i.e., heated height of the core) are considered as reference velocity and reference height for the reactor system scaling analysis. is the core inlet velocity.

3.2 Scaling Approaches Applicable to Reactor System

The preliminary ideal scaling analysis should be evaluated with system code analysis (for example, RELAP5 model) for steady-state, transient, and accident conditions. The next step is to modify the ideally scaled IET to engineering-scale facility. Factors that need to be considered during engineering scaling to design a realistic test facility are: (a) the use of a component that can actually be obtained or built; (b) scaling ratios that should be the same as IET; and (c) using commercially available pipe diameters and thicknesses.

Several scaling approaches are in consideration for various scaling parameters, as shown in Table 2, as well as used in reactor system IETs, as shown in Table 3. The most common types are linear-scaling, power-to-volume scaling, three-level scaling, and hierarchical two-tiered scaling (H2TS), as follows [16]:

- Linear-scaling (1960s) can develop a miniature replica of the prototype, but it incurs distortion in acceleration and energy transfer and acceptability of the scaling factors are low.

- Power-to-volume scaling (1970s) preserve time and height, but have distortions in pressure drop, heat transfer, and multi-dimensional phenomena.
- Three-level scaling, as described by Ishii (1980s–1990s), reduces construction costs and focuses on the local phenomenon. However, the time-scale gets shifted and also incurs distortion for multi-dimensional phenomena.
- H2TS (1990s) maintains a scaling hierarchy for complex systems, as well an emphasis on the important phenomena; however, there is a contradiction with the scaling criteria.
- Fractional scaling analysis (FSA) method (2000s).

Table 2. Scaling parameters related to IET and SET facilities scaling analysis/methods [10].

Name	Parameter Symbol	Linear Scaling	Power/ Volume	Ishii Scaling	H2TS Scaling
Length ratio	l_R	l_R	1	l_R	l_R
Diameter ratio	d_R	l_R	d_R	d_R	$l_R f_R$
Area ratio	a_R	l_R^2	$d_R{}^2$	d_R^2	d_R^2
Volume ratio	V_R	l_R^3	d_R^2	$l_R\, d_R^2$	$l_R\, d_R^2$
Velocity ratio	u_R	1	1	$l_R^{1/2}$	$l_R^{1/2}$
Time ratio	t_R	l_R	1	$l_R^{1/2}$	$l_R^{1/2}$
Power-volume ratio	q_R''	l_R^{-1}	1	$l_R^{-1/2}$	$l_R^{-1/2}$
Power ratio	P_R	l_R^2	d_R^2	$d_R^2\, l_R^{1/2}$	$d_R^2\, l_R^{1/2}$
Acceleration ratio	g_R	l_R^{-1}	1	1	1

Table 3. Scaling methods used in IET facilities [16].

Scaling method	Facilities
Linear scaling	LOFT, SEMISCALE, LSTF, PKL, ROSA, BETHSY, LOBI
Power-to-volume scaling	PANDA, INKA, PKL, PACTEL, BETHSY, SMART, SPES, ROSA IV
Three-level scaling, Ishii	PUMA, ATLAS
H2TS	APEX, ACME

Note: ACME – Advanced Core-cooling Mechanism Experiment; APEX — Advanced Plant Experiment; ATLAS – Advanced Thermal-hydraulic Test Loop for Accident Simulation; BETHSY – Boucle d'Etudes Thermohydrauliques Système; INKA – Integral Test Stand Karlstein; LOBI – LWR Off-Normal Behavior Investigation; LOFT – Loss of Fluid Test; LSTF – Large Scale Test Facility; PACTEL – Parallel Channel Tests Loop; PANDA – PAssive Nachzerfallswärmeabfuhr und Druck-Abbau Testanlage; PKL – Parallel Channel Test Loop; PUMA – Purdue University Multi-Dimensional Integral Test Assembly; ROSA – Rig of Safety Assessment; SEMISCALE – INL IET facility; SMART – System integrated Modular Advanced ReacTor; and SPES – Simulatore per Esperienze di Sicurezza.

These IETs were scaled-down facilities from the respective prototype reactors developed to obtain data for reactor system transient and accident analysis. These data were used to validate computer model reactor behavior and demonstrate that a proposed design meets safety and performance requirements.

3.3 Review of the Previous (selected) Scaled Test Facilities for Reactor System

A brief review of the important scaled test facilities is presented in Table 4 and Table 5. Information in Table 1 covers the important scaled facilities developed for reactor system experiments, including facility design purpose, testing, and model used. Likewise, Table 2 data covers the power, power rating, pressure, and scaling methods for prototype nuclear power plant (NPP) and scaled facilities.

Table 4. IET Facilities, origin country, reference reactor and verified codes [3–4, 10–12].

Name (Activity, Country, Organization)	Prototypical NPP, (Code Validated)	Design Purpose, Testing, and Model Assessment
SEMISCALE IET (INL, U.S. for AEC, ERDA, DOE, and NRC)	PWR single, 1-1/2 loop, and 4-loop (RELAP)	The INL SEMISCALE test program considered several IET facilities with series testing numbers 500, 600, 700, 800, and 900, and Mod 1, 2A, 2B, 2C, and 3, which supported PWR accidents and operational transient analysis. Test datasets were used for RELAP5 model validations and supported other codes like RELAP4 and TRAC.
LOFT (Inactive, INL, U.S., NRC; OECD-partnered)	Four-loop PWR, (TRACE and MELCOR)	The only IET facility used real fuel rods, where Best Estimate Plus Uncertainty (BEPU) and sensitivity analysis was conducted. Utilized for RELAP, TRAC, TRACE, RETRAN, and MELCOR models. Specifically used for RELAP5 and partially used for ASTEC.
ROSA (Inactive, Japan, JAERI)	AP600, (TRACE and RELAP5)	Evaluated the passive safety system of AP600. Verified TRACE and RELAP5 for SBLOCA and SGTR.
PUMA (Active, U.S., Purdue)	SBWR/ ESBWR, (RELAP5 and TRACE)	Mainly developed for BWR and proved independent data to NRC. Supported TRACE and RELAP5 models.
ATLAS (Active, Korea, KAERI)	APR1400, (MARS, RELAP5, and ATHLET)	Developed for transient testing of PWR, including DVI line break. Supported MARS code development. Used for MARS, RELAP5, and ATHLET models.
SMART-ITL (Active, Korea, KAERI)	SMART, (TASS/SMR-S, MARS, and RELAP)	Evaluate the SMART design and passive safety performance. Used for MARS-KS and Transient and Set-point Simulation/Small and Medium (TASS/SMR-S).
SPES-2 (Active, Italy, ENEA/ENEL)	AP600, (RELAP5, SPESAN)	Evaluated the AP600 design, reactor accidents, and passive safety system performance. Used for RELAP5 and SPESAN models.
APEX (Inactive, U.S., OSU)	AP600/ AP1000, (RELAP5 and TRACE)	Used the H2TS scaling method to develop facilities. Conducted reactor accidents and transient tests, including CCFL, PZR surge line break, and U-tube steam condensation in SG. Used for TRACE and RELAP5 models assessment.
ACME (Active, China, SNPTC)	CAP1400, (RELAP5 and NOTRUMP)	Used the H2TS scaling method. Experiment to support licensing of CAP1400, including LOCAs, ACC nitrogen injection, DVI, CCFL, and IRWST. Validate RELAP5 models.
PANDA (Active, Switzerland, PSI)	SBWR/ ESBWR, (GOTHIC and CFD)	Supported heat removal studies and containment response. Used for GOTHIC and CFD models for containment test data with hydrogen distribution. The counterpart test is ISP-42 PUMA.
INKA (Active, Germany, CQL)	KERENA, (OpenModelica)	Evaluated the performance of the BWR system, including reactor LOCAs testing and a few complementary SET tests. Used to verify and validate OpenModelica and numerical models.
PKL (Active, Germany, AREVA NP)	PWR, (RELAP5, CATHARE, and TRACE)	Investigate 4-loop PWR system performance and accident scenarios. Used for RELAP5, CATHARE, and TRACE models.
PACTEL (Active, Finland, VTT Energy)	VVER-440, (APROS, RELAP5, CATHARE)	Developed to model TH behavior of VVER-440, systems (like passive safety injection), postulated accidents (i.e., SBLOCA, MSLOCA) and major components, and accessing TH codes (e.g., APROS, RELAP5, CATHARE).
BWXT-IST (Active, Liberty University, U.S.)	mPower SMR (RELAP5-3D)	Evaluated reactor transient and accident testing, including flow visualization and experimentation. Used for RELAP5-3D model assessment and validation.

Name (Activity, Country, Organization)	Prototypical NPP, (Code Validated)	Design Purpose, Testing, and Model Assessment
NIST, NuScale (active)	NuScale, (GOTHIC and RELAP5)	Evaluated reactor system TH conditions during normal operation and PSS response during transients on GOTHIC and RELAP5 models.
MASLWR (OSU, U.S.)	NuScale, (GOTHIC and RELAP5)	Evaluated the reactor system with possible accident scenarios and safety performance testing on GOTHIC and RELAP5 models.

Note: ACME – Advanced Core-cooling Mechanism Experiment; AEC – Atomic Energy Commission; APEX — Advanced Plant Experiment; ASTEC — Accident Source Term Evaluation Code; ATLAS – Advanced Thermal-hydraulic Test Loop for Accident Simulation; DOE – U.S. Department of Energy; ERDA – Energy Research and Development Administration; INL – Idaho National Laboratory; INKA – Integral Test Stand Karlstein; JAERI – Japan Atomic Energy Research Institute; KAERI – Korea Atomic Energy Research Institute; LOFT – Loss of Fluid Test; MASLWR – Multi-Application Small Light Water Reactor; NIST – National Institute of Standards and Technology; NRC – U.S. Nuclear Regulatory Commission; OECD – Organization for Economic Cooperation and Development; OSU – Oregon State University; PACTEL – Parallel Channel Tests Loop; PANDA – PAssive Nachzerfallswärmeabfuhr und Druck-Abbau Testanlage; PKL – Parallel Channel Test Loop; PUMA – Purdue University Multi-Dimensional Integral Test Assembly; PSS – passive safety shutdown; RELAP – Reactor Excursion and Leak Analysis Program; ROSA – Rig of Safety Assessment; SEMISCALE – INL IET facility; SMART-ITL – System integrated Modular Advanced ReacTor Integral Test Loop; and SPES – Simulatore per Esperienze di Sicurezza.

Table 5. Summary of the features of the main integral TH test facilities [3-4, 10-12].

Name (loop number)	Prototype			Scaled IET		
	Power (MW)	Power rating	P (MPa)	Type	Power (MW)	P (MPa)
LOFT (2)	3000	full		P/Vol.	50	15.5
LOBI (2)	3773	full			5	15.5
ROSA 1V	480	reduce	20	P/Vol.	10	16
PUMA SBWR	40	full	1	3-level	0.2	1.05
ATLAS (2)	400	reduce	17.5	3-level	2	16
SMART-ITL	330	full	15	P/Vol.	2	18
SPES (3)	2775	full			9	20
SPES-2	3555	full	20	P/Vol.	9	20
APEX AP1000	162	reduce	2.8	H2TS	6	2.76
APEX AP600	96	reduce			1	
ROSA AP600	300	reduce			10	16
ACME	4040	full		H2TS	4.2	10
PACTEL	1375	full	12.3	P/Vol.	1	8
BETHSY (3)	300	reduce	16	P/Vol.	3	17.2
BWXT-IST	425	full		H2TS	1.7	
NIST-1	150	module		DSS	0.406	
MASLWR	150	reduce	7.6	H2TS	0.6	7.6

Note: ACME – Advanced Core-cooling Mechanism Experiment; APEX — Advanced Plant Experiment; ATLAS – Advanced Thermal-hydraulic Test Loop for Accident Simulation; BETHSY – Boucle d'Etudes Thermohydrauliques Système; LOBI – LWR Off-Normal Behavior Investigation; LOFT – Loss of Fluid Test; LSTF – Large Scale Test Facility; MASLWR – Multi-Application Small Light Water Reactor; PACTEL – Parallel Channel Tests Loop; PANDA – PAssive Nachzerfallswärmeabfuhr und Druck-Abbau Testanlage; PKL – Parallel Channel Test Loop; PUMA – Purdue University Multi-Dimensional Integral Test Assembly; ROSA – Rig of Safety Assessment; SEMISCALE – INL IET facility; SMART-ITL – System integrated Modular Advanced ReacTor Integral Test Loop; and SPES – Simulatore per Esperienze di Sicurezza.

4. SMR-focused Test Facilities and Challenges Addressed

The SMR-focused IET facilities are mainly for water-cooled SMRs, considering both integral PWR-type and BWR-type [3–4, 10, 12, 17, 18]. The IET facilities for SMRs are mostly first-of-a-kind (FOAK) facilities requiring custom design and development. The following IET facilities are evaluated specifically for water-cooled SMRs:

a. SMART IET facility Experimental Verification by Integral Simulation of Transients and Accidents-Integral Test Loop (VISTA-ITL) is used for experimentations to support safety analysis and code assessment for standard design approval. It supported safety analysis tests like SBLOCA, a complete loss of RCS flow, passive residual heat removal, and natural circulation flow. Additional tests included reactor core internal flow distribution and pressure drop, ECCS performance, and critical heat flux tests.

b. NIST-1 was developed with a physically scaled, electrically heated core with 56 heater rods that generated 406 kW thermal power to assess the NuScale power module. Testing addressed TH conditions during normal operation and passive safety systems (PSS) response during transients. It is an integral PWR system consisting of an RPV, PZR, containment and cooling pool vessels, and all passive safety cooling systems.

c. The MASLWR test facility—an integrated design with the reactor core, SG, and PZR housed in a single steel pressure vessel—at OSU was developed to support the development of the NuScale SMR [12]. The removal of containment heat was modeled in MASLWR using two separate vessels. One vessel represented the suppression pool and the condensing surface, whereas the other represented the cooling pool outside.

d. BWXT-IST was developed for testing and verification of the mPower SMR concept. It includes primary and secondary systems, with full feedwater and steam systems. It used the three-level scaling approach with the H2TS. Tests included high-pressure and low-pressure pumps. The high-pressure pump provides flow for the letdown and makeup of RCS; however, the low-pressure pump provides flow for long-term cooling.

Reactor system analysis using experimental facilities and system code models exhibits the following challenges that need to be addressed adequately:

- **Keeping the scaling consistency in the model to prototype:** Having the scaling consistency are supportive to obtain test data that incorporate the scaled IET transient and aligned with the prototypic models. Besides, having a same time-ratio between IET and prototype are supportive.
- **Minimizing scaling distortion:** Distortion of the model transient analysis can be reduced by identifying the component or phenomenon most responsible for the distortion. Scaling distortion incurred due to the low scalability perspectives of geometry, physical properties, and phenomena (i.e., event timing and order).
- **Model Verification and Validation (V&V):** Model results are based on system code-defined physics, models, and correlations. The performance of system code models depends on data availability, scalability, and adequacy. It is necessary to use the best estimate method to V&V both the models and the results.

5. Conclusion

This study discusses the research advances in scaled facilities development for water-cooled SMR system design and analysis with the following observations and findings:

- An overview of the PIRT studies that identify POI and ranked phenomena with SOKs and FOMs is provided.

- Scaling approaches and considerations with pros and cons for developing appropriate scaled facilities for experiments are discussed.
- A review of the scaled facilities developed for SMR systems and corresponding scaling modes are presented.
- A discussion regarding finding the limitations and prospective solutions for the test data and computer models is also provided.

Acronyms

1D	one-dimensional
3D	three-dimensional
ACME	Advanced Core-cooling Mechanism Experiment
ADS	automatic depressurization system
AEC	Atomic Energy Commission
AP-1000	Advanced Passive 1000 (pressurized water reactor)
APEX	Advanced Plant Experiment
APR	Advanced Pressurized water Reactor
APROS	Advanced PROcess Simulator (computer code)
AREVA	French multinational NPP designer (Company)
ASME	American Society of Mechanical Engineers
ASTEC	Accident Source Term Evaluation Code
ATHLET	Analysis of Thermal Hydraulics of Leaks and Transients
ATLAS	advanced thermal-hydraulic test loop for accident simulation
BDBA	beyond-design-basis accident
BEPU	best estimate plus methods uncertainty and sensitivity evaluation
BETSHY	Boucle d'Etudes Thermohydrauliques Système
BWR	boiling water reactor
BWXT	BWX Technologies, Inc. (Company)
CAP	China Advanced PWR (Company)
CATHARE	Code for Analysis of TH during an Accident of Reactor & Safety Evaluation
CCFL	counter-current flow limitation
CCTF	Cylindrical Core Test Facility
CFD	computational fluid dynamics
DBA	design-basis accident
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling
DOE	U.S. Department of Energy
DVI	direct vessel injection
ECCS	emergency core cooling system

EMDAP	Evaluation Model and Development Assessment Process
ERDA	Energy Research and Development Administration
ESBWR	European Simplified Boiling Water Reactor
FIST	full integral simulation test
FLECHT	Full Length Emergency Cooling and Heat Transfer
FOAK	first-of-a-kind
FOM	figures of merit
FSA	fractional scaling analysis
H2TS	hierarchical two-tiered scaling
IAEA	International Atomic Energy Agency
IET	integral effects test
INKA	INtegral test facility KArlstein
INL	Idaho National Laboratory
IRWST	in-containment refueling water storage tank
ISP	International Standard Problem
IST	integral separate test
ITL	integral test loop
JAERI	Japan Atomic Energy Research Institute
KAERI	Korea Atomic Energy Research Institute
LBLOCA	large break loss of coolant accident
LOBI	LWR Off-Normal Behavior Investigation
LOCA	loss of coolant accident
LOFT	Loss of Fluid Test
LSTF	Large Scale Test Facility
LWR	light water reactor
MARS	Modular Accident Response System
MASLWR	multi-application small light water reactor
MBLOCA	medium-break loss-of-coolant accident
MDNBR	minimum departure from nucleate boiling ratio
MELCOR	Methods for Estimation of Leakages and Consequences of Releases
MSLB	main steam-line-break
MWe	megawatt electrical
NIST	NuScale Integral System Test
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
OECD	Organization for Economic Co-operation and Development
OSU	Oregon State University

PACTEL	parallel channel test loop
PANDA	Passive Nachwarmeabbfiihr and DrtickAbbau Test Anlage
PCCS	passive containment coolant system
PCS	primary coolant system
PIRT	phenomena identification and ranking table
PKL	Primaer-KreisLauf
POI	phenomena of interest
PSI	Paul Scherrer Institute
PSS	passive safety system
PUMA	Purdue University Multidimensional Integral Test Assembly
PWR	pressurized water reactor
PZR	pressurizer
RCP	reactor coolant pump
RCS	reactor coolant system
RELAP	Reactor Excursion and Leak Analysis Program
RETRAN	RELAP4 – TRANsient
ROSA	Rig of Safety Assessment
RPV	reactor pressure vessel
SBLOCA	small break loss of coolant accident
SBWR	simplified boiling water reactor
SEMISCALE	INL integral effects test facility
SET	separate effects test
SG	steam generator
SGTR	steam generator tube rupture
SMART	system-integrated modular advanced reactor
SMART-ITL	SMART integral test loop
SMR	small modular reactor
SOK	state of knowledge
SPES	Simulatore PWR per Esperienze di Sicurezza
TASS/SMR-S	transient and set-point simulation/small and medium
TH	thermal hydraulics
TRACE	TRAC/RELAP Advanced Computational Engine
U.S.	United States
V&V	verification and validation
VISTA	experimental verification by integral simulation of transients and accident
VVER	Vodo-Vodyanoi Energetichesky Reactor

Nomenclature

a_c	cross-sectional area of core (m^2)
a_i	cross-sectional area of *ith* section (m^2)
C_{pl}	constant pressure specific heat of liquid (J/kg K)
C_{vl}	constant-volume specific heat (J/kg K)
d_h	hydraulic diameter (m)
f	Darcy friction factor
g	acceleration due to gravity (m/s^2)
Gr	Grashof number
K	loss coefficient
l_i	axial length of *ith* section (m)
L_{th}	thermal center length (m)
$\dot{m}$	mass flow rate (kg/s)
M_{sys}	system mass (kg)
n	polytropic exponent
Nu	Nusselt number
P	pressure (kPa)
q	heat transfer rate (W)
Re	Reynolds number
Ri	Richardson number
St	modified Stanton Number
t	time (s)
T	temperature (K)
u_i	component velocity (m/s)
u_{co}	core inlet velocity (m/s)
u_o	characteristic velocity for natural convection (m/s)
v	specific volume (m^3/kg)

Greek Letters

β	volumetric thermal expansion coefficient (K^{-1})
μ	absolute viscosity (N s/m^2)
ρ	fluid density (kg/m^3)
ν	kinematic viscosity (m^2/s)
τ	time constant (s)
Π	nondimensional scaling parameter (Pi group)

Subscripts

c-core, *C*-cold, *e*-energy, *E*-equilibrium, *f*-saturated liquid, friction, *fg*-difference between saturated vapor and saturated liquid, g-saturated vapor, *H*-hot, *i-ith* component, *K*-form loss, *l*-liquid, L-reference length, *m*-model, *M*-mean, *o*-initial or steady-state, *P*-prototype, *R*-ratio, and *SG*-steam generator

Acknowledgments

The authors thank the Irradiation Experiment and Thermal Hydraulics department at INL for their support in completing this task.

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