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Scaling methodologies and similarity analysis for thermal hydraulics test facility development for water-cooled small modular reactor

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ABSTRACT

Small modular reactors (SMRs) represent a promising option for providing clean and sustainable energy due to their potential for enhanced safety, reduced capital costs, and increased siting flexibility. However, new reactor systems require the development and operation of representative scaled-down test facilities to support the verification and validation of system computer codes and models. This study reviews the research on scaling methodologies and similarity principles pivotal in developing non-nuclear integral effects test and separate effects test facilities for water-cooled SMRs. The study focuses on a review of the scaling methods, similarity approaches, and possible challenges posed by the unique and compact design features of integral-pressurized water reactor-type SMRs, and their representative test facilities. This study also reviews previous research related to scaling and similarity methodologies and provides insights into design considerations for achieving prototypic conditions in test facilities. The findings and recommendations emphasize the broader impact of appropriate scaling and similarity principles to ensure meaningful and transferable results from non-nuclear test facilities to accelerate the safe and efficient deployment of next-generation water-cooled SMRs.

1. Introduction

Small modular reactors (SMRs), as well as other advanced nuclear reactor technologies, are gaining priority due to an increasing emphasis on cleaner, sustainable, and efficient sources of power. SMRs, being smaller in size, modular by design, and versatile in application, offer numerous advantages compared to gigawatt-scale commercial fleet reactors—most of which are light water reactors (LWRs). The potential benefits of SMRs include enhanced safety, reduced capital costs, and the ability to be sited in a variety of locations—from remote areas to industrial hubs (IAEA, 2022; Subki, 2016). The development of SMRs can help promote the adoption of nuclear energy by reducing upfront costs, barriers to entry, and financial risk associated with nuclear power (Vaya Soler et al., 2021; Schlegel and Bhowmik, 2023). However, the design, development, and licensing of these new advanced reactor systems require: (a) the development of appropriate computer codes/models representing the reactor operational and safety performance; (b) the completion of required safety and programmatic analysis using these codes; and (c) the acquisition of adequate representative system performance data using scaled test facilities for model verification and validation (V&V). Regulatory and licensing changes to address the unique benefits and concerns associated with SMRs and new advanced reactors will continue to be a challenge as regulators adapt to the unique features emerging from the design process (Schlegel and Bhowmik, 2023; Black et al., 2021; Sam et al., 2023; Thomas and Ramana, 2022).

According to IAEA, more than 80 SMRs are under consideration in various stages: construction, design, development, demonstration and deployment. Among all, there are 25 SMRs that are water-cooled types: pressurized water-cooled reactor (PWR) and boiling water-cooled reactor (BWR). Recently, the United States (U.S.) Nuclear Regulatory Commission (NRC) issued its final rule to certify NuScale Power's 50-megawatt electrical (MWe) power module, the first U.S. SMR to receive approval for domestic use (NRC, 2020). The NRC Commission later voted to certify the design on July 29, 2022—making it the first SMR approved by the NRC for use in the U.S (DOE-NE, 2023a). Previously, SMART received design licenses from ROK's Nuclear Safety and Security Commission (NSSC) on July 4, 2012 (Kim et al., 2014). The U.S.

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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)
Ec	Eckert Number (i.e., mechanical energy/thermal energy)
Eu	Euler number (i.e., pressure losses/dynamic pressure)
f	Darcy friction factor
Fr	Froude number (i.e., inertia forces/body forces)
G	acceleration due to gravity (m/s^2)
Gr	Grashof number (i.e., buoyancy forces/viscous forces)
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)
Ma	Mach number (i.e., local flow velocity/speed of sound in the medium)
N	polytropic exponent
Nu	Nusselt number (i.e., convective heat transfer/conductive heat transfer)
N_d	Drift number (i.e., phase velocity/average velocity)
N_ρ	Density ratio (i.e., ratio of momentum of each phase)
N_σ	Surface tension ratio (i.e., surface tension force/pressure losses)
$N_{th,i}$	Thermal inertia ratio
N_{sub}	Subcooling number
P	pressure (kPa)
Pe	Peclet number (i.e., convection heat transfer/conduction heat transfer)
Pr	Prandtl number (i.e., momentum diffusion/thermal diffusion)
Q	heat transfer rate (W)
Re	Reynolds number (i.e., inertia forces/viscous forces)
Ri	Richardson number (i.e., buoyancy term/flow shear term)
St	Stanton number (heat transfer coefficient/heat capacity)
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	velocity (m/s)
V	volume (m^3)
V	specific volume (m^3/kg)
We	Weber number (i.e., dynamic pressure/surface tension force)
Zu	Zuber number (i.e., mass transfer rate/inertia)

Acronyms

ACME	Advanced Core-cooling Mechanism Experiment
ADS	automatic depressurization system
APEX	Advanced Plant Experiment
ASME	American Society of Mechanical Engineers
ATLAS	advanced thermal–hydraulic test loop for accident simulation
BDBA	beyond-design-basis accident
BEPU	best estimate plus uncertainty
BETSHY	Boucle d'Etudes Thermohydrauliques Système
BWR	boiling water reactor
BWXT	BWX Technologies, Inc. (Company)
CCTF	Cylindrical Core Test Facility
CSAU	code scaling and applicability
DBA	design-basis accident
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOE	U.S. Department of Energy
DSS	dynamical system scaling
DVI	direct vessel injection
ECCS	emergency core-cooling system
EM	evaluation model
EMDAP	Evaluation Model and Development Assessment Process
FLECHT	Full Length Emergency Cooling and Heat Transfer
FOAK	first-of-a-kind
FOM	figure of merit
FSA	fractional scaling analysis
GA	General Atomic
GLSS	generalized linear least-squares
H2TS	hierarchical two-tiered scaling
i-PWR	integral-pressurized water reactor
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
IST	Integrated System Test
ITL	Integral Test Loop
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
MASLWR	Multi-Application Small Light Water Reactor
MBLOCA	medium-break loss of coolant accident
MIT	Massachusetts Institute of Technology
ML	machine-learning
MSLB	main steam line break
MWe	megawatt electrical
NEA	Nuclear Energy Agency
NIST	NuScale Integral System Test
NRC	U.S. Nuclear Regulatory Commission
NSSC	Nuclear Safety and Security Commission
ORNL	Oak Ridge National Laboratory
OSU	Oregon State University
PACTEL	parallel channel test loop
PANDA	Passive Nachwarmeabbfiihr and DrtickAbbau Test Anlage
PCM	physical coverage mapping
PCS	primary coolant system
PCT	peak cladding temperature
PIRT	phenomena identification and ranking table
PKL	Primary coolant loop test facility
POI	phenomena of interest
PUMA	Purdue University Multi-dimensional Integral Test Assembly
PWR	pressurized water reactor
RAVEN	Risk Analysis Virtual ENvironment
RCP	reactor coolant pump
RCS	reactor coolant system
RELAP	Reactor Excursion and Leak Analysis Program
RG	Regulatory Guide
ROK	Republic of Korea
ROSA	Rig of Safety Assessment
RPV	reactor pressure vessel
SBLOCA	small-break loss of coolant accident
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	System-integrated Modular Advanced ReacTor-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
TREAT	Transient Test Reactor
U.S.	United States
USD	U.S. dollar
V&V	verification and validation
VISTA	experimental verification by integral simulation of transients and accident
VVER	Vodo-Vodyanoi Energetichesky Reactor

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)
α_0	void fraction ratio
Π	nondimensional scaling parameter (Pi group)

Department of Energy (DOE) is supporting ten U.S. advanced reactor designs to help mature and demonstrate their technologies (DOE-NE, 2023b), including water cooled SMRs. Few other examples of water-cooled land-based SMRs worldwide, which are in the 10-years of deployment horizon: CAREM (30 MWe output power, PWR-type, origin: Argentina), ACP100 (100 MWe output power, PWR-type, origin: China), NUWARD (2x170 MWe output power, PWR-type, origin: France), BWRX300 (270–290 MWe output power, BWR-type, origin: U.S.), and RITM-200 (2x53 MWe output power, PWR-type, origin: Russia) (IAEA, 2022; Schlegel and Bhowmik, 2023).

Similar to any new technology, transitioning SMRs from concept to commercialization requires rigorous testing. Historically, reactor system thermal–hydraulic (TH) testing facilities have been non-nuclear scaled-down versions—meaning no nuclear fuel material is used (Bestion et al., 2017)—with the notable exception of the Loss-of-Fluid-Test (LOFT) program (1983–1989) with a budget of about $100 million U.S. dollar (USD) for pressurized water reactors (PWRs) at Idaho National Laboratory (INL) (NEA, 2020). The LOFT project was originally set up by the NRC; part of it was later broadened into an international collaboration project under the aegis of the Nuclear Energy Agency (NEA) (NEA, 2020). Just as with large-capacity reactors, conducting full-scale nuclear tests for SMR systems is neither feasible nor safe. This underscores the importance of scaled-down non-nuclear test facilities. These facilities, which essentially are scaled-down versions of the prototype, provide a controlled environment where various parameters and conditions can be simulated and tested (NEA, 2020; Bhowmik, 2021; D'Auria, 2023). Through such testing, designers and engineers can gain insights into the behavior of a reactor system under a multitude of operational and postulated accident scenarios (Sainati et al., 2015; Ghosh et al., 2021; Obaidurrahman et al., 2021; Lien and Rohatgi, 2023). The motivations for using a scaled-down test facility to represent a prototypic reactor are:

- Economic: Elevated-capacity full-scale test facilities are expensive to build due to their high construction costs and development time. In addition to the financing and investment expenses, the size of the facility is restricted by infrastructure (e.g., building capacity) and available resources (e.g., land, water, and power supply).
- Safety: An added challenge is to maintain safety standards in terms of testing at the elevated pressures and temperatures representative of PWRs, as system-level performance and interaction between components and the overall system are yet to be understood.
- Management: Full-scale system-level testing requires a large amount of manpower and resources, which is difficult to manage in early-stage of reactor system design and development because the design evolves with lesson learning and business strategy.

This study consider i-PWR type SMRs as the reference reactor system, emphasizing the importance of safety considerations and design validation, as well as the significance of scaling and similarity principles in supporting the design and development of required test facilities for testing of safety systems and mechanisms. Data sets obtained from these facilities are essential for the V&V of computer codes and models (Aksan, 2019; Long et al., 2021; Bhowmik and Sabharwall, 2023a). This process ensures reactor system designs adhere to the regulatory safety envelope with adequate safety margins. While non-nuclear test facilities offer a promising avenue for testing, the challenge lies in ensuring that the results obtained from these scaled-down reactors are representative of what would occur in an actual SMR. Application of scaling and similarity principles based on fundamental physics and engineering concepts ensures that behaviors and phenomena observed in test facilities can be adequately translated to full-scale SMRs. In addition to the non-nuclear IETs, various scaled-down SETs are required for component-level TH analysis, such as electrically heated rod bundle tests (Bajorek and Cheung, 2019; Yang et al., 2013; Katono et al., 2022), steam generator (SG) tube bundle tests (Yu et al., 2023; Yao et al., 2021; Lee, Kim, and Ha, 2022; Abdellatif et al., 2024b,c), scaled-down reactor containment passive cooling tests (Bhowmik et al., 2023a,b; Hui et al., 2021), valve tests (Bhowmik and Sabharwall, 2023b; Bhowmik et al., 2023d; Bhowmik and Suh, 2021), reactor coolant pump tests (Park, Kim, and Lee, 2020; Ni et al., 2020; Bae, Jung, and Yu, 2023), and boron mixing tests (Hertlein et al., 2003; Yu et al., 2020).

This study focuses on the intricacies of scaled facility development for water-cooled SMRs—particularly the integral PWR type. Several first-of-a-kind (FOAK) IET facilities have been developed to cater to this need. The System-integrated Modular Advanced ReacTor (SMART) utilized experimental Verification by Integral Simulation of Transient and Accident–Integral Test Loop (VISTA-ITL) facility supports safety and transient analyses (Li et al., 2021). These analyses covered anticipated transient and postulated accident scenarios, such as small-break loss of coolant accidents (SBLOCAs), natural circulation flow, and critical heat flux tests, which are aimed at standard design approval (Park et al., 2023; Yang et al., 2020). The NuScale Integral System Test (NIST)-1, designed with a scaled, electrically heated core of 56 heater rods, assesses the NuScale power module by focusing on normal operation and passive safety system responses (Mundy, 2021). The Multi-Application Small Light Water Reactor (MASLWR) test facility at Oregon State University (OSU) aids in the development of the NuScale SMR with an integrated design, utilizing two vessels for heat removal—one representing the suppression pool and the other representing the external cooling pool (Morton, 2019; Modro et al., 2003). The BWXT-Integrated System Test (IST) was developed for the mPower SMR concept, leveraging a three-level scaling approach and testing systems such as high-pressure and low-pressure pumps (Morton, 2019). Nevertheless, challenges such as maintaining scaling consistency, minimizing distortions, and rigorous V&V of the model persist, thus highlighting the need for meticulous attention to design of the test facilities and testing protocols (Morton, 2019; Modro et al., 2003). This study covers the methods and approaches used to develop scaled-down non-nuclear TH IETs for water-cooled SMRs, especially for integral-pressurized water reactors (i-PWRs) with built-in pressurizers. The scope of this study is limited to the basic understanding about scaling and similarity methodologies with pros and cons applicable to reactor system thermal-hydraulics test facility development. The study also explores the challenges brought by the unique and compact design of SMRs in scaling and similarity principles and how to handle them. Through a synthesis of past research and contemporary findings, this study aspires to offer insights that can pave the way for developing scaled non-nuclear IETs and SETs to support the water-cooled SMRs design and development efforts.

The IET and SET facilities used for the 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 geometry similar to the prototypical nuclear fuel rods are used in the IET as a heat source. Similarly, a cooling tower or adequate capacity chillers are used as heat sinks to condense steam and return it to the feedwater system as part of the closed-loop reactor cooling system. Otherwise, steam produced from the IET can be released to the environment—an open-loop reactor cooling system, avoiding the requirement of a cooling tower or adequate capacity chillers—with additional safety precautions. The provided scaling ratio formulations support the loop length, area, volume, and power scale for specific components (e.g., core and SG) in the scaled-down or full-scale system. These specified scale ratios play a pivotal role in ensuring that the scaled-down experiments accurately capture the essence and behavior of the prototypic systems, as observed in previous and on-going reactor system experimental and demonstration programs. The presented scaling approaches for determining the numbers of heater rods and SG tubes elucidate how much attention should be paid to each component and its role in the overall system. This ensures the scaled model represents its larger counterpart—the prototypic structures, systems, and components—as accurately as possible.

2. State-of-the-art and steps in non-nuclear TH facility development

The development of non-nuclear TH and IET facilities start with phenomena identification and ranking table (PIRT) studies, which identify the phenomena of interest (POI), rank the importance of the POIs to the figures of merit (FOMs), and rank the state of knowledge (SOK) of the POI for the specific reactor design parameters (Wilson and Boyack, 1998; D'Auria, and Bestion, 2022; Schultz et al., 2017; Bhowmik et al., 2023d). 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 the data supports the Evaluation Model and Development Assessment Process (EMDAP) specified by the NRC Regulatory Guide (RG) 1.203 (NRC, 2005). If the data are not able to support the EMDAP, then IET and SET experiments are required to obtain the necessary data to support reactor design. Historically, IET and SET facilities were developed based on appropriate scaling methods to reduce cost and ensure safety. An agreeable solution for the scaling issue is mandatory within the framework of the so-called best estimate plus uncertainty (BEPU) and for the U.S. NRC code scaling and applicability (CSAU) methodology, which is later refined in the EMDAP to support the licensing (Rohatgi and Kaizer, 2020). Scaled-down test facility design requires the application of a scaling analysis to maintain scalability from perspectives such as geometry, fluid properties, event timing, and phenomena (e.g., boiling, condensation heat transfer, counter current flow limitation [CCFL], critical flow, entrainment, level swelling, entrainment, post critical heat flux, and departure of nucleate boiling). The data obtained from the experimental program (e.g., IETs and SETs) are considered first category data (Bhowmik et al., 2023e). Data obtained from the literature or data generated using the standard handbook are considered second category data (Bhowmik et al., 2023e). Test facilities are used to obtained adequate test data to qualify the computer codes by means of the EMDAP for regulatory approval.

2.1. Safety and design-basis considerations

The safety of a nuclear reactor is of utmost importance and its design must take all possible scenarios into account—including normal operation, control of abnormal operation and detection of failure, design-basis accidents (DBAs), and beyond-design-basis accidents (BDBAs). The coupling of TH, neutronic, and fluid-structural behavior must be carefully considered when setting design limits to ensure the reactor remains safe and stable under all conditions (Bhowmik et al., 2021a; Todreas et al., 2021). 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 to minimize the risk of harm to people and the environment.

2.2. Brief overview of IET facilities for PWRs worldwide

Over the past 50 years, several IET facilities have been developed worldwide to support the design and licensing of various reactor designs. These include INL's SEMISCALE and LOFT, OSU's Advanced Plant Experiment (APEX), Purdue University's Multi-Dimensional Integral Test Assembly (PUMA), and Westinghouse's Full Length Emergency Cooling and Heat Transfer (FLECHT) systems in the U.S.; the Rig of Safety Assessment (ROSA)/Large-Scale Test Facility (LSTF) and Cylindrical Core Test Facility (CCTF) of a PWR in Japan; the Simulatore per Esperienze di Sicurezza (SPES) in Italy; the Boucle d'Etudes Thermohydrauliques Système (BETHSY) in France; the Primary Coolant Loop Test Facility (PKL) in Germany; the Advanced Thermal-hydraulic Test Loop for Accident Simulation (ATLAS) in Korea; and the Advanced Core-cooling Mechanism Experiment (ACME) in China (Bestion et al., 2017; Bhowmik, 2021; Deng et al., 2019; Yu and Choi, 2016; Reyes and Hochreiter, 1998; Loomis, 1987). These programs and facilities were developed for targeted commercial LWRs, like PWRs and BWRs, to examine reactor safety issues related to plant response during a loss of coolant accident (LOCA) and operational transient. Few IET facilities (e.g., APEX, ACME, VISTA-ITL, and SPES-2) 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.; and details are available on recent studies (Bhowmik, 2021; Deng et al., 2019).

2.3. PIRT studies

A PIRT study is an important element for the EMDAP, which is used to ensure that system simulation computer codes properly simulate system behavior. Understanding transient reactor accident progression is important during the PIRT development work, because various kinds of reactor accident scenarios can be summarized into typical time sequences and POIs are assessed for each time sequences.

During the PIRT, experts gather information about a specific reactor design concept and process and phenomena involved and rank its' importance according to a decision-making objective and established figure of merit (FOM). A PIRT study identifies the POIs and ranks the phenomena with different FOMs and SOKs. 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 steps (Bestion et al., 2017; Zuber et al., 2007, Shaw et al., 1985; Wilson and Boyack, 1998; Liao et al., 2021): motivation, objective, database, hardware-scenario, FOMs, phenomena identification, importance ranking, knowledge level, and documentation. These steps are presented briefly as follows:

- 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 time sequences.
- FOMs: Select key FOMs used to judge importance.
- Phenomena Identification: Identify all plausible phenomena plus definitions.
- Importance Ranking: Assign importance relative to FOMs; document the rationale.
- Knowledge Level: Assess the current level of knowledge regarding each phenomenon.
- Document PIRT: Document the effort with sufficient coverage so 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 that are considered are breaks in piping connected to the reactor coolant system (RCS) pressure boundary. These breaks consist of POIs that are tightly coupled and strongly dictate the accident progression.

2.4. Reactor accident progression

Reactor accident progression varies with reactor system design. For example, a few PWR-type reactors (e.g., the AP1000) are equipped with an automatic depressurization system (ADS) to depressurize the RCS during a LOCA to inject cooling water by passive safety systems, which differ from pumped flow injection of traditional emergency core-cooling systems (ECCSs). 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 tanks and cavities (Bhowmik et al., 2021b). However, the initiation of the ADS also depends on the RCS breaks. ADS is started by the system pressure set point, and design study explaining the time sequences for various ADS stages (e.g., ADS-1, ADS-2, and ADS-3) are required for reactor system safety and accident analysis (Xing et al., 2023).

Several piping breaks in an RCS are possible, such as a main steam line break (MSLB), steam generator tube rupture (SGTR), and direct vessel injection (DVI) line break, as well as a break in the reactor coolant pump (RCP) lines (Abdellatif et al., 2024a,b). Likewise, there are various LOCA scenarios that are based on break size: (1) an 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. However, for Generation III and advanced PWR (e.g., AP1000) after the TMI-2 accident, SBLOCA proves challenging as the reactor accident advances through several stages and involves subcooled blowdown, saturated natural circulation, ADS depressurization, in-containment refueling water storage tank (IRWST) injection, and long-term cooling. These accident time sequences can be identified by analyzing the pressure trend versus the time progression of the primary coolant system (PCS), as presented in Fig. 1 (although, ROSA-AP600 ranged all the transient, not until ADS critical discharge). Fig. 1 on a full-passive large reactor accident response can be generalized to be applicable for water-cooled integral-PWR type SMR (Abdellatif et al., 2024b). Developing an IET facility that covers required and established accident time sequences is crucial to reactor licensing applications.

Fig. 1 shows a typical SBLOCA time sequence that may arise in a so-called full-passive reactor such as AP1000 (Westinghouse), which include subcooled blowdown, saturated natural circulation, ADS initiation, IRWST injection, and long-term cooling. It is possible to identify the time sequences by means of the primary pressure transient during the SBLOCA. The IET facilities should cover all required and established accident time sequences. Testing plans and test matrixes are required to obtain data from the experimental facilities according to the time sequences according to the time sequence of the reactor accident scenarios specific to the reactor design of interest (Deng et al., 2019).

Scaling analysis, including computer/system code modeling, should fully address the IET facility reactor system design and assessment, which specifies the POIs and FOMs during complex transient accident scenarios with several consecutive time sequences (Dzodzo et al., 2019). SBLOCA analysis on the reactor system covers the wider time stages, as presented in Fig. 1, however, varied time sequences are required for specific reactor system design to estimate the prominent FOMs such as: (a) reactor and containment vessel pressure and temperature responses, and (b) reactor vessel/steam generator (mixture level) transient that may define peak cladding temperature (PCT) as FOM in the core. Simulating a transient response in the IET requires the consideration of the systems, subsystems, interconnecting components, and opening/closing valves, as well as their respective control logic in the reference reactor.

2.5. Scaling and similarity: approaches and considerations

Scaling analyses are required to guide development of an IET (i.e., scaled facility) and 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

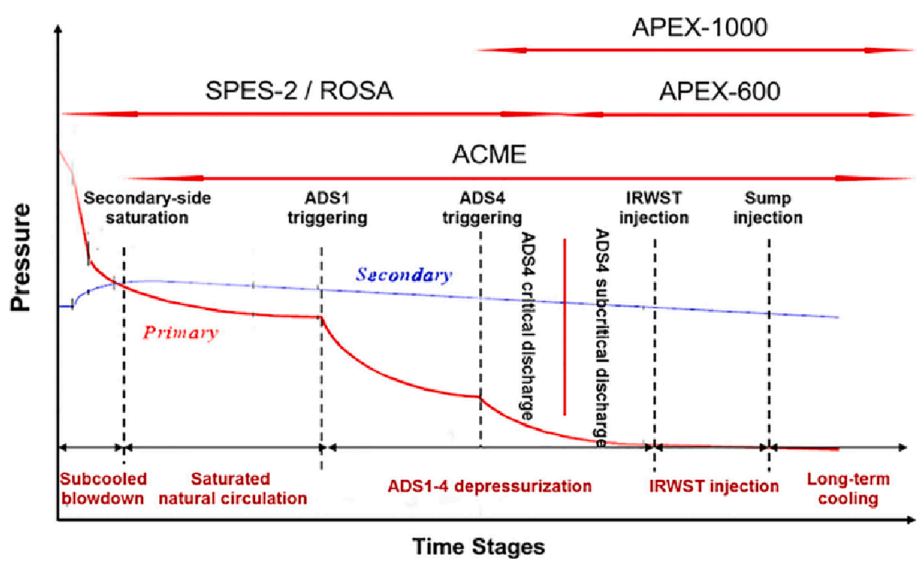

Fig. 1. Scoping experiments comparison between selected IET facilities (Li et al., 2012).

simulator, but a properly scaled IET that can represent the important high-ranking phenomena needed for reactor transient analysis.

Scaling for reactor system TH 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-to-volume, acceleration, residence time and frequencies, etc. Traditionally, scaling parameters are expressed using scaling ratios of nondimensional numbers. The ratio of 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 Eq. (1).

$$\Pi_R = \frac{\Pi_{model}}{\Pi_{prototype}} = \frac{\Pi_m}{\Pi_p} = 1 \tag{1}$$

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

Four types of similarities exist in THs: (a) geometric similarity; (b) kinematic similarity; (c) dynamic similarity; and (d) thermal similarity. Geometric similarity pertains solely to the geometry of objects, irrespective of fluids or flow rates. It is often the simplest form of similarity and necessitates that both the model and the prototype maintain similar shape and proportions, such as a 1/10th-length-scale model heat exchanger. It is distinct from other types of similarities in that it is not determined by dimensionless numbers, but instead by ratios like height to length. Kinematic similarity implies that velocities and streamlines between the model and prototype differ solely by a constant factor. For this type of similarity, geometric similarity is a precondition. Its foundation lies in the continuity equation, with examples being maintaining ratio of residence. Dynamic similarity, on the other hand, ensures that forces on a model and its prototype differ only by a constant magnitude. This similarity places the most stringent requirements on scale models, as all forces crucial to flow must be factored in, with its basis in the momentum equation. Finally, thermal similarity dictates that temperatures and heat fluxes between the model and prototype differ by a consistent factor, making it especially challenging in the domain of heat transfer problems, with its basis in the energy equation. The basic features of these similarity approaches are presented in Table 1.

Scaling extends beyond mere similarity. It delves into system design and its overarching effects on the entire system. At times, conducting experiments at full-scale is not only expensive, but hazardous as well.

Table 1
Similarity approaches and considerations in TH system modeling.

Similarity Approaches	Approaches and Considerations
Geometric Similarity	• Only depends on geometry – not fluids, flow rates, etc. • Often the easiest type of similarity, not based on dimensionless numbers • Requires that model and prototype have similar shape • Height/length ratio, etc. (i.e., a 1/100th-scale model airplane).
Kinematic Similarity	• Means that velocities differ only by a constant factor • Streamlines also related by a constant factor • Geometric similarity is a prerequisite • Arises from continuity equation.
Dynamic Similarity	• Means that forces differ only by a constant factor • Most restrictive requirements for most scale models • All forces important to flow must be considered • Arises from momentum equation.
Thermal Similarity	• Means that temperatures and heat fluxes differ by factor(s) with allowable ranges • Also can be very restrictive for heat transfer problems. • Arises from energy equation.

Thus, a necessity arises to employ scaled models and simulations. The challenge lies in determining how a system can be designed so that it retains its inherent similarity. This endeavor is about identifying pivotal design parameters that demand attention. Instead of focusing narrowly on a single component, the emphasis often shifts to a holistic system design. One example is consideration how to maintain parameters integral to the system, such as form loss (needed to obtain dynamic similarity).

Dimensionless numbers are used after normalizing the TH governing equations for continuity, momentum, and energy and deriving dimensionless variables, as presented in Table 2. The representative dimensionless numbers used in TH systems with their mathematical formulation and importance are presented in Table 3.

3. Scaling methodologies for reactor system thermal-hydraulics

The nuclear industry uses several scaling methods to develop non-nuclear IET facilities for specific reactor systems. These scaling methods include: (a) linear scaling; (b) power-to-volume scaling; (c) three–level scaling; (d) hierarchical two-tiered scaling (H2TS); (e) fractional scaling analysis (FSA); and (f) other scaling methods, such as power-to-mass scaling and dynamic system scaling (DSS). Historically, these scaling methods have evolved while addressing specific challenges. An overview of the scaling methods/approaches with pros and cons and their applicability are presented in Table 4.

3.1. Scaling models for reactor system (single-phase and natural circulation)

The ideal scaled facility—directly derived from governing equations—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. For geometric scaling need to modify the ideally scaled IET to an engineering-scale facility, as for example, the SG tube diameter cannot be scaled similar to SG shall diameter scaling. If the ideal scaling end up with a shell diameter that has no commercial pipe size, then it might need engineered scaling with allowable scaling distortion. The geometric ratios consist of: (a) the length ratio—to match the available laboratory space, which determines the velocity ratio; (b) the area ratio—to match commercially available pipe sizes; and (c) the volume ratio—from the length and area ratios, which determine the power ratio. The scaling analysis of a reactor involves the governing equations, models, correlations, and nondimensional numbers, as presented in Table 5. A recent experimental study using H2TS scaling method, three group of test facilities of describes the scaling distortion of natural circulation steady-state characteristics, and provided relation curves between dimensionless numbers such as Reynolds number (Re), equivalent resistance coefficient, Grashof number (Gr) and modified Grashof number (Cheng et al., 2023). Researchers estimated scaling distortion and uncertainties for single phase natural circulation in a prototype and scaled-down test facilities using system code (e.g., RELAP5 or other code) (Kim et al., 2021, Lorduy-Alós et al., 2021, Li et al., 2020).

The constituent energy equations can be nondimensionalized using the core inlet velocity as the reference velocity, the core height as the reference length, the core temperature difference as the reference temperature difference, the core flow area as the reference flow area, and the conduction thickness, δ, as the reference thermal depth for the solid. The geometry-level governing equations and nondimensional parameters, similarity criteria and scale ratios, and energy scale ratios are presented in Table 6, Table 7, and Table 8, respectively.

The significances 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

Table 2
The general governing equations and variables representing a TH system.

Governing Equations and Models	Equations and Variables
Start with an incompressible fluid with constant viscosity in a gravity field, and consider metal mass energy balance and heat transfer to fluid	For single-phase fluid flow: incompressible fluid with constant viscosity Continuity: $\nabla\cdot\overrightarrow{v} = 0$ Momentum: $\rho\left(\frac{\partial\overrightarrow{v}}{\partial t}+\overrightarrow{v}\cdot\nabla\overrightarrow{v}\right) = -\nabla p + \rho\overrightarrow{g} + \mu\nabla^2\overrightarrow{v}$ Thermal Energy: $\rho c\left(\frac{\partial T}{\partial t}+\overrightarrow{v}\cdot\nabla T\right) = k\nabla^2 T + \nabla\cdot(\mu\nabla^2\overrightarrow{v}\cdot\overrightarrow{v})$ where TH variables are velocity, v; pressure, p; gravity, $\overrightarrow{g}$; and temperature, T; and the fluid properties are density, ρ; conductivity, k; and viscosity, μ. Metal mass energy equation: The energy balance of a heat structure wall is given by: Metal mass energy: $M_w c_p\frac{dT_w}{dt} = -h_w SA_w(T_w - T_f)$ where, variables are mass of wall, M_w; specific heat of wall, c_p; temperature change of wall with respect to time, $\frac{dT_w}{dt}$; heat transfer coefficient at wall, h_w; surface area of wall, SA_w; temperature difference between wall and fluid, $(T_w - T_f)$.
Define dimensionless variables: normalized or scale value representation	Dimensionless length, $\overrightarrow{x}^* = \frac{\overrightarrow{x}}{L}\overrightarrow{x} = \overrightarrow{x}^* L$; Dimensionless velocity, $\overrightarrow{v}^* = \frac{\overrightarrow{v}}{V}\overrightarrow{v} = \overrightarrow{v}^* V$ Dimensionless time, $t^* = \frac{t}{\tau}\Rightarrow t = t^*\tau$; Dimensionless pressure, $p^* = \frac{p}{p_0}\Rightarrow p = p^* p_0$ Dimensionless temperature, $T^* = \frac{T}{T_0}\Rightarrow T = T^* T_0$.
Substituting the dimensionless variables to the TH governing equations	For single-phase fluid flow: incompressible fluid with constant viscosity Continuity: $\nabla^*\cdot\overrightarrow{v}^* = 0$ Momentum: $\left(\frac{L}{V\tau}\right)\frac{\partial\overrightarrow{v}^*}{\partial t^*} + \overrightarrow{v}^*\cdot\nabla^*\overrightarrow{v}^* = -\left(\frac{p_0}{\rho V^2}\right)\nabla^* p^* + \left(\frac{\lvert\overrightarrow{g}\rvert L}{V^2}\right)\frac{\overrightarrow{g}}{\lvert\overrightarrow{g}\rvert} + \left(\frac{\mu}{\rho VL}\right)\nabla^{*2}\overrightarrow{v}^*$ Thermal Energy: $\left(\frac{L}{V\tau}\right)\frac{\partial T^*}{\partial t^*} + \overrightarrow{v}^*\cdot\nabla^* T^* = \left(\frac{k}{\mu c}\right)\left(\frac{\mu}{\rho VL}\right)\nabla^{*2}T^* + \left(\frac{V^2}{cT_0}\right)\left(\frac{\mu}{\rho VL}\right)\nabla^*\cdot(\nabla^{*2}\overrightarrow{v}^*\cdot\overrightarrow{v}^*)$.
Considering two-phase flow conditions with dimensionless variables to the TH governing equations	For two-phase fluid flow: Momentum: $\rho_k^* = \frac{\overline{\overline{\rho}}_k}{\rho_{k0}}, v_k^* = \frac{\widehat{\overrightarrow{v}}_k}{v_{k0}}, t^* = \frac{t}{\tau_0}, \nabla^* = L_0\nabla, \Gamma_k^* = \frac{\overline{\overline{\Gamma}}_k}{\Gamma_{k0}} \Downarrow$. $\frac{1}{Sl_k}\frac{\partial}{\partial t^*}\alpha_k\rho_k^* + \nabla^*\cdot\alpha_k\rho_k^* v_k^* = Zu_k\Gamma_k^*$ $P_k^* = \frac{\overline{\overline{P}}_k - P_{k0}}{\Delta P_0}, \tau_k^* = \frac{\overline{\overline{\tau}}_k}{\mu_{k0}v_{k0}/L_0}, M_{ik}^* = \frac{\overrightarrow{M}_{ik}}{a_{i0}(\rho_{d0}+\rho_{c0})(v_{d0}-v_{c0})^2}, g^* = \frac{\overrightarrow{g}}{\lvert\overrightarrow{g}\rvert}$ Thermal Energy: $\frac{1}{SI_k}\frac{\partial}{\partial t^*}\alpha_k\rho_k^* v_k^* + \nabla^*\cdot\alpha_k\rho_k^* v_k^* v_k^* = -Eu_k\alpha_k\nabla^* P_k^* + \frac{1}{\mathrm{Re}_k}\nabla^*\cdot\alpha_k\left(\tau_k^* + \tau_{k,T}^*\right)$. $+\frac{1}{Fr_k}\alpha_k\rho_k^* g^* + N_{D,k}M_{ik}^* - \frac{1}{\mathrm{Re}_k}\nabla^*\alpha_k\cdot\tau_{ki}^* + Zu_k\Gamma_k^*(v_{ki}^* - v_k^*)$ $+Eu_k(P_{ki}^* - P_k^*)\nabla^*\alpha_k$ Enthalpy: $i_k^* = \frac{\widehat{i}_k - i_{k0}}{\Delta i_0}, (q_k'')^* = \frac{\overline{\overline{\overline{q}}}_k'' L_0^2}{k_{k0}\Delta T_0 L_0}, (q_{ki}'')^* = \overline{\overline{\overline{q}}}_{ki}'' a_{i0}k_{k0}(T_{i0} - T_{k0}) \Downarrow$ $\frac{1}{Sl_k}\frac{\partial}{\partial t^*}\alpha_k\rho_k^* i_k^* + \nabla^*\cdot\alpha_k\rho_k^* h_k^* i_k^* = -\frac{1}{Pe_k}\nabla^*\cdot\alpha_k\left(q_k'' + q_{k,T}''\right)^*$. $+Eu_k Ec_k\alpha_k\left\{\frac{1}{Sl_k}\frac{\partial P_k^*}{\partial t^*} + v_k^*\cdot\nabla^* P_k^*\right\} + \frac{Ec_k}{\mathrm{Re}}\alpha_k\tau_k^* : \nabla^* v_k^*$ $+\frac{Ec_k}{\mathrm{Re}_k}\nabla^*\alpha_k\cdot\tau_{ki}^*\cdot(v_{ki}^* - v_k^*) + N_{D,k}Ec_k M_{ik}^*(v_{ki}^* - v_k^*) + Zu_k\Gamma_k^*(i_{ki}^* - i_k^*) + N_{q,k}a_i^*(q_{ki}'')^*$

Based on the term used to normalize the equations, various (additional) dimensionless numbers can be derived (Ruzicka, 2008; Dzodzo, 2023).

normal operation and certain accident scenarios (e.g., RCP line break and SGTR) and also provide a basis 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 number (*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 length. 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.

3.2. Scaling approaches applicable to reactor system

The preliminary ideal scaling analysis can be evaluated against predictions obtained using a system code (e.g., Reactor Excursion and Leak Analysis Program [RELAP]5 model) for steady-state, transient, and accident conditions. The next step is to modify the ideally scaled IET to an engineering-scale facility. Factors that need to be considered during engineering scaling to design a realistic test facility are: (a) the use of components that can actually be obtained or built; (b) scaling ratios that

Table 3
Dimensionless group or numbers used in TH system modeling.

Dimensionless Group or Numbers	Mathematical Representation	Representation and Importance
Strouhal Number	$St = \frac{L}{V\tau}$	• Ratio of local inertia to convective inertia • Important in unsteady (especially periodic) flows.
Euler Number	$Eu = \frac{\text{pressure losses}}{\text{dynamic pressure}} = \frac{p_0}{\rho V^2} = \frac{\Delta p_0}{\rho_{k0} v_{k0}^2}$(2-phase flow)	• Ratio of pressure forces to inertia • Important when large pressure changes occur • Also can be written as pressure coefficient or cavitation number • Subscript k for 2-phase flow (i.e., liquid or gas).
Reynolds Number	$\mathrm{Re} = \frac{\text{inertia}}{\text{viscous forces}} = \frac{\rho v L}{\mu} = \frac{\rho_{k0} v_{k0} L_0}{\mu_{k0}}$(2-phase flow)	• Ratio of inertia to viscous forces • Important in almost all flows.
Froude Number	$Fr = \frac{\text{inertia}}{\text{body forces}} = \frac{v}{\sqrt{gL}} = \sqrt{\frac{v_{k0}^2}{gL_0}}$(2-phase flow)	• Ratio of inertia to gravitational forces • Typically important when flows involve a free surface (open channels).
Grashof Number	$Gr = \frac{g\beta\Delta T L^3}{\upsilon^2}$	• Ratio of the buoyancy to viscous forces acting on a fluid • Analogous to the Re.
Richardson Number	$Ri = \frac{\beta g \Delta T L_c}{u^2} = \frac{\frac{g\beta\Delta T L^3}{\upsilon^2}}{\left(\frac{uL}{\upsilon}\right)^2} = \frac{Gr}{Re^2}$	• Ratio of the buoyancy term to the flow shear term, represents the importance of natural convection relative to the forced convection • General considerations: $Ri \gg 1 \Longrightarrow$ ignore forced convection; $Ri \approx 1 \Longrightarrow$ combined forced and free convection; $Ri \ll 1 \Longrightarrow$ ignore free convection.
Prandtl Number	$\mathrm{Pr} = \frac{k}{\mu c} = \frac{k}{\rho c}\frac{\rho}{\mu} = \frac{\alpha}{v}$	• Ratio of momentum diffusion to thermal diffusion.
Eckert Number	$Ec = \frac{\text{mechanical energy}}{\text{thermal energy}} = \frac{v^2}{cT_0} Ec_k = \frac{v_{k0}^2}{\Delta i_{k0}}$(2-phase flow)	• Ratio of kinetic energy to thermal energy • Usually very small, but can be important for compressible flow (shock waves, etc.).
Weber Number	$We = \frac{\text{dynamic pressure}}{\text{surface tension force}} = \frac{\rho V^2 L}{\sigma}$	• Ratio of inertia to surface tension forces • Important when there is an interface between two substances • Arises from boundary condition of many problems.
Mach Number	$Ma = \frac{v}{c}$	• Measures compressibility of the flow, here c is the speed of sound.
Zuber Number	$Zu_k = \frac{\text{mass tranfer rate}}{\text{inertia}} = \frac{\Gamma_{k0} L_0^2}{\rho_{ko} v_{k0} L_0}$	• Scales change in momentum due to phase change with momentum.
Drift Number	$N_d = \left(\frac{\overline{v}_{g,i}}{v_0}\right)_i$	• Scales the velocity difference between phases to the average velocity (importance of relative velocity).
Peclet Number	$Pe_k = \frac{\text{convection}}{\text{conduction}} = \frac{\rho_{k0} v_{k0} \Delta i_{k0} L_0}{k_{k0} \Delta T_{k0}}$	• Ratio of convection to conduction.
Density Ratio	$N_\rho = \frac{\rho_{d0}}{\rho_{c0}} = $ density ratio	• Ratio of momentum of each phase.
Surface Tension Number	$N_\sigma = \frac{\text{surface tension force}}{\text{pressure losses}} = \frac{H_{cc0}\sigma_0}{\Delta P_0}$	• Ratio of surface tension forces to mixture inertia.
Other Relevant Numbers for Two-phase Flow	Time Ratio Number $T_i^* = \left(\frac{L_0/v_0}{\delta^2/\alpha_s}\right)_i$; Thermal Inertia Ratio $N_{th,i} = \left(\frac{\rho_s c_{ps}\delta}{\rho_f c_{pf} D}\right)_i$ Subcooling Number $N_{sub} = \left(\frac{i_{sub}}{i_{fg}}\right)\left(\frac{\Delta\rho}{\rho_g}\right)$Void Ratio $\alpha_0 = \left(\frac{\rho_f}{\Delta\rho}\right)\left[\frac{1}{1+(N_d+1)/(Zu-N_{sub})}\right]$.	

should be maintained as closely as possible by considering engineering scale value (example, using commercially available standard pipe diameters and thicknesses) instead of ideal scale value. Various scaling parameters can be considered for several scaling approaches, as shown in Table 9. The most common types are linear scaling, power-to-volume scaling, three-level scaling, and H2TS, as follows (discussed in Table 4):

- 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) 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 as an emphasis on the important phenomena; however, there is a contradiction with the scaling criteria.

Historically, various IET facilities were developed to support reactor system design, development, and demonstration. Some of these facilities are: Advanced Core-cooling Mechanism Experiment (ACME); Advanced Plant Experiment (APEX); Advanced Thermal-hydraulic Test Loop for Accident Simulation (ATLAS); Boucle d'Etudes Thermohydrauliques Système (BETHSY); Integral Test Stand Karlstein (INKA); LWR Off-Normal Behavior Investigation (LOBI); Loss of Fluid Test (LOFT); Large Scale Test Facility (LSTF); Parallel Channel Tests Loop (PACTEL); Passive Nachzerfallswärmeabfuhr und Druck-Abbau Testanlage (PANDA); Primary Coolant Loop Test Facility (PKL); Purdue University Multi-Dimensional Integral Test Assembly (PUMA); Rig of Safety Assessment (ROSA); INL IET facility (SEMISCALE); System-integrated Modular Advanced ReacTor (SMART); and Simulatore per Esperienze di Sicurezza (SPES).

These IETs are scaled-down facilities derived from their respective prototype reactors. They were developed to obtain data for reactor system transient and accident analysis. These data were used to validate computer models of reactor behavior and to demonstrate that a proposed design meets safety and performance requirements. Currently, more than 80 different SMRs are proposed and are in the design and development phases. Most of these new SMRs and advanced reactors will require experimentation using scaled-down non-nuclear TH IET and SET facilities. The design of these facilities might evolve based on new findings, changes in market demand, and business strategies. Evolving reactor designs and analyses can benefit from the lessons learned from previous studies, as effectively reviewed and evaluated in this study.

Table 4
Scaling methods/approaches and their advantages and disadvantages (Wang and Yan, 2021; D'Auria and Galassi, 2010 Yun et al., 2004).

Scaling Methods/Approaches	Advantages and Disadvantages
Linear Scaling: Introduced by Carbiener and Cudnik (1969) and Nahavandi et al. (1979) proposed method that retains gravitational effects Produced the same design laws for test facilities and resizes prototype dimensions by a common factor, scales both fluid and sound movement time.	Advantages: Smaller version of the prototype, can interpret component interaction especially when gravity's influence is less than system pressure drop. Disadvantages: Lower limit acceptability of scaling factors and higher distortion and challenges in energy transfer simulations and down-sizing fuel rods.
Power-to-Volume Scaling: Introduced by Nahavandi et al. (1979) as a solution for nuclear reactor scaling (Deng et al., 2019), simple method for designing full-height test facilities featuring full pressure, and time consistency Maintains time and heat flux, aiding in recreating strong gravity effects Typically associated with the full-pressure, full-height and time-preserving requirements. Example test facilities: LOFT, SEMISCALE, LSTF, LSTF, BETHSY, LOBI	Advantages: Time and height preserving Disadvantages: Cost of the facility and strong impact upon the transient evolution of thermal power release from passive structures. Distortion in pressure drop, heat loss and multi-dimensional phenomena (e.g., high distortion in entrance effects simulation)
Three-Level Scaling: Proposed by Ishii and Kataoka (1983, 1984), focuses on conserving natural circulation in LOCA scenarios Assumes a 1-D system for components, uses single-phase and drift-flux two-phase flow formulations Employs three levels: (a) system response function scaling; (b) control volume and boundary flow scaling; and (c) local phenomena scaling For systems involving both single and two-phase flow in a reduced length model, real-time scaling is not appropriate. Example test facilities: PUMA, ATLAS	Advantages: Maintains flow regimes, local phenomena, and reduces scaling distortions. Identifying the 'reduced length (and height)' as a scaling advantage and reduction of construction cost. Disadvantages: If axial length is reduced in the model, then the time scale is shifted (e.g., in the two-phase flow natural circulation loops). In such a case, the events are accelerated in the scaled-down model (i.e., time scale shifting).
Hierarchical Two-Tiered Scaling (H2TS): Developed by Zuber's team (Zuber, 1991; Zuber et al., 1998; Zuber, 2001; Zuber et al., 2005; Zuber et al., 2007), ranking of processes based on importance and follows a hierarchical approach in spatial, temporal, and energy terms Comprises four stages: (a) system breakdown; (b) scale identification; (c) top-down scaling; and (b) bottom-up scaling (see Note 1). Example test facilities: APEX, ACME, OSU-MASLWR	Advantages: Scaling hierarchy for complex system decompositions to subsystem, modules, constituents, phases, geometric configuration, fields, and process. Disadvantages: Contradiction of scaling criteria and exhibit distortion due to complex multidimensional phenomena.
Fractional Scaling Analysis (FSA): Originates from Zuber (Zuber, 1991; Zuber et al., 1998; Zuber, 2001; Zuber et al., 2005; Zuber et al., 2007), ranks components based on their influence on certain parameters, and focuses on state variables influenced by convection, diffusion, and wave propagation.	Advantages: Efficiently reduces experimental and computational work with a systematic approach that optimizes facility design, and useful for analyzing and identifying critical processes.

Note 1 The H2TS comprises four stages: (a) system breakdown: divide the system into its smaller parts, such as subsystems, modules, and geometrical setups; (b) scale identification: create a structure considering volume, space, and time; (c) top-down scaling: formulate a scaling structure based on conservation equations and adjust these equations at each level to identify the time ratios and similarities; and (d) bottom-up scaling: conduct a thorough scaling study for essential processes to determine similarity criteria for local events.

Table 5
The governing models, equations, and variables related to reactor system scaling analysis.

Governing Equations and 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: $\sum_{i=1}^{N}\left(\frac{l_i}{a_i}\right)\bullet\frac{d\dot{m}}{dt} = \beta g\rho_l(T_H - T_c)L_{th} - \frac{\dot{m}^2}{\rho_l a_c^2}\sum_{i=1}^{N}\left[\frac{1}{2}\left(\frac{fl}{d_h}+K\right)_i\left(\frac{a_c}{a_i}\right)^2\right]$ where the i subscripts refer to the ith component and L_{th} is the distance (length) between thermal center. The core cross-sectional flow area, a_c, is used as the reference flow area.
1-D loop momentum with assumptions	The following assumptions are made: (a) 1-D flow along the loop axis; therefore, the fluid properties are uniform at every cross-section; (b) The Boussinesq approximation is applicable where all of the fluid densities within the loop are equal to the average fluid density except for those that make up the buoyancy term; and (c) The fluid is incompressible. Based on these assumptions, the momentum equation of a single-phase: $\rho_0\frac{du_0}{dt}\sum_{i=1}^{n}L_i\left(\frac{A_0}{A_i}\right) = \beta g\rho_0(T_H - T_C)L_{TC} - \frac{\rho_0 u_0^2}{2}\sum_{i=1}^{n}\left(\frac{fL}{D_h}+k\right)\left(\frac{a_0}{a_i}\right)^2$ where, density, ρ; velocity, u; length, L; area, a; thermal expansion coefficient, β; gravity, g; exit temperature of heat source, T_H; exit temperature of heat sink and inlet temperature toheat source, T_C; distance (length)between thermal centers, L_{TC}; subscript 0 for the location at the core inlet; friction factor, f; minor loss coefficient, k; and hydraulic diameter, D_h.
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 and reference length number	$\tau_{loop} = \sum_{i=1}^{N}\frac{l_i}{u_i} = \sum_{i=1}^{N}\tau_i = \frac{M_{sys}}{\dot{m}_o} = \frac{M_{sys}}{\rho_l u_{co} a_c}$ $\Pi_L = \sum_{i=1}^{N}\frac{l_i}{l_{ref}}\frac{a_c}{a_i}$, where $l_{ref} = \frac{M_{sys}}{\rho_l a_c}$
Nondimensional number: Richardson number (Ri)	$Ri = \frac{\beta g(T_H - T_C)_o L_{th}}{u_{co}^2} = \frac{\beta g q_{co} L_{th}}{\rho_l a_c C_{pl} u_{co}^3}$ These two forms of Ri comes from an energy balance across the core $q_{co} = \rho_l a_c u_{co} C_{pl}(T_H - T_c) = \dot{m}_{co} C_{pl}(T_H - T_C)$ $Ri = \frac{\beta g\Delta T L_c}{u^2} = \frac{\frac{g\beta\Delta T L^3}{\upsilon^2}}{\left(\frac{uL}{\upsilon}\right)^2} = \frac{Gr}{Re^2} = \frac{u_o^2}{u^2} = \frac{u_o^2}{u_{co}^2}$ For the reactor system natural circulation, considering $u = u_{co}$ $u_o = \sqrt{\beta g(T_H - T_C)L_{th}}$ and $u = u_{co}$ Therefore, core inlet velocity, $u_{co} = \left(\frac{\beta q_{co} L_{th} g}{\rho_l a_c C_{pl}\Pi_{Fl}}\right)^{1/3}$.
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}, represents 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.

Table 6
The geometry-level governing equations, and nondimensional parameters related to reactor system scaling analysis.

Geometry-level Equations and Models	Equations and Variables
Conduction thickness and hydraulic diameter	For $1-\Phi$ natural circulation $\delta_i = \frac{a_{s_i}}{P_w}$ and $d_{h_i} = 4\delta_i \frac{a_i}{a_{s_i}}$ where conduction thickness, δ, defined as the ratio of the solid cross-sectional area of each section divided by the wetted perimeter, which is similar to the definition of hydraulic diameter (flow cross-sectional area divided by wetted perimeter).
Fluid energy equation for 1-Φ natural circulation	$\rho C_p \left\{\frac{\partial T}{\partial t} + u\frac{\partial T}{\partial z}\right\} = \frac{4h_{conv}}{d_h}(T_s - T)$ Examines fluid-solid transient heat transfer processes and the associated scaling issues.
Solid energy equation for 1-Φ natural circulation	$\rho_s C_{ps}\frac{\partial T_s}{dt} + k_s \nabla^2 T_s = \dot{q}_s$ Consideration of this energy transfer mechanism necessitates the inclusion of an energy equation for both the fluid (1-Φ or 2-Φ) and the solid.
Boundary condition	$-k_s \frac{\partial T_s}{\partial y} = h_{conv}(T_s - T)$ Balancing between conductive heat transfer in solid and convective heat transfer in fluid.
Nondimensional number: modified Stanton number (St_i) conduction time number (T_i^*), Biot number (Bi_i)	$St_i = \left(\frac{4h_{conv} l_o}{\rho C_p u_{co} d_h}\right)_i$, $T_i^* = \left(\frac{\alpha_s l_o}{\delta^2 u_{co}}\right)_i$, and $Bi_i = \left(\frac{h_{conv}\delta}{k_s}\right)_i$ where reference velocity is the core inlet velocity and reference length is the heated height of the core.
Heat source number (Q_{si}) and reference temperature rise (ΔT_o)	$Q_{si} = \left(\frac{\dot{q}_s l_o}{\rho_s C_{p_s} u_{co} \Delta T_o}\right)_i$ and $\Delta T_o = \left(\frac{\dot{q}_s l_o}{\rho C_p u_{co}}\right)\left(\frac{a_{s_o}}{a_o}\right)$ These are obtained from an energy balance across the core.

Table 7
The similarity criteria and scale ratios for $1-\Phi$ natural circulation related to reactor system scaling analysis.

Similarity Criteria and Scaling Ratios	**Equations and Variables**
Scale ratios for: length (l_R), time constant ($\tau_{loop,R}$), fluid velocity ($u_{co,R}$)	For $1-\Phi$ natural circulation $l_R = (L_{th})_R$, $\tau_{loop,R} = \left(\frac{M_{sys}}{\rho_l u_{co} a_c}\right)_R = \left(\frac{l_{ref}\rho_l a_c}{\rho_l u_{co} a_c}\right)_R = \left(\frac{l_{ref}}{u_{co}}\right)_R$ and $u_{co,R} = \left(\frac{l_{ref}}{\tau_{loop}}\right)_R$ where ratio, R (subscript) represents a model-to-prototype (i.e., IET-to-SMR) ratio, and time-scale ratio less than 1.0 to allow for more reasonable velocity ratios for reduced-height test loops.
For steady-state natural circulation, requirements	• To maintain kinematic similarity, require geometric similarity in terms of cross-sectional flow areas, the ratio of each component flow area to the core area should be the same in the model as in the prototype. $\left(\frac{a_i}{a_c}\right)_R = 1$ • Full transient buoyancy and friction scaling adopted by steady-state natural circulation scaling approach, $Ri_R = \Pi_{Fl,R}$ • Full pressure testing with fluid property similitude to match temperature rise across the core, and assuming that the minor pressure losses and area ratios can be met by geometric similarity, $\left(\frac{L_{th}}{u_{co}^2}\right)_R = \left(\frac{l_{ref}}{d_h}\right)_R$.
Core power scale ratio ($q_{co,R}$), and power-to-volume (P/Vol.) scale ratio ($(q_{co}/V_i)_R$)	$q_{co,R} = \left(\frac{l_{ref}}{L_{th}}\right)_R \left(\frac{a_c u_{co}^3}{d_h}\right)_R$, which considers the simplified form of the friction number scale ratio with assuming fluid property similitude, $Ri_R = \Pi_{Fl,R}$.
Nondimensional number: Richardson number (Ri_R), and friction number ratio ($\Pi_{Fl,R}$)	$Ri_R = \left(\frac{L_{th}}{u_{co}^2}\right)_R = \left(\frac{\beta g q_{co} L_{th}}{\rho_l a_c C_{pl} u_{co}^3}\right)_R$ which can be obtained from any of the above two equations. $\Pi_{Fl,R} = \left(\frac{l_{ref}}{d_h}\right)_R$

Reactor system scaled-down test facilities were developed targeting specific reactor system and accident conditions. The NRC conducted a global system scaling analysis for the AP600 reactor to investigate the scaling performance of these three test facilities (i.e., APEX, ROSA, and SPES-2) for specific accident conditions (e.g., 1-in. cold leg break) for five accident time sequences (Wulff and Rohatgi, 1999). The NRC study intended to: (a) establish thermal hydraulics similarity among the test facilities—focusing on overall system response and the system-component's dynamic interaction; (b) rank global transport processes according to their importance; and (c) identify possible deviations from thermohydraulic similarity, or scale distortion.

Finally, based on considerations here from Section 3.2, provide a brief on how scaling approaches evolves in reactor system design with pros and cons for each approach and exhibit a general solution approach as discussed in following sections.

4. Findings and recommendations

SMRs represent some of the most innovative new reactor designs that promise scalability, flexibility, and potentially, a quicker route to deployment. However, the appropriate scaling of SMRs for design, development, and demonstration testing could face challenges specific to the early stages of design. It should be noted that SMRs vary in terms of capacity, fuel, coolant, system, and applicability, based on their unique design specifications and targeted applications. Some of these design-specific requirements may pose scaling challenges, for example, commercially availability of the pipe/tube diameter and thickness, which forced to use engineered scaling numbers instead of ideal scaling numbers. The specific challenges and associated recommendations are as follows:

- Natural circulation for passive system: Unlike conventional reactors, many SMRs employ passive safety systems. These systems rely on natural phenomena, such as gravity- or density-driven coolant flow and heat transport, like natural convection, rather than on active components. Scaling these passive systems (as discussed in section 3.1) can be particularly challenging because of their inherent dependencies on gravity, fluid properties, and geometric characteristics. Properly capturing the dynamics of these systems in a scaled model is crucial for ensuring both safety and functionality.
- Recommendations: Natural circulation thermal–hydraulic testing (e.g., IET/SET) is recommended for demonstrating passive reactor system. For instance, if the reactor power and height are scaled-down to a certain range, the testing facility (e.g., IET) might need to add an active pressure source (like a pump) to achieve the required flow for prototypical passive reactor system behavior. However, adequate justifications are required to communicate with the regulatory approval process how the basic physics phenomena and testing conditions are preserved.

Table 8
Additional energy scale ratios for $1-\Phi$ natural circulation related to PWR-type system scaling analysis.

Similarity Criteria and Scaling Ratios	Equations and Variables
Loop energy scale ratio	For $1-\Phi$ natural circulation $E_R = \left[\frac{(T_H - T_C)_o}{(T_M - T_C)_o}\right]_R = \frac{(T_H - T_C)_{o,R}}{(T_M - T_C)_{o,R}}$, which require, $(T_M - T_C)_{o,R} = 1$which indicates that the ratio of the temperature difference across the core to the difference between the mixed mean system temperature and the core inlet temperature remain fixed in the model.
Scale ratios for: steam generator power $\left(\frac{q_{SGo}}{q_{co}}\right)_R$, heat losses $\left(\frac{q_{loss,o}}{q_{co}}\right)_R$, and fluid-solid heat transfer (Q_s)	• The ratio of SG heat transfer to core power should be the same in the IET and the SMR, so, the ratios of the SG heat transfer and heat losses to core power should be preserved in the model. $\left(\frac{q_{SGo}}{q_{co}}\right)_R = 1$ • Heat loss scale ratio: due to its inherently higher surface area-to-volume ratio, relative heat losses will tend to be larger in the IET compared to the prototype, need thermal insulation and guard heating, $\left(\frac{q_{loss,o}}{q_{co}}\right)_R = 1$ • Fluid-solid heat transfer scale ratios: $St_{i_R} = T^*_{i_R} = Bi_{iR} = Q_{so_R}$ where, S_t and B_i ratios involve heat transfer coefficient T_i and B_i includes conduction thickness, which will not be automatically matched in the IET.
Number of heater rods ($N_{rod,R}$) and number of SG tubes ($N_{tube,R}$)	$N_{rod,R} = \frac{a_{c,R}}{d^2_{rod,R}}$ and $N_{tube,R} = \frac{a_{tube,R}}{d^2_{tube,R}}$
SG tube surface area ratio ($a_{S,tube,R}$)	$a_{S,tube,R} = N_{tube,R} \bullet d_{tube,R} \bullet L_R$ The tube heat flux ratio for all of the $a_{S,tube,R}$cases is equal to 1.0 indicating that prototypic average heat flux for each tube is assumed/desired.

- Compactness and integrated features: SMRs are distinguished by their compactness and the integration of various components including RCP(s), pressurizer, and steam generator(s), are incorporated into a single vessel (Zeliang et al., 2020; Hussein, 2020). This feature makes the scaling process and the development of test facilities challenging (Lien and Rohatgi, 2023; Bhowmik et al., 2023e). Due to manufacturing complexity and this component-in-component design is generally not practical in the test facility development (Lien and Rohatgi, 2023). For example, getting the boundary conditions for each component, and checking individual component performance testing would require separating the components in the IET or having independent SETs such as SG, containment, RCP and fuel bundle. The thermal hydraulics scaled test facility requires to cover all the POIs, especially those are high-rank and low SOK; for phenomena like boiling, condensation, stratification, entrainment, liquid–vapor mixing, SG level swelling, and CCFL. The design of reactor systems evolves through experimentation and modeling. Experimental facilities, such as IETs and SETs, demand more intricate instrumentation and control mechanisms (e.g., the inclusion of multiple control valves for break analysis). These facilities need to measure parameters like pressure, temperature, and mass flow rate at component interconnections and provide boundary conditions for simulations and modeling. However, the compact nature of these reactor designs often results in narrower margins for instrumentation, control, and measurements. Consequently, any alterations or distortions during the scaling process can have more significant effects on the performance of the reactor system.

Table 9
Scaling parameters related to IET and SET facilities scaling analyses/methods (Deng et al., 2019; Yun et al., 2004).

Name	Parameter Symbol	Linear Scaling	Power/ Volume	Three-level 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

- Recommendations: In test facilities (i.e., IET), there might be a need to separate the components (e.g., RPV and SG) to add instrumentation (e.g., flow meters, temperature sensors, pressure sensors) and control elements (e.g., control valves). This is done to obtain test data at boundary conditions to support simulation and modeling V&V. Instruments that cause minimal disturbance to fluid flow and heat transfer should be used.
- Non-linear behaviors in complex systems: Reactor systems are complex, much like IETs. When scaling such intricate systems, behaviors might not linearly translate (e.g., reactor core downcomer flow and cross flow) from the prototype to the scaled-down IET. This phenomenon was observed in many of the IET facilities developed using scaling methods (mostly, power-to-volume) at INL sites in the U.S. from the 1950 s to the 1980 s (Loomis, 1987; NEA, 2020; Lien and Rohatgi, 2023). Interactions between components, feedback loops, and system-wide effects can behave differently as the size and scale of the system changes. This results in non-linear scaling distortion, which presents further challenges in interpreting test data and in the V&V of computer codes/models.
- Recommendations: Appropriate scaling methods, such as H2TS, and similarity criteria and approaches are recommended. Moreover, it would be prudent to cross-check scaling numbers using different scaling approaches to avoid potential errors. It is advised to prioritize scaled-height than scaling volume and power. However, a scaled facility might require a reduction in height due to infrastructural limitations. The scaled height needs to be qualified for natural circulation (i.e., gravity and density-driven flow) for passive system design and analysis. During the accident simulation, time-scale may be distorted if the facility height is changed to smaller (lower) than prototype.
- Restrictions due to non-nuclear testing environments: Reactor system thermal-hydraulics testing often takes place in non-nuclear environments (i.e., without the use of nuclear fuel) to ensure economic feasibility, safety, and proper management, as discussed in the previous section (Bestion et al., 2017; NEA, 2020). However, this approach introduces challenges in scaling to replicate the exact conditions and behaviors of a reactor under nuclear operations. Achieving geometric and heat source similitude with representative commercially available electric heater rods is challenging. Additionally, most test facilities keep the diameter of the heater rods and the rod bundle configuration consistent with the prototypical reactor fuel and core (i.e., fuel assemblies) configuration (Yang et al., 2021; Yang et al., 2013). A similar approach is applied for the SG tube bundle (i.e., using the same diameter and tube bundle arrangement as the prototype). These considerations are supportive to scaling simulating the geometric conditions between prototype and test facility. However, there still need to cross-check using computer

codes/models if there any scaling distortions (e.g., metal heat/energy imbalance) introduced or not.

- Recommendations: If possible, electric heater rods with similar geometry (to mimic fuel rods) that also maintain scaled power capacity and a similar heater rod configuration should be used. Likewise, maintaining similar geometry for SG tubes is advisable to ensure similarity in heat transfer surface area and coolant passage. The use of special instrumentation that causes the minimal distortion in fluid (e.g., non-intrusive sensors) flow channel is also suggested (Maurin et al., 2022; Julio et al., 2022; Pacio et al., 2022; Carolina et al., 2023), which is important due to SMR compactness and integrated design feature as discussed earlier finding and recommendation.
- Fluid and system prototypic conditions: It is preferable to design IETs with the same fluid, system conditions such as temperature and pressure, and boundary conditions. However, scaling supported by adequate M&S could be leveraged to consider reduced pressure testing condition with acceptable scaling distortion. Accurate representation of fluid properties, such as viscosity, density, and thermal conductivity, is crucial for successful scaling. Boundary conditions, which define how the system interacts with its surroundings, are also paramount in capturing the true behavior of SMRs.
- Recommendations: Using the same fluid and prototypic conditions for i-PWR SMRs is recommended. For advanced and non-light water-cooled systems, there might be a need to use surrogate fluids and scaled test conditions (Zweibaum et al., 2020). However, applicability of the surrogate fluid decisions in the test facility design and scaling should be supported by the adequate simulation and modeling scoping results.
- IET vs. SET scaling: IET focuses on system-wide scaling, while SET concentrates on individual components like heat exchangers and pumps, emphasizing local thermal–hydraulic phenomena with appropriate measurement instrumentation (Bhowmik et al., 2023e; Upadhyaya et al., 2015; Korsah et al., 2016). Deciding between IET and SET involves striking a balance, determining whether to prioritize system-wide interactions or the fidelity of individual components. Therefore, the scaling and design scopes of IET and SET should be independent, yet complementary.
- Recommendations: In addition to a scaled IET, the suggested SET components for i-PWR SMR include SG, containment, RCP (if it is an active system), and rod bundle. SET scaling should focus on the detailed physics of components and local phenomena (e.g., local heat transfer coefficient, void fraction, critical heat flux, departure from nucleate boiling ratio [DNBR]), complemented by suitable measurements and instrumentation.

5. Conclusion and path forward

This study presents a general approach for scaling analysis and similarity criteria for water-cooled reactor systems, focusing on the PWR-type SMRs. Reactor system scaling and transient studies provide an in-depth understanding of system behavior and various geometric and operational settings. The scaling and similarity approach that is introduced begins with the foundational governing equations, determining the nondimensional parameters that dictate the fluid dynamics in circulation. In many cases, a need exists to use surrogate fluids and scaled-down system conditions to simplify testing and facility development. However, it is crucial to ensure appropriate scaling ratios and similarities between the IET and the prototype to adequately characterize flow, heat transfer, and related phenomena. Below are a few other general and specific observations and findings:

- The path to scaling SMRs is fraught with challenges, but understanding these challenges is the first step toward overcoming them. The unique design considerations of SMRs—for example, a passive SMR system with no coolant pump varies from the active SMR system with a coolant pump—necessitate a deep understanding of systems, structures, and components, compact design principles, and tightly coupled component interactions.
- The governing equations and associated nondimensional parameters offer a detailed understanding of how different geometric configurations impact the system's scaling. Due to SMR compact features, the test facility may need some alteration in component layout to reduce manufacturing and instrumentation complexity, and to preserve the major thermal hydraulic phenomena.
- Given that some-level of distortions are unavoidable, researchers employ compensation techniques—supported by adequate modeling and simulations—to adjust and correct the observed behaviors in scaled models. Validating these techniques is essential to ensure the compensated scaled model still provides accurate, actionable insights into the full-sized reactor's behaviors.

The challenges listed below are associated with finalizing the scaling analysis and proceeding to facility design and development:

- During the reactor systems design phase, theoretical and computational work was translated into tangible, physical systems. The development of the experimental facility is pivotal to obtained required experimental dataset to support V&V of the computer code/ models for successful completion of the EMDAP towards reactor system licensing. This stage ensured the facility was robust and capable, and set up to provide relevant and reliable test data for specific test (e.g., accidents and anticipated transients) conditions.
- Key components, such as valves and instrumentation, are crucial in IET design and development, even if they are not often considered during the IET scaling analysis. Valves and instrumentation not only facilitate operations, but also ensure that the captured data is accurate and reflects the conditions being tested, thereby bridging the gap between theory and practice.
- The experiments and test data offered empirical evidence supporting the observations, hypotheses, and predictions made during the scaling analysis. Experimental data from a well-defined scaled facility is invaluable, serving not only as a validation tool, but also as a guide for future research and development of reactor system design and development.

In conclusion, it is evident that scaling provides the means to process information in an efficient manner. It is also clear that scaling analysis is considered a primary step for developing a thermal hydraulics test facility for reactor system design and analysis. Keeping this in mind this study presented scaling analysis with associated governing equations/ models, similarity criteria, scaling ratios in a systematic form, as well as addresses challenges and offering recommendations for experimental thermal hydraulics facility development for PWR-type SMR. While a certain level of scaling distortions in test facilities are a reality, the industry's ability to identify, compensate for, and validate against these distortions will determine a practical method how to successfully perform scaling for the SMR of interest. The findings, research gaps, and recommendations serve as a robust foundation for the scaling of reactor systems to support design, development, and demonstration.

6. 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.

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CRediT authorship contribution statement

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Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

Data availability

No data was used for the research described in the article.

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Disclaimer

References

Abdellatif, H.H., Bhowmik, P.K., Arcilesi, D., Sabharwall, P., 2024. Evaluating direct vessel injection accident-event progression of AP1000 and key figures of merit to support the design and development of water-cooled small modular reactors. Nucl. Eng. Technol. https://doi.org/10.1016/j.net.2024.01.049.

Abdellatif, H.H., Bhowmik, P.K., Arcilesi, D., Sabharwall, P., 2024a. Accident event progression, gaps, and key performance indicators for steam generator tube rupture events in water-cooled SMRs: A review. Prog. Nucl. Energy 168, 105021. https://doi.org/10.1016/j.pnucene.2023.105021.

Abdellatif, H.H., Ambrosini, W., Arcilesi, D., Bhowmik, P.K., Sabharwall, P., 2024b. Flow Instabilities in boiling channels and their suppression methodologies—A review. Nucl. Eng. Des. 421, 113114 https://doi.org/10.1016/j.nucengdes.2024.113114.

Aksan, N., 2019. An overview on thermal-hydraulic phenomena for water cooled nuclear reactors; Part I: SETs, and ITFs of PWRs, BWRs. Vvers. Nucl. Eng. Des. 354, 110212 https://doi.org/10.1016/j.nucengdes.2019.110212.

Bae, B., Jung, J., Yu, J.Y., 2023. Hydraulic performance and flow resistance tests of various hydraulic parts for optimal design of a reactor coolant pump for a small modular reactor. Nucl. Eng. Technol. 55 (3), 1181–1190. https://doi.org/10.1016/j.net.2022.10.042.

Bajorek, S.M., Cheung, F.B., 2019. Rod bundle heat transfer thermal-hydraulic program. Nucl. Technol. 205 (1–2), 307–327. https://doi.org/10.1080/00295450.2018.1510697.

Bestion, D., D'Auria, F.S., Lien, P., Nakamura, H., 2017. A State-of-the-Art Report on Scaling in System Thermal-Hydraulics Applications to Nuclear Reactor Safety and Design, NEA/CSNI/R(2016)14, March 2017, Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI), Washington, D.C., USA. Available at: https://www.oecd-nea.org/upload/docs/application/pdf/2021-02/csni-r2016-14.pdf (accessed 6 February 2024).

Bhowmik, P.K., Sabharwall, P., 2023a. Non-Nuclear Test Facility Development to Support Verification and Validation Needs. INL/CON-23-73945, Idaho National Laboratory, Idaho Falls, ID, USA. Available at: https://inldigitallibrary.inl.gov/sites/sti/sti/Sort_67054.pdf (accessed 6 February 2024).

Bhowmik, P.K., Sabharwall, P., Johnson, J., O'Brien, J. E., Lietwiler, C., 2023d. Advances in Integral and Separate Effects Experiments for Water Cooled Small Modular Reactors. INL/CON-22-70513, Idaho National Laboratory, Idaho Falls, ID, USA. Available at: https://inldigitallibrary.inl.gov/sites/sti/sti/Sort_66222.pdf (accessed 6 February 2024).

Bhowmik, P.K., Sabharwall, P., 2023b. Sizing and selection of pressure relief valves for high-pressure thermal-hydraulic systems. Processes 12 (1), 21. https://doi.org/10.3390/pr12010021.

Bhowmik, P.K., Suh, K.Y., 2021. Flow mapping using 3D Full-scale CFD simulation and hydrodynamic experiments of an ultra-supercritical turbine's combined valve for nuclear power plant. Int. J. Energy Environ. Eng. 12 (3), 365–381. https://doi.org/10.1007/s40095-021-00394-0.

Bhowmik, P.K., Shamim, J.A., Chen, X., Suh, K.Y., 2021a. Rod bundle thermal-hydraulics experiment with water and water-Al_2O_3 nanofluid for small modular reactor. Ann. Nucl. Energy 150, 107870. https://doi.org/10.1016/j.anucene.2020.107870.

Bhowmik, P.K., Schlegel, J.P., Kalra, V., Mills, C., Usman, S., 2021b. Design of condensation heat transfer experiment to evaluate scaling distortion in small modular reactor safety analysis. ASME J. Nucl. Eng. Radiat. Sci. 7 (3), 031406 https://doi.org/10.1115/1.4050211.

Bhowmik, P.K., Shamim, J.A., Sabharwall, P., 2023. A review on the sizing and selection of control valves for thermal hydraulics for reactor system applications. Prog. Nucl. Energy 164, 104887. https://doi.org/10.1016/j.pnucene.2023.104887.

Bhowmik, P.K., Ormiston, S.J., Schlegel, J.P., Chowdhury, D., 2023a. State-of-the-art and review of condensation heat transfer for small modular reactor passive safety: computational studies. Nucl. Eng. Des. 410, 112366 https://doi.org/10.1016/j.nucengdes.2023.112366.

Bhowmik, P.K., Usman, S., Schlegel, J.P., 2023b. Film condensation with high heat fluxes and scaled experiments using pure steam for reactor containment cooling. Appl. Therm. Eng. 229, 120610 https://doi.org/10.1016/j.applthermaleng.2023.120610.

Bhowmik, P.K., Perez, C.E.E., Fishler, J.D., Prieto, S.A.B., Reichow, I.D., Johnson, D.T., Sabharwall, P., O'Brien, J.E., 2023e. Integral and separate effects test facilities to support water cooled small modular reactors: A review. Prog. Nucl. Energy 160, 104697. https://doi.org/10.1016/j.pnucene.2023.104697.

Bhowmik, P.K., 2021. Scaling, Experiments, and Simulations of Condensation Heat Transfer for Advanced Nuclear Reactors Safety. Ph.D. Dissertation, 2999, Missouri University of Science and Technology, Rolla, MO, USA. Available at: https://scholarsmine.mst.edu/doctoral_dissertations/2999 (accessed 6 February 2024).

Black, R.L., 2021. Chapter 11 – Licensing of Small Modular Reactors (SMRs). In: Carelli, M.D., Ingersoll, D.T. (Eds.), Handbook of Small Modular Nuclear Reactors. Woodhead Publishing, Sawston, Cambridge, U.K, pp. 279–292.

Carbiener, W.A., Cudnik, R.A., 1969. Similitude considerations for modeling nuclear reactor blowdown. Trans. Am. Nucl. Soc. 12, 361.

Cheng, C., Lu, D., Su, Q., Huang, Y., Wu, X., 2023. Experimental study on distortion of scale-down methodology for natural circulation steady-state phenomena. Ann. Nucl. Energy 182, 109600. https://doi.org/10.1016/j.anucene.2022.109600.

D'Auria, F., 2023. Scaling, passive systems, and the AP-1000. Nucl. Sci. Eng. 197 (5), 987–999. https://doi.org/10.1080/00295639.2023.2178874.

D'Auria, F., Bestion, D., 2022. Nuclear thermal-hydraulic phenomena: bases and challenges. Nucl. Technol. 208 (6), 990–1011. https://doi.org/10.1080/00295450.2021.1997059.

D'Auria, F., Galassi, G.M., 2010. Scaling in nuclear reactor system thermal-hydraulics. Nucl. Eng. Des. 240 (10), 3267–3293. https://doi.org/10.1016/j.nucengdes.2010.06.010.

Deng, C., Zhang, X., Yang, Y., Yang, J., 2019. Research on scaling design and applicability evaluation of integral thermal-hydraulic test facilities: A review. Ann. Nucl. Energy 131–290. https://doi.org/10.1016/j.anucene.2019.03.042.

U.S. Department of Energy, Office of Nuclear Energy (DOE-NE), 2023a. NRC Certifies First U.S. Small Modular Reactor Design. DOE-NE, Washington, D.C., USA. Available at: https://www.energy.gov/ne/articles/nrc-certifies-first-us-small-modular-reactor-design (accessed 11 April 2024).

U.S. Department of Energy, Office of Nuclear Energy (DOE-NE), 2023b. Advanced Reactor Demonstration Program. DOE-NE, Washington, D.C., USA. Available at: https://www.energy.gov/ne/advanced-reactor-demonstration-program (accessed 6 February 2024).

Dzodzo, M.B., 2023. Scaling and connection with computational fluid dynamics. Nucl. Eng. Des. 401, 112086 https://doi.org/10.1016/j.nucengdes.2022.112086.

Dzodzo, M.B., Oriolo, F., Ambrosini, W., Ricotti, M., Grgic, D., Ferri, R., Meloni, P., 2019. Application of fractional scaling analysis for development and design of integral effects test facility. J. Nucl. Eng. Radiat. Sci. 5 (4), 041208 https://doi.org/10.1115/1.4042496.

Ghosh, S.T., Esmaili, H., Hathaway, A., Bixler, N., Brooks, D., Dennis, M., Wagner, K., 2021. State-of-the-art reactor consequence analyses project: uncertainty analyses for station blackout scenarios. Nucl. Technol. 207 (3), 441–451. https://doi.org/10.1080/00295450.2021.1875737.

Hertlein, R.J., Umminger, K., Kliem, S., Prasser, H.M., Höhne, T., Weiss, F.P., 2003. Experimental and numerical investigation of boron dilution transients in pressurized water reactors. Nucl. Technol. 141 (1), 88–107. https://doi.org/10.13182/NT03-A3353.

Hui, K., Chen, W., Li, S., Zhao, Q., Yan, J., 2021. "Experimental study on transient thermal-hydraulic characteristics of an open natural circulation for the passive containment cooling system. Int. J. Heat Mass Transf. 179, 121680 https://doi.org/10.1016/j.ijheatmasstransfer.2021.121680.

Hussein, E.M., 2020. Emerging small modular nuclear power reactors: A critical review. Physics Open 5, 100038. https://doi.org/10.1016/j.physo.2020.100038.

International Atomic Energy Agency (IAEA), 2022. Small Modular Reactors: A New Nuclear Energy Paradigm, 22-02957E, IAEA, Vienna, Austria. Available at: https://nucleus.iaea.org/sites/smr/Shared%20Documents/Small%20Modular%20Reactors%20a%20new%20nuclear%20energy%20paradigm.pdf (accessed 6 February 2024).

Ishii, M., and Kataoka, I., 1983. Similarity Analysis and Scaling Criteria for LWRs Under Single-Phase and Two-Phase Natural Circulation. NUREG/CR-3267/ANL-83-32, Argonne National Laboratory, Argonne, IL, USA. https://doi.org/10.2172/6312011.

Ishii, M., Kataoka, I., 1984. Scaling laws for thermal-hydraulic system under single phase and two-phase natural circulation. Nucl. Eng. Des. 81 (3), 411–425. https://doi.org/10.1016/0029-5493(84)90287-5.

Katono, K., Fujimoto, K., Arai, T., Miwa, S., Hibiki, T., 2022. Experimental investigation of rod pitch effect on void fraction in 5×5 rod-bundle under elevated pressure conditions. Int. J. Heat Mass Transf. 195, 123035 https://doi.org/10.1016/j.ijheatmasstransfer.2022.123035.

Kim, K.K., Lee, W., Choi, S., Kim, H.R., Ha, J., 2014. SMART: the first licensed advanced integral reactor. J. Energy Power Eng. 8 (1), 94. https://www.davidpublisher.com/Public/uploads/Contribute/559ddf2f554d6.pdf.

Kim, G.W., Park, G.C., Cho, H.K., 2021. Scaling analysis for single-phase natural circulation under dynamic motion and its verification using MARS-KS code. Ann. Nucl. Energy 159, 108308. https://doi.org/10.1016/j.anucene.2021.108308.

Korsah, K., Kisner, R. A., Britton Jr, C. L., Ramuhalli, P., Wootan, D. W., Anheier Jr, N. C., ... & Stanford, A. (2016). Assessment of sensor technologies for advanced reactors. Oak Ridge National Laboratory (ORNL), Oak Ridge, TN (United States).

Lee, B., Kim, S.I., Ha, K.S., 2022. "Separate and integral effect tests of aerosol retention in steam generator during tube rupture accident. Nucl. Eng. Technol. 54 (7), 2702–2713. https://doi.org/10.1016/j.net.2022.01.014.

Li, Y.Q., Wang, H., Ye, Z.S., Chen, L., Qin, B.K., 2012. ACME test facility scaling analysis report (Ver. A). State Nuclear Power Research and Development Certer, K001BG0016.

Li, X., Li, H., Liu, Y., Wu, Q., Zhan, B., 2020. Numerical scaling assessment on natural circulation in core makeup tank. Ann. Nucl. Energy 140, 107105. https://doi.org/10.1016/j.anucene.2019.107105.

Li, L., Long, B., Yu, J., Su, Q., Lu, D., Wu, Y., Qiu, S., Su, G., Tian, W., Wu, X., Liu, J., Ju, P., 2021. Research on scaling analysis and design parameters of integral test facilities for PWR. Ann. Nucl. Energy 163, 108557. https://doi.org/10.1016/j.anucene.2021.108557.

Liao, J., Ferroni, P., Wright, R.F., Bachrach, U., Scobel, J.H., Sofu, T., Tentner, A.M., Lee, S.J., Epstein, M., Frignani, M., Tarantino, M., 2021. Development of phenomena identification and ranking table for Westinghouse lead fast reactor's safety. Prog. Nucl. Energy 131, 103577. https://doi.org/10.1016/j.pnucene.2020.103577.

Lien, P.H., Rohatgi, U.S., 2023. "Scaling challenges in small modular reactor. Nucl. Eng. Des. 407, 112309 https://doi.org/10.1016/j.nucengdes.2023.112309.

Long, J., Zhang, B., Yang, B.-W., Wang, S., 2021. Review of researches on coupled system and CFD Codes. Nucl. Eng. Technol. 53 (9), 2775–2787. https://doi.org/10.1016/j.net.2021.03.027.

Loomis, G.G., 1987. Summary of the Semiscale Program (1965 to 1986). NUREG/CR-4945/EGG-2509, July 1987, U.S. Nuclear Regulatory Commission, Washington, D. C., USA. https://doi.org/10.2172/6302855.

Lorduy-Alós, M., Gallardo, S., Verdú, G., 2021. Appliance of scaling methodologies for designing counterpart experiments dominated by natural circulation phenomena. Nucl. Eng. Des. 384, 111489 https://doi.org/10.1016/j.nucengdes.2021.111489.

Maurin, L., Roussel, N., Laffont, G., 2022. Non-intrusive pipe internal pressure measurement. 6-11 March 2022. 19th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH19), Brussels, Belgium. https://cea.hal.science/cea-03687703/document.

Modro, S.M., Fisher, J.E., Weaver, K.D., Reyes, J.N., Groome, J.T., Babka, P., Carlson, T. M., 2003. Multi-Application Small Light Water Reactor Final Report. INEEL/EXT-04-01626, Idaho National Laboratory, Idaho Falls, ID, USA. Available at: https://inldigitallibrary.inl.gov/sites/sti/sti/2546531.pdf (accessed 6 February 2024).

Morton, T.J., 2019. Primary System Test Facility Review. INL/EXT-19-55536, Idaho National Laboratory, Idaho Falls, ID, USA. https://doi.org/10.2172/1562115.

Mundy, T., 2021. "NuScale Power: A Mature, Near-Term Deployable, SMR Solution. DOE-NuScale-08928-12, NuScale Power, LLC, Corvallis, OR, USA. Available at: https://www.osti.gov/servlets/purl/1813797 (accessed 6 February 2024).

Nahavandi, A.N., Castellana, F.S., Moradkhanian, E.N., 1979. "Scaling laws for modeling nuclear reactor systems. Nucl. Sci. and Eng. 72 (1), 75–83. https://doi.org/10.13182/NSE79-A19310.

NEA, 2020. Loss-of-Fluid Test (LOFT) Program, Nuclear Energy Agency, Available at: https://www.oecd-nea.org/jcms/pl_25963/loss-of-fluid-test-loft-project (accessed 6 February 2024).

Ni, D., Zhang, N., Gao, B., Li, Z., Yang, M., 2020. Dynamic measurements on unsteady pressure pulsations and flow distributions in a nuclear reactor coolant pump. Energy 198, 117305. https://doi.org/10.1016/j.energy.2020.117305.

U.S. Nuclear Regulatory Commission (NRC), 2005. Transient and Accident Analysis Methods, Regulatory Guide 1.203, December 2005. NRC, Washington, D.C., USA. Available at: https://www.nrc.gov/docs/ML0535/ML053500170.pdf (accessed 6 February 2024).

U.S. Nuclear Regulatory Commission (NRC), 2020. Design Certification - NuScale US600, Standard Design Certification for an Integrated Pressurized Water Reactor Assembly Comprised of Twelve NuScale Small Modular Reactors (SMR). NRC, Washington, D. C., USA. Available at: https://www.nrc.gov/docs/ML2024/ML20247J564.pdf (accessed 6 February 2024).

Obaidurrahman, K., Arul, A.J., Ramakrishnan, M., Singh, O.P., 2021. Chapter 8 – Nuclear Reactor Safety. In: Mohanakrishnan, P., Singh, O.P., Umasankari, K. (Eds.), Physics of Nuclear Reactors. Academic Press, Cambridge, MA, USA, pp. 449–510.

Pacio, J., Van Tichelen, K., Eckert, S., Wondrak, T., Di Piazza, I., Lorusso, P., Tarantino, M., Daubner, M., Litfin, K., Ariyoshi, G., Obayashi, H., Sasa, T., 2022. Advanced Thermal-Hydraulic experiments and instrumentation for heavy liquid metal reactors. Nucl. Eng. Des. 399, 112010 https://doi.org/10.1016/j.nucengdes.2022.112010.

Park, H.S., Bae, H., Ryu, S.U., Jeon, B., Yang, J.H., Yi, S.J., Chung, Y.J., 2023. "Thermal-hydraulic research supporting the development of SMART. Nucl. Technol. 209 (10), 1617–1635. https://doi.org/10.1080/00295450.2023.2217370.

Park, J.S., Kim, J.W., Lee, J.S., 2020. Complete and homologous pump characteristics for a reactor coolant pump. Nucl. Eng. Des. 357, 110425 https://doi.org/10.1016/j.nucengdes.2019.110425.

Reyes, J. N., Hochreiter, L., 1998. Scaling analysis for the OSU AP600 test facility (APEX), Nucl. Eng. Des. 86 (1–2), 1998, p53-109, ISSN 0029-5493. https://doi.org/10.1016/S0029-5493(98)00218-0.

Rohatgi, U.S., Kaizer, J.S., 2020. Historical perspectives of BEPU research in US. Nucl. Eng. Des. 358, 110430 https://doi.org/10.1016/j.nucengdes.2019.110430.

Ruzicka, M.C., 2008. On dimensionless numbers. Chem. Eng. Res. Des. 86 (8), 835–868. https://doi.org/10.1016/j.cherd.2008.03.007.

Sainati, T., Locatelli, G., Brookes, N., 2015. Small modular reactors: licensing constraints and the way forward. Energy 82, 1092–1095. https://doi.org/10.1016/j.energy.2014.12.079.

Sam, R., Sainati, T., Hanson, B., Kay, R., 2023. Licensing small modular reactors: A state-of-the-art review of the challenges and barriers. Prog. Nucl. Energy 164, 104859. https://doi.org/10.1016/j.pnucene.2023.104859.

Schlegel, J.P., Bhowmik, P.K., 2023. Chapter 14 – Small Modular Reactors. In Wang, J., Talabi, S., and Bilbao y León, S. (Eds.), Nuclear Power Reactor Designs: From History to Advances, Academic Press, Cambridge, MA, USA. pp. 283–308. https://doi.org/10.1016/B978-0-323-99880-2.00014-X.

Schultz, R. R., Gougar, H., Vegendla, P., Obabko, A., Thomas, J., 2017. Identification and Characterization of Thermal Fluid Phenomena Associated with Selected Operating/Accident Scenarios in Modular High-Temperature Gas-Cooled Reactors. INL/EXT-17-43218, Idaho National Laboratory, Idaho Falls, ID, USA. https://doi.org/10.2172/1408770.

Shaw, R.A., Rouhani, S.Z., Larson, T.K., Dimenna, R.A., 1985. Development of a phenomena identification and ranking table (PIRT) for thermal hydraulic phenomena during a PWR large-break LOCA. US Nuclear Regulatory Commission, Washington, DC 20555, NUREG/CR-5074, November 1985.

Subki, H., 2016. Technology Development Status Report: Small Modular Reactors. 10th GIF-IAEA Interface Meeting, IAEA, Vienna, Austria.

Thomas, S., Ramana, M.V., 2022. A hopeless pursuit? National efforts to promote small modular nuclear reactors and revive nuclear power. Wiley Interdiscip. Rev. Energy Environ. 11 (4), e429.

Todreas, N.E., Kazimi, M.S., nd Massoud, M., 2021. Nuclear Systems, Volume II: Elements of Thermal Hydraulic Design, 2nd Edition. CRC Press, Boca Raton, FL, USA.

Upadhyaya, B.R., Lish, M.R., Hines, J.W., Tarver, R.A., 2015. Instrumentation and control strategies for an integral pressurized water reactor. Nucl. Eng. Technol. 47 (2), 148–156. https://doi.org/10.1016/j.net.2015.01.001.

Vaya Soler, A., Berthélemy, M., Verma, A., White, A., Sexton Nick, K., Bilbao y León, S., Kwong, G., Vásquez-Maignan, X., and Rouyer, V., 2021. Small Modular Reactors: Challenges and Opportunities, 7560, Nuclear Energy Agency (NEA), Boulogne-Billancourt, France. Available at: https://www.oecd-nea.org/upload/docs/application/pdf/2021-03/7560_smr_report.pdf (accessed 6 February 2024).

Wang, L.S., Yan, B.H., 2021. "The scaling technology in nuclear reactor thermal hydraulic. Ann. Nucl. Energy 161, 108440. https://doi.org/10.1016/j.anucene.2021.108440.

Wilson, G.E., Boyack, B.E., 1998. The role of the PIRT process in experiments, code development and code applications associated with reactor safety analysis. Nucl. Eng. Des. 186 (1–2), 23–37. https://doi.org/10.1016/S0029-5493(98)00216-7.

Wulff, W., Rohatgi, U.S., 1999. System Scaling for the Westinghouse AP600 Pressurized Water Reactor and Related Test Facilities: Analysis and Results, Division of Systems Technology, Office of Nuclear Regulatory Research. U.S. Nuclear Regulatory Commission, Washington, D.C., USA.

Xing, J., Yu, P., Meng, Z., Lv, D., Shang, Z., Sun, Z., 2023. Depressurization strategy of automatic depressurization system for advanced pressurized water reactor. Ann. Nucl. Energy 191, 109911. https://doi.org/10.1016/j.anucene.2023.109911.

Yang, B.W., Anglart, H., Han, B., Liu, A., 2021. Progress in rod bundle CHF in the past 40 years. Nucl. Eng. Des. 376, 111076 https://doi.org/10.1016/j.nucengdes.2021.111076.

Yang, J.-H., Bae, H., Ryu, S.-U., Jeon, B.G., Yi, S.-J., Park, H.-S., 2020. Comparison of two different sized small-break LOCAs on the passive safety injection line using SMART-ITL data. Nucl. Technol. 206 (9), 1421–1435. https://doi.org/10.1080/00295450.2020.1775450.

Yang, X., Schlegel, J.P., Liu, Y., Paranjape, S., Hibiki, T., Ishii, M., 2013. Experimental study of interfacial area transport in air-water two phase flow in A scaled 8×8 BWR rod bundle. Int. J. Multiph. Flow 50, 16–32. https://doi.org/10.1016/j.ijmultiphaseflow.2012.10.006.

Yao, H., Chen, G., Lu, K., Wu, Y., Tian, W., Su, G., Qiu, S., 2021. Study on the systematic thermal-hydraulic characteristics of helical coil once-through steam generator. Ann. Nucl. Energy 154, 108096. https://doi.org/10.1016/j.anucene.2020.108096.

Yu, X.G., Choi, K.Y., 2016. Systematic and exact scaling analysis of the single-phase natural circulation flow: the hydraulic similarity. Prog. Nucl. Energy 89, 78–87. https://doi.org/10.1016/j.pnucene.2016.02.001.

Yu, H., Ju, H., Wang, M., Zhang, J., Qiu, S., Tian, W., Su, G.H., 2020. Study of boron diffusion models and dilution accidents in nuclear reactor: A comprehensive review. Ann. Nucl. Energy 148, 107659. https://doi.org/10.1016/j.anucene.2020.107659.

Yu, Q., Qiu, S., Wang, C., Zhang, D., Tian, W., Su, G.H., 2023. An experimental review of steam generator tube rupture accident in lead-cooled fast reactors: thermal-hydraulic experiments classification and methods introduction. Prog. Nucl. Energy 160, 104711. https://doi.org/10.1016/j.pnucene.2023.104711.

Yun, B.J., Cho, H.K., Euh, D.J., Song, C.H., Park, G.C., 2004. Scaling for the ECC bypass phenomena during the LBLOCA reflood phase. Nucl. Eng. Des. 231 (3), 315–325. https://doi.org/10.1016/j.nucengdes.2004.04.005.

Zeliang, C., Mi, Y., Tokuhiro, A., Lu, L., Rezvoi, A., 2020. Integral PWR-type small modular reactor developmental status, design characteristics and passive features: A review. Energies 13 (11), 2898. https://doi.org/10.3390/en13112898.

Zuber, N., 1991. A hierarchical, two-tiered scaling analysis, an integrated structure and scaling methodology for severe accident technical issue resolution. US Nuclear Regulator Commission.

Zuber, N., 2001. The effects of complexity, of simplicity, and of scaling in thermal hydraulics. Nucl. Eng. Des. 204 (1–3), 1–27. https://doi.org/10.1016/S0029-5493(00)00324-1.

Zuber, N., Rohatgi, U.S., Wulff, W., Catton, I., 2005. Application of Fractional Scaling Analysis (FSA) to Loss of Coolant Accidents (LOCA), Part 1: Methodology Development. In: Lemonnier, H. (ed.)., International Topical Meeting on Nuclear Reactor Thermal-Hydraulics (NURETH-11), CEA Grenoble, Dept. Etudes des Reacteurs (DER), 2–6 October 2005, Avignon, France.

Zuber, N., Wilson, G.E., Ishii, M., Wulff, W., Boyack, B.E., Dukler, A.E., Griffith, P., Healzer, J.M., Henry, R.E., Lehner, J.R., Levy, S., Moody, F.J., Pilch, M., Sehgal, B.R., Spencer, B.W., Theofanous, T.G., Valente, J., 1998. An integrated structure and scaling methodology for severe accident technical issue resolution: development of methodology. Nucl. Eng. Des. 186 (1–2), 1–21. https://doi.org/10.1016/S0029-5493(98)00215-5.

Zuber, N., Rohatgi, U.S., Wulff, W., Catton, I., 2007. Application of fractional scaling analysis (FSA) to loss of coolant accidents (LOCA) methodology development. Nucl. Eng. Des. 237 (15–17), 1593–1607. https://doi.org/10.1016/j.nucengdes.2007.01.017.

Zweibaum, N., Blandford, E., Gerardi, C., Peterson, P., 2020. Scaling methodology for integral effects tests in support of fluoride salt–cooled high-temperature reactor technology. Nucl. Sci. Eng. 194 (8–9), 793–811. https://doi.org/10.1080/00295639.2019.1710976.