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High-Temperature Gas-Cooled Reactor Research Survey and Overview: Preliminary Data Platform Construction for the Nuclear Energy University Program

August 2022

Changing the World's Energy Future

Sunming Qin, Minseop Song, Stefan Hans Vietz, Cam Binh T Pham, Mitchell A Plummer, Gerhard Strydom

Idaho National Laboratory

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SUMMARY

Since the U.S. Department of Energy Office of Nuclear Energy initiated the Nuclear Energy University Program (NEUP) in 2009, there are 29 NEUP projects focusing on high-temperature gas-cooled reactor (HTGR) research up to July 2022. The resultant research product, either experimental or computational, were published as final NEUP reports, journal articles and conference proceedings. However, these federally funded products have been scattered and sometimes cannot be easily accessed.

To improve access to this valuable HTGR validation data and optimize the return on the significant investment made by the Department of Energy, the Advanced Reactor Technologies (ART) Gas-Cooled Reactor (GCR) program started a survey of completed and ongoing HTGR NEUP projects to develop a public-access database specific for HTGRs applications that can be used to retrieve computational fluid dynamics and system code validation data. This effort will help guide future NEUP-funded research, define new state of the ART Phenomena Identification and Ranking Table (PIRT), and promote the usage of this data in the codes validation matrices. This report provides an overview of the NEUP-funded HTGR-related research projects from Fiscal Year (FY) 2009–2021 and identifies validation knowledge gaps still existing in HTGR thermal-fluid research.

A preliminary data platform has been developed for the 29 NEUP projects investigating HTGR thermal hydraulics, including their final reports as well as the available scientific publications. As an ultimate goal for this work, the ART-GCR program will create a central database at Idaho National Laboratory to identify, organize, and store these datasets generated by experimental investigations or computational models, experimental facility descriptions, and publicly-available academic products from the HTGR-related NEUP projects and provide future guidance for the storage and transmission of important project documentations for later NEUP projects as well.

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ACRONYMS

ART	Advanced Reactor Technologies
CCNY	City College of New York
CFD	computational fluid dynamics
DLOFC	depressurized loss of forced cooling
DOE	Department of Energy
FY	Fiscal Year
GA	General Atomics
GCR	gas-cooled reactor
HTGR	High-Temperature Gas-cooled Reactor
HTTF	High Temperature Test Facility
HWA	hot-wire anemometry
INL	Idaho National Laboratory
IRP	Integrated Research Project
LOFC	loss of forced cooling
MHTGR	Modular High-Temperature Gas-Cooled Reactor
NEUP	Nuclear Energy University Program
NGNP	Next Generation Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NSTF	Natural convection Shutdown heat removal Test Facility
ORNL	Oak Ridge National Laboratory
OSU	Oregon State University
PBR	pebble-bed reactors
PI	Principal Investigator
PIRT	Phenomena Identification and Ranking Table
PIV	Particle Image Velocimetry
PLOFC	pressurized loss of forced cooling
RANS	Reynolds-Averaged Navier-Stokes
RCCS	reactor cavity cooling system
RoBuT	Rotatable Buoyancy Tunnel
TAMU	Texas A&M University
TRISO	tristructural isotropic
UM	University of Michigan
USU	Utah State University

UW	University of Wisconsin
VHTR	Very-High-Temperature Gas-Cooled Reactor
V&V	verification and validation

High-Temperature Gas-Cooled Reactor Research Survey and Overview

Preliminary Data Platform Construction for the Nuclear Energy University Program

1. INTRODUCTION AND BACKGROUND

The high-temperature gas-cooled reactor (HTGR) is an advanced nuclear reactor design that can supply high-temperature heat energy of 750–950°C by using a specialized fuel coated with ceramics, such as carbon and silicon carbide, inert helium gas as a coolant, and graphite as a moderator. In recent decades, HTGRs have been attracting more attention from the community due to their potential to provide high-temperature process heat in addition to their high thermal-to-electric power conversion efficiency and inherent safety features [1, 2]. The concept of HTGRs, or known as Very-High-Temperature Gas-Cooled Reactor (VHTR), was promulgated in the Generation IV technology roadmap in 2001 as one of the Next Generation Nuclear Power Plant (NGNP) designs [3]. Researchers have concluded that HTGRs have a high technology-readiness-level, which will enable near-term deployment in flexible energy missions [4, 5]. Beyond electricity, HTGRs provide another impactful way to increase the application of nuclear energy and reduce greenhouse gas emissions by substituting high-quality nuclear-generated steam for fossil fuels in energy-intensive industrial applications given their high operating temperatures. Due to their high outlet temperatures for cogeneration and process heat production capabilities, applications that could use the HTGR high-temperature process heat include desalination, hydrogen production, hydrogasification, methanol and ammonia synthesis, and direct iron ore reduction, all without the production of carbon dioxide (CO_2) emissions.

Since the United States (U.S.) Department of Energy (DOE)'s Office of Nuclear Energy initiated the Nuclear Energy University Program (NEUP) in 2009, a total of 29 NEUP projects focused on HTGR thermal-fluid experiments were funded up to Fiscal Year (FY) 2021. This represents a total DOE investment of approximately $23 million over the 12-year period, covering thermal-fluid phenomena important to both pebble-bed and prismatic HTGR designs. The NEUP projects have produced a large amount of high-quality experimental and computational data that were published in final project reports, journal articles, dissertations, and conference proceedings, but in most cases, the actual data sets, supporting information, such as facility and instrumentation descriptions, and developed simulation models have to be retrieved from the single university, making it difficult if not impossible to have a clear picture of all the work performed in the last decade. To the authors' best knowledge, a database that organizes and summarizes these NEUP-funded projects specific for HTGRs research does not currently exist.

1.1 Work Scope and Objectives

NEUP projects have produced valuable results with both computational and experimental studies, but detailed information regarding the project products is not always available online. For example, to develop a detailed computational fluid dynamics (CFD) model, the detailed facility description, instrumentation locations, and boundary conditions applied in the tests would likely be required by most CFD analysts. Another specific concern is that the raw, or even processed experimental data, for most of these NEUPs are not available in a central location where access can be requested by the HTGR community. Since each university followed their own data storage procedures, access to the data vary widely from public websites maintained by the university or Principal Investigator (PI) to personal email requests for the data to the PI or student who processed the data and has since graduated.

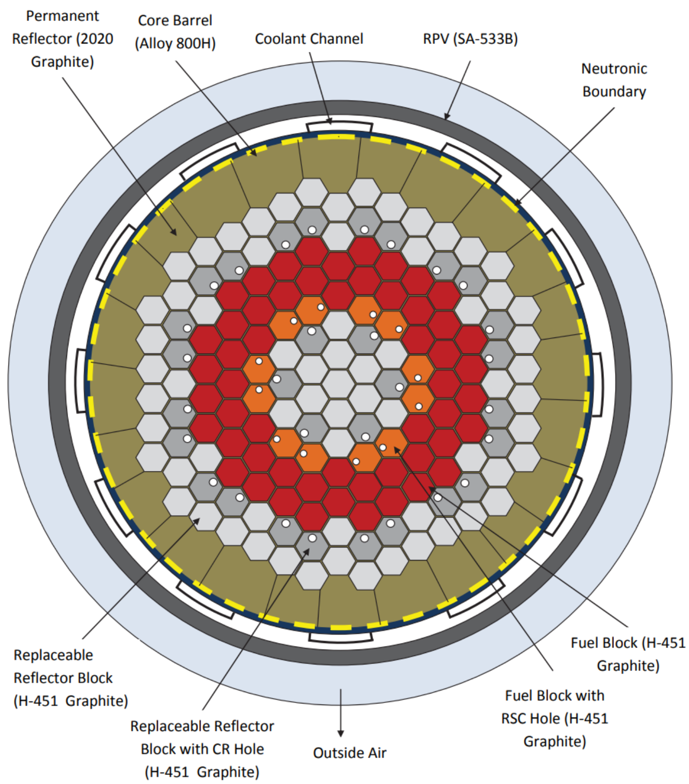
Overall, this is not a transparent, sustainable, or equitable process for the HTGR community to harvest the important HTGR thermal-fluid validation data created with U.S. taxpayer funding. Therefore, the Advanced Reactor Technologies (ART) Gas-Cooled Reactor (GCR) program will create a central database at Idaho National Laboratory (INL) to identify, organize, and store the results of these valuable research projects and provide future guidance for the storage and transmission of important NEUP project documentation. To improve access to this HTGR validation data and optimize the return on the significant investment made by DOE, the ART-GCR program has initiated an extensive literature survey of completed and ongoing HTGR NEUP projects to develop a public-access database that can retrieve CFD simulations or other system code validation data.

1.2 High-Temperature Gas-Cooled Reactor

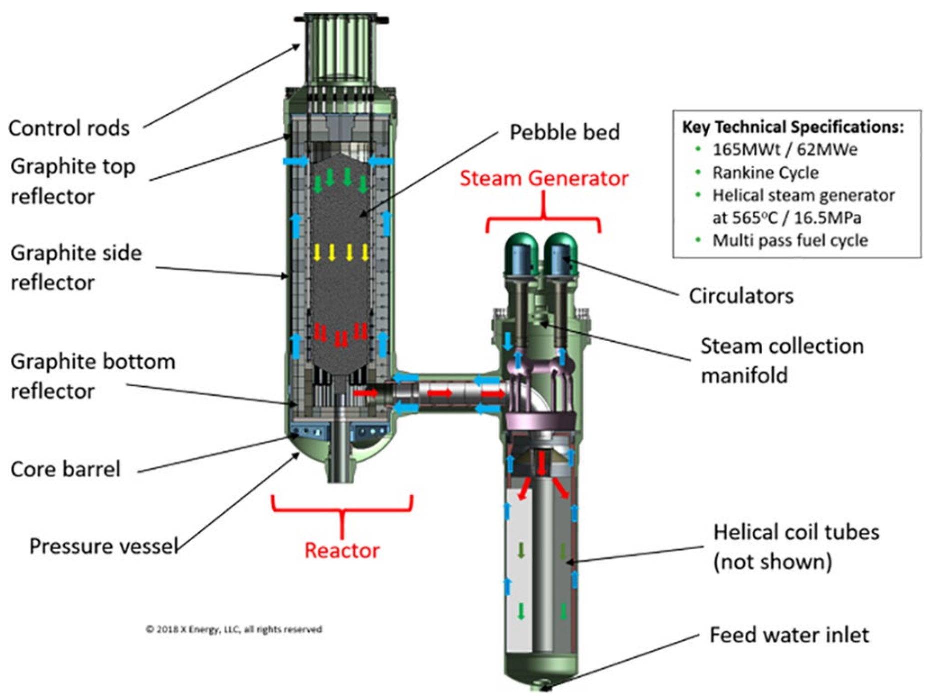
As one promising design among the Generation IV advanced reactors, HTGRs supply high-temperature heat energy by utilizing tristructural isotropic (TRISO) fuel [6], inert helium gas as a coolant, and graphite as a moderator. Currently, the two most promising HTGR candidate designs are the hexagonal prismatic block type and pebble-bed type. Figure 1(a) shows the top view of the Modular High-Temperature Gas-Cooled Reactor (MHTGR) of the General Atomics (GA) MHTGR-350 design [7] as an example of a prismatic HTGR type. The active part of the prismatic core consists of a large number of vertical stacks of a fuel block, existing prismatic GCR designs examples are Japan's High Temperature Engineering Test Reactor [8] and High Temperature Test Facility (HTTF) at Oregon State University (OSU) [9].

Meanwhile, the pebble-bed reactor concept is designed to be a graphite-moderated, helium-cooled reactor. As shown in Figure 1(b), the X-energy Xe-100 reactor is as an example of a current demonstration pebble-bed HTGR project in the U.S. [10]. The reactor core consists of spherical fuel elements called pebbles. Another example would be the 10-MW High Temperature Gas-cooled Reactor-Test Module built at the Institute of Nuclear and New Energy Technology of Tsinghua University [11]. It uses 6-cm-diameter pebbles, which are made of pyrolytic graphite [12]. Each fuel element contains approximate 12,000 0.92-mm-diameter TRISO-coated fuel particles. Also, the first unit of the world's first HTGR pebble-bed module demonstration power plant reached criticality in November 2021 at the Shidao Bay site in Shandong Province, China [13].

To develop, demonstrate, and deploy HTGRs, it is crucial to identify phenomena that may occur during normal operation and transient conditions. An extensive HTGR Phenomena Identification and Ranking Table (PIRT) assessment was led by the Oak Ridge National Laboratory (ORNL) and prepared for the U.S. Nuclear Regulatory Commission (NRC) in 2007 [14], detailing the important phenomena related to heat generation, distribution, and helium flows in the core and reflector regions during normal operation and off-normal accident scenarios for both HTGR reactor types, and this work was known as the NRC PIRT study. Some of the common and highest-ranked design-basis event scenarios for both prismatic and pebble-bed HTGR designs [15, 16] have been identified as pressurized and depressurized loss of forced cooling (P/DLOFC), air and steam ingress following a DLOFC, plenum mixing and interactions, reactor cavity cooling system (RCCS) performance, etc. After the release of the NRC PIRT assessment, the phenomena ranking determined the scope of the 29 NEUP projects at various U.S. universities. The focus of these experiments were on those phenomena with high importance but medium-to-low knowledge levels, such as core coolant bypass flow during normal operation, the RCCS performance during accidents, air ingress rates and flow reversal during DLOFC transients, and flow mixing in the upper and lower plena structures.



(a)



(b)

Figure 1. Two HTGR reactor core designs: (a) prismatic fuel block [7] and (b) pebble-bed [10].

1.3 Thermal-Hydraulics Studies in High-Temperature Gas Reactors

HTGR designs are continually evolving based on safety and other technical improvements but also due to consumer demands and changes within the regulatory environment. Considerable attention has been given to near-term deployment and minimizing technical risk. For these design concepts to develop HTGR technologies, it is critical to capture the results of these research efforts to prioritize remaining safety or licensing issues.

To support these ongoing efforts to improve HTGR designs, our literature survey in this work focuses on the HTGR's thermal-hydraulic behaviors during the normal operations and off-normal events. As stated in the NRC PIRT study on NGNP safety-relevant phenomena [15], the expert panel has identified and ranked phenomena critical to both steady-state and accident conditions. Together with these key points mentioned in the PIRT study, this section summarizes some of the most important and common thermal-fluid topics for the HTGR design and operation concerns.

1.3.1 Core Bypass Flow

As fuel configurations increase in complexity, understanding the core-wide heat transfer allows for the prediction of hot spot locations and the potential for sections of the core to exceed design temperature limits. In contrast to many reactor core designs for which subchannel or single fuel pin modeling is enough, high-fidelity graphite prismatic block temperature calculations require modeling of all the assembly fuel pins, coolant, and surrounding bypass flow. This is due to the heat transport within the block and between the fuel pins and coolant channels, known as the core bypass flow, as illustrated in Figure 2.

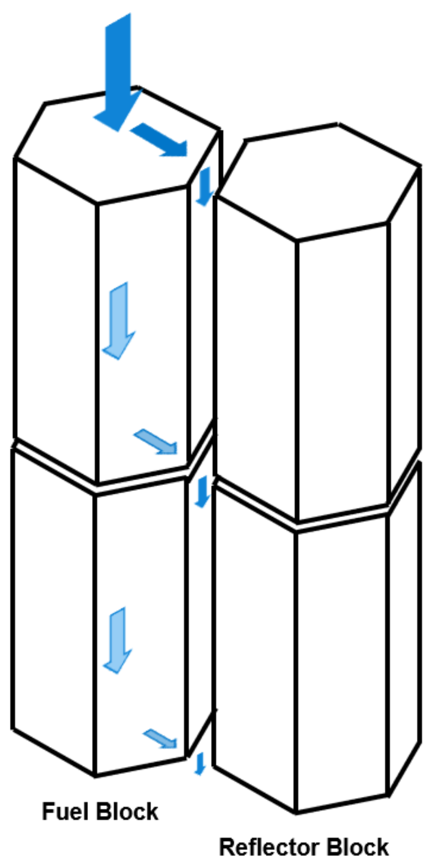


Figure 2. Illustration of core bypass flow.

Previous CFD investigations by Sato et al. [17] have indicated that bypass flow is a complex and multidimensional phenomenon inherently coupled to neutronics, thermal-mechanics, and fluid dynamics in the HTGR systems. Also, as found by MacDonald et al. [18], the maximum fuel and helium exit temperatures were occurring with larger bypass flow fractions. Their primary finding was that reducing the bypass flow to 10% decreased the maximum fuel temperature by 50°C and the maximum helium temperature by 75°C. Therefore, they suggested several design modifications that could be employed to reduce bypass flow. Some of the modifications include using high-temperature control rod materials so that less bypass flow would be needed to adequately cool the rods, using lateral core restraint mechanisms, and using a sealant between the core barrel and outer reflector gap to increase the flow resistance of the bypass pathway.

In the NRC PIRT study [15], the panel judged the core bypass flow of high importance with a low or medium knowledge level. Most of the panel recommendations are justified and should be considered for any future HTGR design efforts. Without fully addressing issues, such as bypass and cross flows, detailed thermal hydraulic and core heat transfer studies may have a high degree of uncertainty. The recommendation to develop an accurate cross and bypass flow network could pose difficulties without realistic experimental results to support the analysis. The validation highlights many issues with prediction and modeling of bypass flow. One of the modeling challenges is that, since graphite changes physical shape as a function of irradiation, bypass flow will vary axially along the core. It is clear that more experimental and computational investigations are needed to fully understand and characterize the core bypass flow and its resultant flow interactions in the core region.

1.3.2 Plenum Mixing

Analyzing coolant flow entry into and mixing in the lower plenum is important for identifying hot spots and investigating material or mechanical concerns due to excessive heating. The upper plenum is primarily important for accident situations without forced circulation. The mixing and formation of turbulent eddies inhibit natural circulation through the vessel. Some experiments have been performed, and more are underway, to test and observe these flow patterns.

1.3.2.1 Lower Plenum

Lower plenum hot streaking can occur when certain coolant channels and bypass gaps experience higher than average heat flux values due to power peaking [19]. Coolant temperatures near the core center and periphery will be less than that of coolant in the annular active core region. Once the coolant enters the lower plenum, it may impinge on the lower plenum support plate and columns, causing large thermal gradients. If the flow is not well mixed by the time it exits the core, it may impact the performance and safety of the power conversion unit. Lower plenum flow modeling is needed to address both potential issues. As illustrated in Figure 3, the CFD simulation at the lower plenum of the prismatic HTGR design performed by Clifford et al. [20] has demonstrated the multifaceted nature of lower plenum thermal-hydraulics with numerous fundamental processes (thermal stratification, turbulent mixing, vortex shedding, etc.) distinctly exhibited throughout the domain. Clifford et al. found that helium coolant jets enter the lower plenum at vastly different temperatures (due to the nonuniformly heated core and presence of bypass flow) and mix together in a complex turbulent fashion.

The PIRT study conducted by the NRC [15] has identified the lower plenum flow to be of high importance but low knowledge level. In the scaling studies and conceptual experimental design assessment work done by McEligot and McCreery [21], they have pointed out that any new experiments should include the graphite support columns, since no prior experiments have included the full range of possible flow interactions. They postulated that a simple flow visualization could be performed with a representative geometry tank, water, and dye injection. The temperature distribution in the lower plenum could be estimated by performing heated gas experiments with flow entering the representative geometry section and readings taken with a thermal imaging camera. The temperature distribution in the lower plenum could be estimated by performing heated gas experiments with flow entering the representative geometry section and readings taken with a thermal imaging camera. It is obvious that more sophisticated experimental facilities and advanced CFD modeling are needed to better understand this complex phenomenon.

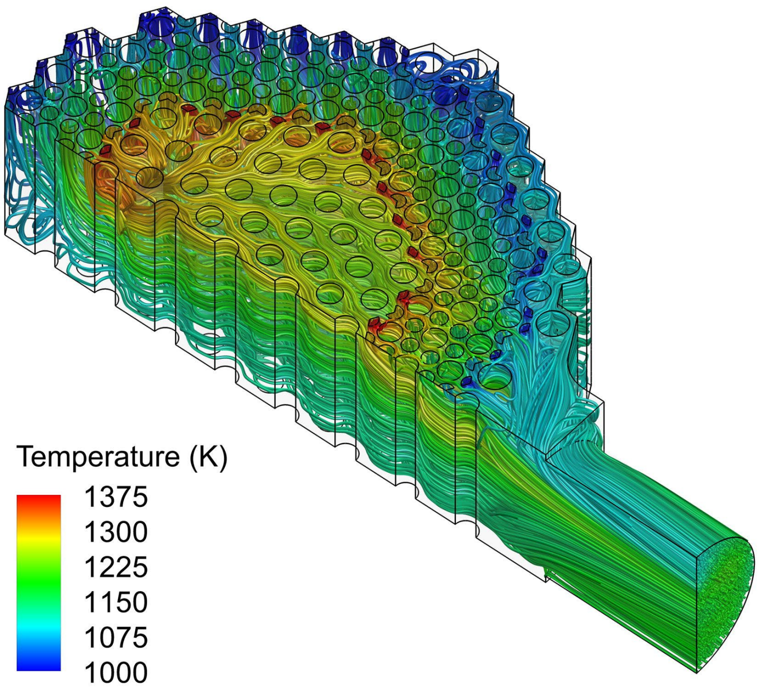


Figure 3. Coolant streamlines colored by temperature (K) in lower plenum of a prismatic HTGR design. Adapted from [20].

1.3.2.2 Upper Plenum

The fluid interactions at the upper plenum can be quite essential to HTGR safety design. As Huning et al. mentioned in their study [19], buoyancy forces result in hot gases flowing upward through the core and into the upper plenum above the core. Due the accident scenario, the natural circulation can act as the passive safety feature for HTGR systems. Once in the upper plenum, the amount of mixing and specific flow patterns that develop will determine the heat transfer to the vessel upper head and into the riser channels along the vessel wall. Finally, heat transfer away from the vessel wall is critical for energy balance and core temperatures to remain under their design limits. Therefore, to know whether natural circulation is established in the HTGR during a loss of forced circulation (LOFC) accident, the coolant flow through the core to upper plenum must be experimentally verified.

With the aid of CFD simulations, Che et al. [22] investigated the local flow field in the upper plenum during an extended accident induced by steam ingress. Figure 4 provides a CFD illustration for the buoyant jet mixing driven by the natural convection at the upper plenum of HTGR. As for the experimental work, Alwafi et al. [23] has performed a single isothermal jet discharging into the upper plenum in the upper plenum of a 1/16th scaled HTGR facility. The study discussed the flow characteristics of the isothermal jet mixing in the upper plenum obtained using time-resolved particle image velocimetry (PIV). Quantities such as mean velocity, root mean square fluctuating velocity, and Reynolds stress were computed using the PIV data.

Given the fundamental complexity of turbulent buoyancy jet mixing, and to better understand the jet flow mixing behaviors at the upper plenum and support HTGR safety analyses, it is crucial to gain more experimental data and computational benchmark studies with isothermal and non-isothermal flow mixing of a single jet and multiple jets in the upper plenum.

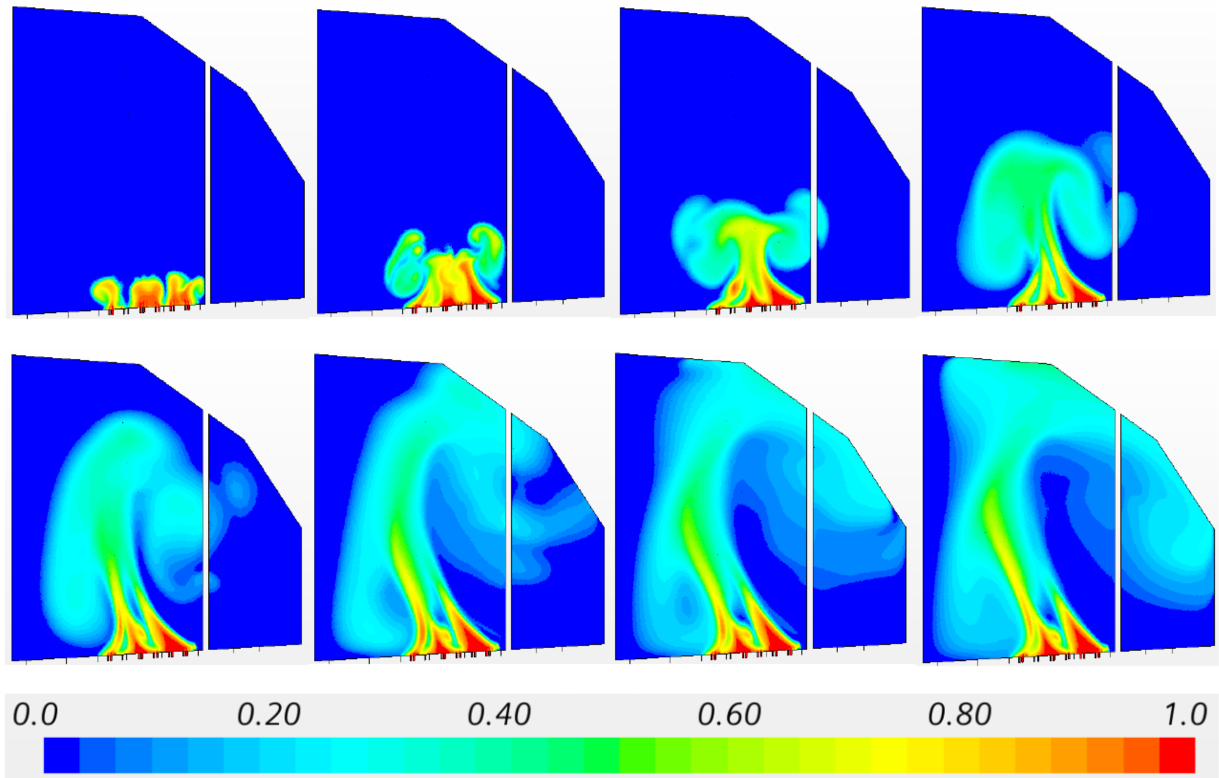


Figure 4. Time evolution of the hot gas mixture mass fraction at the HTGR's upper plenum. Adapted from [22].

1.3.3 Reactor Cavity Cooling System

As a passive safety system utilizing thermal radiation and natural convection, the RCCS operation is relatively simple and requires no moving parts or initiation for it to operate. While sacrificing some thermal efficiency, the simplicity of the system is expected to maintain economic competitiveness due to reduced capital, manpower, maintenance, and operation costs.

For HTGRs, there are mainly two RCCS designs as shown in Figure 5 which as currently under discussion: an air-cooled system initially proposed by GA [24] and an active, cold (<30°C), constant water flow system proposed by AREVA [25]. The Natural Convection Shutdown Heat Removal Test Facility (NSTF) at Argonne National Laboratory has conducted both air-based and water-based RCCS experiments since 2016 [26, 27] to support the DOE vision of aiding U.S. vendors in designing future reactor concepts, advancing the maturity of codes for licensing, and ultimately developing safe and reliable reactor technologies.

Although the RCCS is inherently safe and reliable, one major limitation is that, to ensure acceptable vessel and fuel temperatures during accident conditions, reactor power must be limited so that the shutdown (decay) heat levels are within the heat removal capability of the RCCS. Thus, the RCCS performance limits place significant limitations on the reactor design and requires scientific verification studies. However, given the limited computational and experimental data available, the RCCS performance during normal operations and accidents is of high importance but low-to-medium knowledge by the panel conducting the NRC PIRT study [15] and will need more investigations.

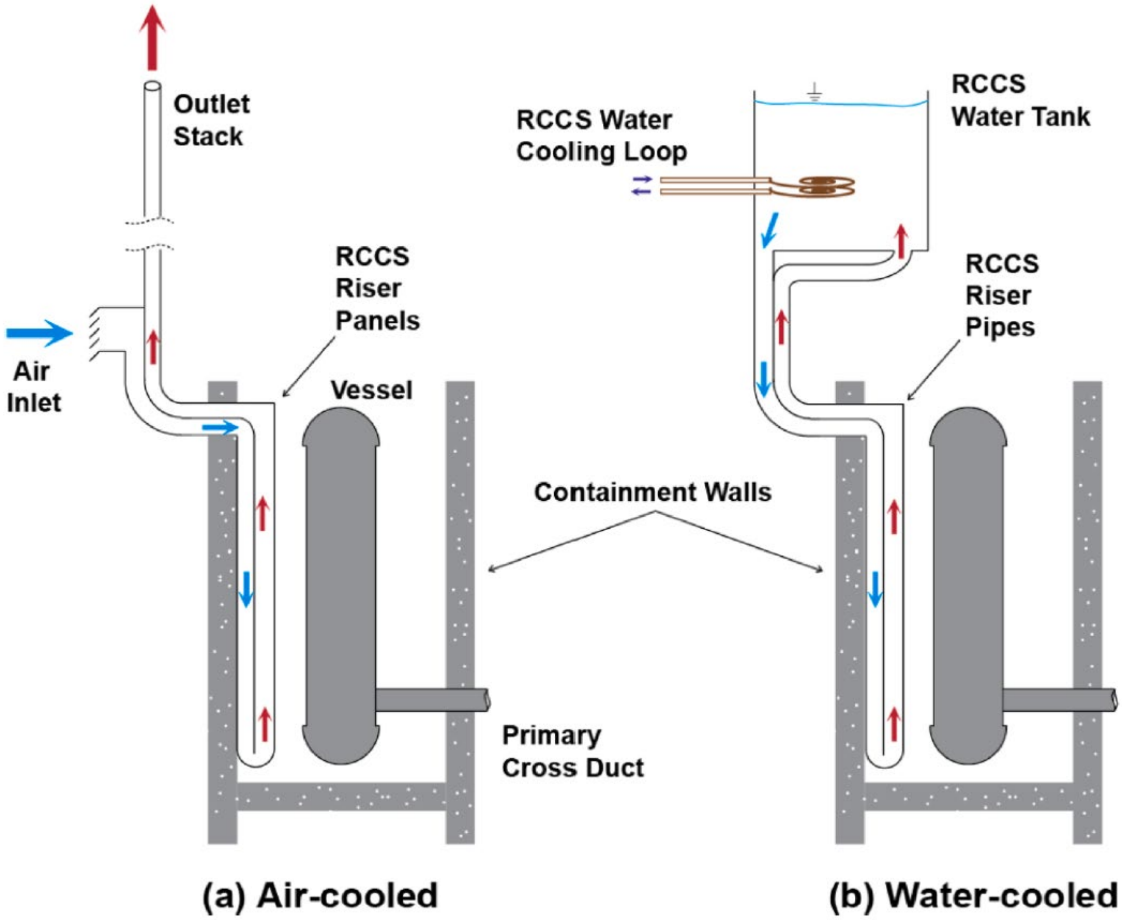


Figure 5. HTGR RCCS designs: (a) air cooled and (b) water cooled. Adapted from [19].

1.3.4 Potential Accident Scenarios

As pointed out by Huning et al. [19], gas reactors have many inherent safety features over conventional light-water reactor designs. These safety features include, but are not limited to, low power density, long transient time for operator actions, passive safety and reliance on conduction, radiation, and some potential natural circulation. Even with these enhanced passive safety aspects, the full range of possible accident scenarios must be analyzed.

Table 1 summarized some of the most common and crucial HTGR accident scenarios that described in the U.S. NRC PIRT discussion [15]. Accident scenarios of high interest include pressurized and depressurized loss of forced cooling (P/DLOFC), air ingress, and water ingress events. These will be used as the base for conducting our investigations for the completed and ongoing NEUP projects in Section 2.2. Given the fact that our work primarily focuses on the thermal-fluid phenomena for HTGRs, other plant-specific and reactivity events could occur but are not fully investigated in this report due to the large effort by the scientific community to fully address the phenomena associated with these principal events.

Table 1. HTGR transient and accident event sequence descriptions. Adapted from [19].

Possible Scenario	**Example Accident Initiator**	**Unique Event Sequence Characteristics**
PLOFC	Helium circulator trip or mechanical failure	• Circulator pump coast down followed by control rod SCRAM • Natural circulation at high pressure evenly distributes core temperatures, lowering peak temperatures from normal operational levels • Chimney effect causes upper head and upper vessel temperatures to be higher than lower elevation temperatures • RCCS functions normally and carries vessel heat to ultimate heat sink
DLOFC	Primary system break (or several smaller breaks)	• Circulation pump trip and coast down followed by control rod SCRAM • Break geometry and location inhibit air ingress • Heat conduction is primary heat removal mechanism since buoyancy driven flow is reduced at low pressure • Core temperatures fall at time of break but gradually increase and peak around 2 days • Peak vessel temperatures occur near center or "beltline" elevation • RCCS functions normally and carries vessel heat to ultimate heat sink
Air Ingress	A large primary system break (or several smaller breaks) followed by air ingress into the vessel	• Large set of possible scenarios depending on number of breaks, locations, and orientations • Circulation pump trip and coast down followed by control rod SCRAM • Break geometry and location form a convective air path between the vessel and the environment • Possible graphite and support structure oxidation resulting in loss of structural integrity • Possible fuel oxidation leading to fission product release
Water Ingress	Steam generator tube rupture or other water source break into the primary system	• Large set of possible breaks and total water volume ingress • Wide range of possible equilibrium system pressures and temperatures • Circulation pump trip and coast down followed by control rod SCRAM • Possible recriticality • Possible oxidation

2. OVERVIEW OF NUCLEAR ENERGY UNIVERSITY PROGRAM

To achieve the net-zero goal, the U.S. DOE has been promoting nuclear energy as a resource capable of meeting the nation's energy, environmental, and national security needs by resolving technical, cost, safety, security, and proliferation resistance through research, development, and demonstration. According to the research performed by Lawrence Livermore National Laboratory in 2021 [28], nuclear power accounted for 22.2% of all electricity generation. Given the market needs, it is apparent that nuclear energy is a key component of U.S. efforts to reduce dependence on fossil fuels and meet its commitments to drastically reduce greenhouse gas emissions, while continuing to ensure adequate domestic energy supplies.

2.1 Nuclear Energy University Program Background and Purpose

Given the importance of advancing nuclear power and its associated technologies, DOE's Office of Nuclear Energy created NEUP back in 2009 [29] to consolidate its university support under one program. As implied by its name, NEUP funds nuclear energy research and equipment upgrades at U.S. colleges and universities, providing student educational support. Since its founding, NEUP has played a key role in helping DOE accomplish its mission of leading the nation's investment in the development and exploration of advanced nuclear science and technology.

Between FY 2009 and FY 2021, 29 NEUP projects were funded to explore HTGR-related thermal-hydraulics research, experimentally or computationally. Figure 6 shows the locations of the universities that participated in NEUP-HTGR-related projects.

Figure 6. Universities participating in NEUP-HTGR-related projects.

2.2 Participants and Status of Nuclear Energy University Program Funded Projects

After reviewing all the projects' description on the NEUP and Office of Scientific and Technical Information webpages posted at www.neup.inl.gov, we have identified and created a list of the 29 NEUP-

HTGR projects that conducted thermal-fluid phenomena studies, as listed in Table 2. The list also contains the information of the PI and affiliated universities from FY 2009 to 2021. Figure 7 shows the same data organized by the 13 universities that received NEUP awards. The completed project identification numbers are in red boxes, while the ongoing ones are in blue (the first two digits in the NEUP numbering indicates the year it was awarded).

Besides organizing the general projects' information, another focus of conducting the NEUP survey was to map completed project scopes with HTGR thermal-hydraulics phenomena and operational conditions and to determine the current status of NEUP-funded HTGR experimental progress. Based on all the project descriptions we have gathered, Table 3 shows a comprehensive summary mapping of the HTGR thermal-hydraulics phenomena covered during various operating scenarios in the finished (denoted in red boxes) and ongoing (denoted in blue boxes) NEUP projects. The information shown in Table 3 not only allows presenting the NEUP project scopes with the crucial HTGR thermal-hydraulics phenomena and its operational conditions but also helps to determine what gaps remain in HTGR research study and code validation space, which can inform future HTGR-related NEUP scope calls.

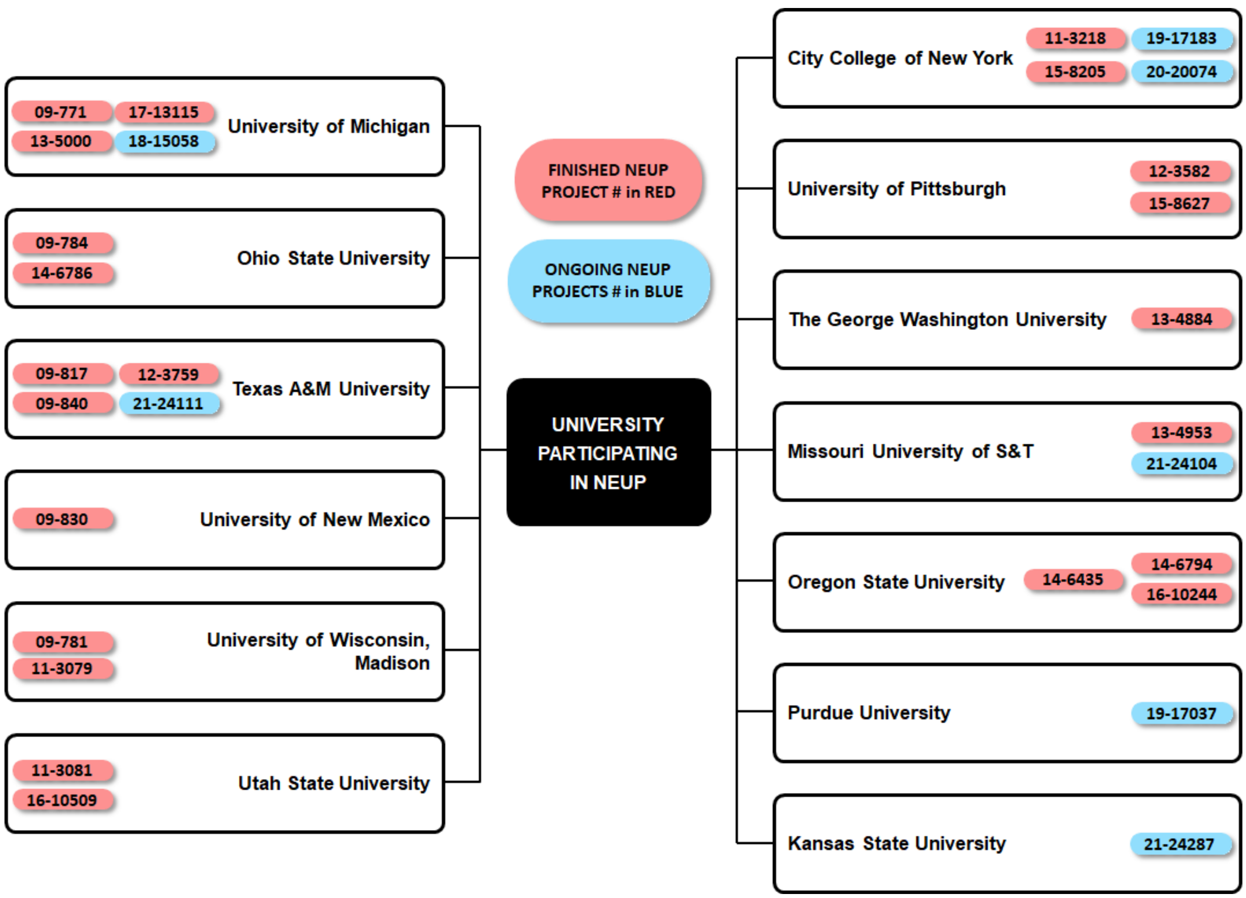


Figure 7. The list of universities (that PI is affiliated with) that have participated with the NEUP, together with their finished (in red) and ongoing (in blue) NEUP project identification numbers.

Table 2. Complete list of NEUP projects that investigate HTGR thermal-hydraulics phenomena.

Project No.	NEUP Project Title	PI
09-771	Creation of a Full-core HTR Benchmark with the Fort St. Vrain Initial Core and Assessment of Uncertainties in the FSV Fuel Composition and Geometry	William Martin (University of Michigan, Ann Arbor)
09-781	Experimental Studies of NGNP Reactor Cavity Cooling System with Water	Michael Corradini (University of Wisconsin-Madison)
09-784	Investigation of Countercurrent Helium-Air Flows in Air-ingress Accidents for VHTRs	Xiaodong Sun (Ohio State University)
09-817	CFD Model Development and Validation for High Temperature Gas Cooled Reactor Cavity Cooling System (RCCS) Applications	Yassin Hassan (Texas A&M University)
09-830	Graphite Oxidation Simulation in HTR Accident Conditions	Mohamed El-Genk (University of New Mexico)
09-840	Investigation on the Core Bypass Flow in a Very High Temperature Reactor	Yassin Hassan (Texas A&M University)
11-3079	Thermal-hydraulic Analysis of an Experimental Reactor Cavity Cooling System with Air	Michael Corradini (University of Wisconsin-Madison)
11-3081	Transient Mixed Convection Validation for NGNP	Barton Smith (Utah State University)
11-3218	Experimental Investigation of Convection and Heat Transfer in the Reactor Core for a VHTR	Masahiro Kawaji (City College of New York)
12-3582	Experimentally Validated Numerical Models of Non-isothermal Turbulent Mixing in High Temperature Reactors	Mark Kimber (University of Pittsburgh)
12-3759	Experimental and CFD Studies of Coolant Flow Mixing within Scaled Models of the Upper and Lower Plenum of a NGNP Gas-Cooled Reactors	Yassin Hassan (Texas A&M University)
13-4884	Validation Data for Depressurized and Pressurized Conduction Cooldown	Philippe M Bardet (The George Washington University)
13-4953	Experimental and Computational Investigations of Plenum-to-Plenum Heat Transfer and Gas Dynamics under Natural Circulation in a Prismatic Very High Temperature Reactor	Muthanna Al-Dahhan (Missouri University of Science &Technology)
13-5000	Model Validation Using CFD-grade Experimental Database for NGNP Reactor Cavity Cooling Systems with Water and Air	Annalisa Manera (University of Michigan, Ann Arbor)
14-6435	Fluid Stratification Separate Effects Analysis, Testing and Benchmarking	Andrew C. Klein (Oregon State University)

14-6786	Experimental Investigation and CFD Analysis of Steam Ingress Accidents in HTGRs	Xiaodong Sun → Richard Christensen (Ohio State University)
14-6794	Scaling Studies for Advanced High Temperature Reactor Concepts	Brian Woods (Oregon State University)
15-8205	Experimental Investigation of Forced Convection and Natural Circulation Cooling of a VHTR Core under Normal Operation and Accident Scenarios	Masahiro Kawaji (City College of New York)
15-8627	Experimental Validation Data and Computational Models for Turbulent Mixing of Bypass and Coolant Jet Flows in Gas-Cooled Reactors	Mark Kimber (University of Pittsburgh)
16-10244	Integral System Testing for Prismatic Block Core Design HTGR	Brian Woods (Oregon State University)
16-10509	CFD and System Code Benchmark Data for Plenum-to-Plenum Flow Under Natural, Mixed and Forced Circulation Conditions	Barton Smith (Utah State University)
17-13115	Experimental Determination of Helium Air Mixing in Helium Cooled Reactor	Victor Petrov (University of Michigan, Ann Arbor)
18-15058	High-resolution experiments for extended LOFC and Steam Ingress Accidents in HTGRs	Xiaodong Sun (University of Michigan, Ann Arbor)
19-17037	Investigation of HTGR Reactor Building Response to a Break in Primary Coolant Boundary	Shripad T. Revankar (Purdue University)
19-17183	Mixing of Helium with Air in Reactor Cavities Following a Pipe Break in HTGRs	Masahiro Kawaji (City College of New York)
20-20074	Characterization of Plenum-to-Plenum Natural Circulation flows in a High Temperature Gas Reactor (HTGR)	Masahiro Kawaji (City College of New York)
21-24104	Thermal Hydraulics Investigation of Horizontally Oriented Layout micro HTGRs Under Normal Operation and PCC Conditions Using Integrated Advanced Measurement Techniques	Muthanna Al-Dahhan (Missouri University of Science &Technology)
21-24111	Experimental Investigations of HTGR Fission Product Transport in Separate-effect Test Facilities Under Prototypical Conditions for Depressurization and Water-ingress Accidents	N.K. Anand, Thien Nguyen, Yassin Hassan (Texas A&M University)
21-24287	Investigating Heat Transfer in Horizontally Oriented HTGR under normal and PCC conditions	Hitesh Bindra (Kansas State University)

Table 3. Thermal-fluids phenomena summary for all the HTGR-related NEUP projects, with finished projects numbers denoted in a red box while ongoing projects are in blue.

FINISHED NEUP PROJECT # in RED

ONGOING NEUP PROJECTS # in BLUE

PHENOMENA		SCENARIOS: NORMAL OPERATION	PRESSURIZED LOSS OF FLOW	DEPRESSURIZED LOSS OF FLOW	LOAD CHANGE (TRANSIENT)	STEAM GENERATOR TUBE BREAK
PLENUM MIXING / JET IMPINGEMENT	LOWER PLENUM	12-3582, 15-8627	16-10244	16-10244		
	UPPER PLENUM	12-3759	18-15058	18-15058		
	PLENUM TO PLENUM	13-4953, 16-10509	20-20074	20-20074		
INGRESS	AIR INGRESS		15-8205	09-784, 13-4884, 14-6435, 15-8205, 17-13115		14-6786, 19-17183
	STEAM/WATER INGRESS		18-15058	18-15058, 21-24111		14-6786
CONJUGATE HEAT TRANSFER (CORE)	FORCED CONVECTION	09-771, 11-3081, 16-10509, 21-24104, 21-24287	11-3218, 21-24104, 21-24287	11-3218		
	NATURAL CONVECTION	14-6794, 16-10509	14-6794	14-6435		
RCCS PERFORMANCE		09-781, 09-817, 11-3079, 13-4953			09-781, 09-817, 11-3079, 13-5000	
CORE BYPASS FLOW		09-830, 15-8627	15-8205	09-840, 15-8205		
FLUID STRATIFICATION				14-6435		
NUCLEAR (DECAY HEAT, REACTIVITY FEEDBACK, FISSION PRODUCT, ETC.)				19-17037, 21-24111		

3. NUCLEAR ENERGY UNIVERSITY PROGRAM EXPERIMENTAL FACILITIES FOR VERIFICATION AND VALIDATION STUDY

Code verification and validation (V&V) is a very important step in the process of designing any complicated systems including advanced nuclear reactors. Experimental facilities can be modeled by CFD simulations or system codes using the well-defined initial and boundary conditions, and experimental data points can be utilized and compared to the simulations to validate the model accuracy. It is therefore important and essential to have the experimental setup as close to real application setup as possible, together with the testing scenarios and boundary conditions clearly defined as experimentalists can. Table 4 summarizes the thermal-fluid phenomena and operating scenarios that the NEUP-funded HTGR projects have investigated.

Table 4. Summary of the investigated phenomena and scenarios for the completed NEUP.

Project No.	Phenomena & Scenarios	Research Outcome			
		Experiment	Data Type	Simulation	Simulation Tool
09-771	Nuclear Fuel Performance	Benchmark study	Power plant data from Fort Sr. Varin Initial Core	o	MCNP5 and RELAP5-3D
09-781	Reactor Cavity Cooling System with water	o	Temperature and mass flow rate (time resolved)	o	FLUENT
09-784	Air-Ingress	o	Flow rate, density	-	-
09-817	CFD for RCCS in VTHR	o	Temperature (k-type TC) and mass flow rate (flow meter)	o	RELAP5-3D
09-830	Graphite Oxidation Simulation during air and water ingress	-	-	o	STAR-CCM+
09-840	Core Bypass Flow	o	Velocity component, velocity field	o	STAR-CCM+
11-3079	RCCS with Air for Modular HTGR	o	Heating power, temperature (k-type TC), air velocity (probe), and humidity (weather)	o	FLUENT
11-3081	PIV vs CFD data validation for a heated flat plate	o	Heat flux, PIV velocity (stress tensor, TKE, y+...)	o	STAR-CCM+
11-3218	Abnormal heat transfer scenarios, such as flow laminarization.	o	Temperature, Nusselt number	o	COMSOL
12-3582	Turbulent jet mixing and CFD study	o	PIV velocity field (TKE), temperature (TC, IR camera for field)	o	OpenFOAM
12-3759	Flow Mixing	o	Velocity field (PIV)	o	STAR-CCM+
13-4884	MTV for gas flow	o	Velocity field	o	FLUENT
13-4953	NC Heat Transfer	o	Temperature	o	COMSOL, STAR-CCM+
13-5000	RCCS and turbulent jet mixing	o	Velocity field (stress tensor and TKE)	o	STAR-CCM+
14-6435	Fluid Stratification	o	Flow rate, pressure, temperature	-	-
14-6786	Steam ingress accident in HTGR	o	Temperature (optical fiber), concentration, pressure	o	COMSOL
15-8205	Natural and forced circulation	o	Velocity (hot-wire measurement), temperature (IR camera, and k-type TC)	o	COMSOL, NEK5000
15-8627	Turbulent jet mixing and bypass flow	o	Velocity Field (PIV), Reynolds Stress, Turbulent kinetic energy, vorticity	o	STAR-CCM+
16-10244	HTTF thermal-fluid testing	o	Flow rate, temperature, pressure	o	STAR-CCM+
16-10509	Buoyancy driven/opposed flow measurements	o	Velocity field (stress tensor and TKE), temperature, pressure, mass flow rate	o	FLUENT
17-13115	He/Air mixing behavior and air ingress during accidents	o	Jet velocity data (PIV and LDV), mass flow rate, temperature, pressure	o	STAR-CCM+

Table 4 also lists whether the NEUP project has conducted experimental investigations, performed computational analyses, or both; it also briefly reviews the experimental data types collected and the simulation tools used by the project. Many of these NEUP projects have produced excellent experimental results, but these NEUP projects were inevitably limited by the time and budget to perform solid V&V studies. Therefore, more efforts are still needed to organize the detailed information of the experimental campaign and utilize these experimental data to conduct code validation and benchmark studies.

3.1 Nuclear Energy University Program Experimental Facilities

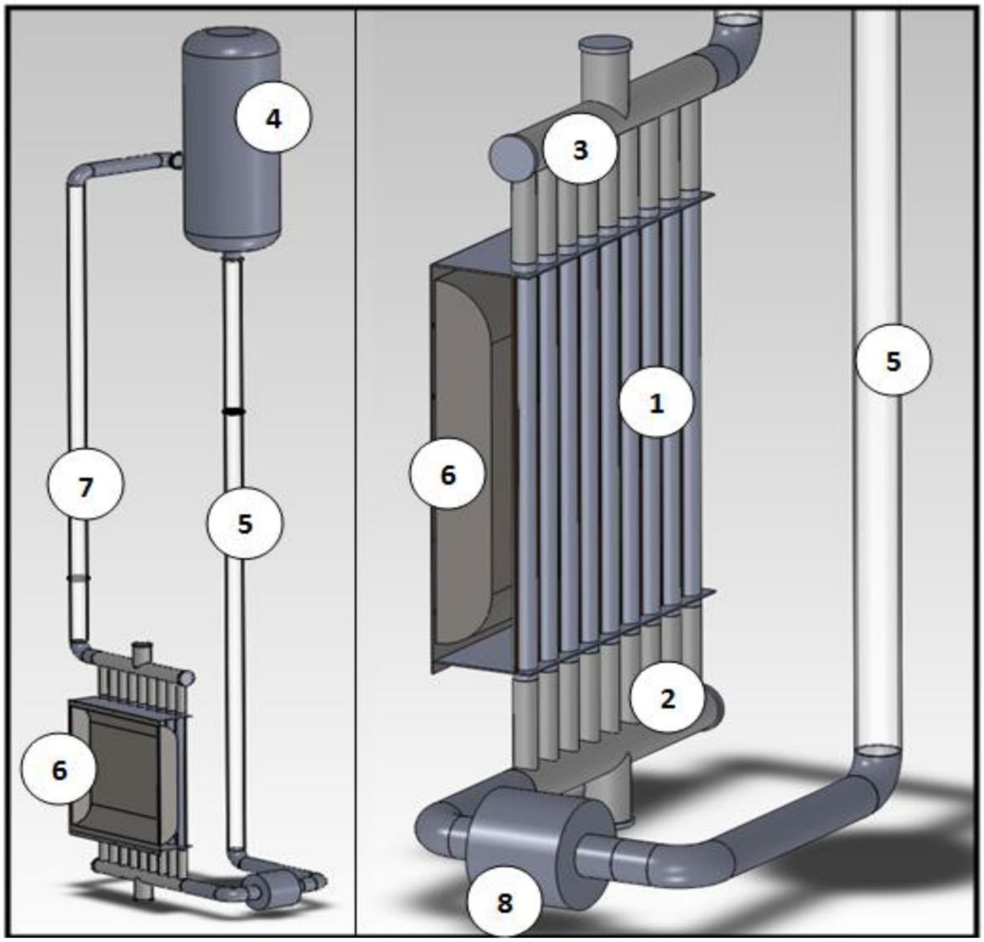
As an example of the typical data produced by the completed HTGR-related NEUP projects shown in Table 4, a subset of eight experiments covering the main areas of interest identified in Section 1.3 were reviewed in chronological order. The selected experimental facilities are summarized here in Table 5 with the awarded universities. The experiment features and applied measurement techniques have been examined and summarized, focusing on the HTGR thermal-fluid phenomena and scenarios during normal and off-normal operations.

Table 5. Selected experimental facilities for HTGR thermal-fluid phenomena investigations funded by DOE NEUP.

Project No.	University (PI affiliated)	Brief Experimental Facility Description
09-817	Texas A&M University	Natural circulation heat transfer in the water-based RCCS system for HTGR
09-840	Texas A&M University	CFD analysis of core bypass flow phenomenon
11-3081	Utah State University	Integral-level validation experiment of heat transfer phenomenon stemming from a vertical plate
11-3218	City College of New York	Both forced convection and natural circulation experiments to investigate abnormal heat transfer scenarios, such as flow laminarization
13-4953	Missouri University of Science and Technology	Heat transfer analysis for differing plenum-to-plenum natural circulation experiments
13-5000	University of Michigan, Ann Arbor	Jet flow mixing in upper plenum of RCCS for CFD code validation
15-8205	City College of New York	Bypass flow phenomenon under natural and forced circulation in plenum-to-plenum flow and natural circulation after air ingress into lower plenum
16-10244	Oregon State University	Different integral tests using the HTTF for scenarios, such as PLOFC

3.1.1 NEUP No. 09-817: TAMU RCCS Experimental Facility [30]

This experiment built at Texas A&M University (TAMU) is focused on validating the RCCS for HTGR and studying the thermal-hydraulic behavior of the water under normal operation and accident scenarios. This facility is a 1/23 axially scaled experimental test loop, with a partial pressure vessel and a large water tank storage loop. The working fluid is water in this RCCS experimental setup. A schematic of the facility is shown in Figure 8. The pressure vessel is electrically heated to simulate a working HTGR, though the vessel was simplified to a flat surface in contact with the RCCS cooling plate. The water tank is elevated to ease downcomer coolant into the cavity and aid in natural circulation. Flow visualization is performed by injecting die and visualizing the flow pattern indicated by the die. The facility is equipped with instrumentation sufficient for validation purposes with flow rates and temperature measurements. Figure 9 shows the different instrumentation and their locations in the test loop. To ensure a steady state was reached, a secondary cooling mechanism shown in Figure 10 was employed to cool down the water in the primary coolant system. Some water from the main coolant channel was sent through a coil that was cooled using ice water. This coil running through the colder water ensured that the water reachs a steady-state operating condition.



Component #	Component Description	Material	Accessibility
1	Risers' Panel	SS304	Elevation 0
2	Bottom Manifold	Glass	Elevation 0
3	Top Manifold	Glass	Elevation 0-1
4	Water Tank	Steel	Elevation 2
5	Downcomer	Polycarbonate	Elevation 0-1
5	Reactor Vessel	SS304	Elevation 0
5	Upward Pipeline	Polycarbonate	Elevation 1-2
5	Flowmeter	-	Elevation 0

Figure 8. Experiment facility overview for the RCCS facility at TAMU. The table shows the detailed components numbered in the facility figure. Adapted from [30].

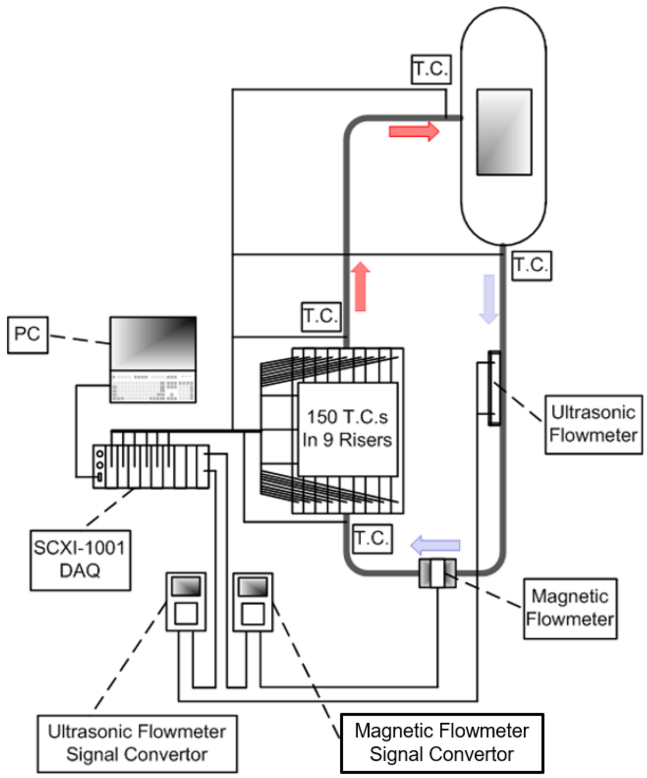


Figure 9. Instrumentation layout for the RCCS facility at TAMU. Adapted from [30].

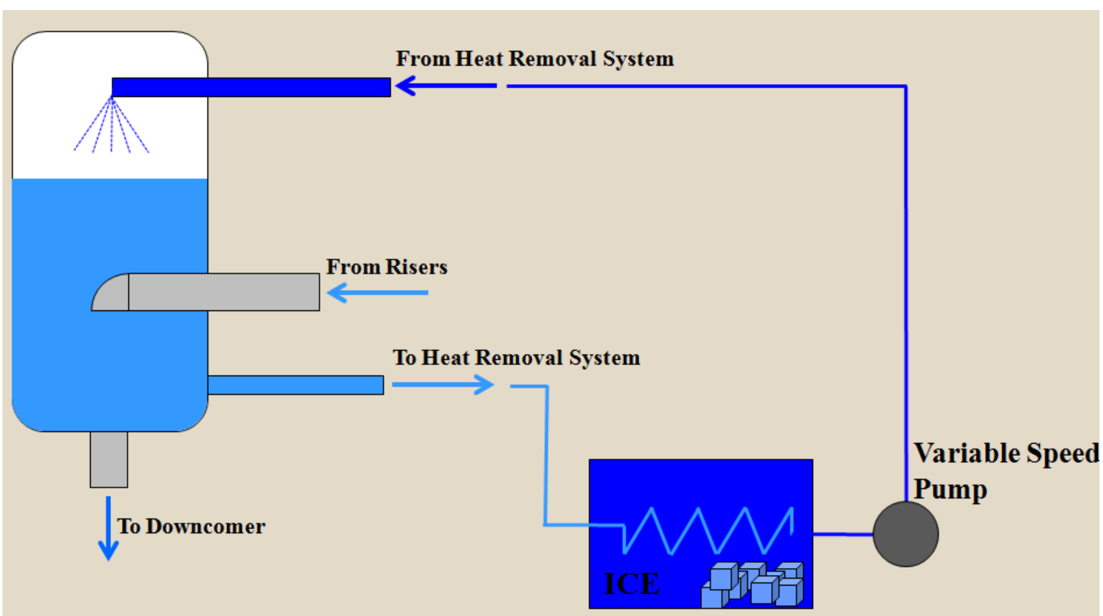


Figure 10. Secondary heat removal system schematic. Adapted from [30].

The experiments performed in this NEUP were to measure the onset of natural circulation in the RCCS system and measure its heat removal capacity during accident scenarios. In order to prove the sufficient heat removal rate of RCCS design, the facility was aimed to achieve a steady-state scenario and quantify heat removal and flow rates. The experimental procedure was started by filling the facility with tap water at room temperature and subsequently started acquiring data about 30 minutes before the experiment began. Then the heater was triggered, allowing the main coolant to begin natural circulation. Once a target temperature for the tank outlet temperature was reached, ice was added to the secondary coolant channel to ensure a steady-state scenario by cooling the main channel coolant. The ice was regularly added into the secondary cooling channel in batches of 5 lbs to keep the secondary coolant channel at a stable temperature.

The RCCS behavior was modeled with the RELAP5-3D code as part of the validation study. After addressing the limitations for the heat losses from the back of the heaters and the cavity thermal inertia for the heater and vessel, the refined RELAP5-3D (new model) matched well with the experimental data compared to their original RELAP5-3D model. As shown in Figure 11, the experiment results showed the onset of natural coolant occurring around 1,700 seconds after the start of the test. This onset was accompanied by a sharp increase in cavity inlet coolant temperature, as well as the coolant flow rate. After the onset of natural circulation was initiated, the coolant inlet and outlet temperature continued to increase by 0.14°C/min with a constant difference of about 2°C between the inlet and outlet temperatures. When the coolant in the tank reached 30°C, ice was added into the secondary cooling tank. This made the facility reach the steady-state operation. This state was maintained for 1,000 seconds until the facility was shut down. The final coolant temperature was 31°C before the experiment shutdown. This showed that the RCCS performance via natural circulation is sufficient for heat loss in accident scenarios.

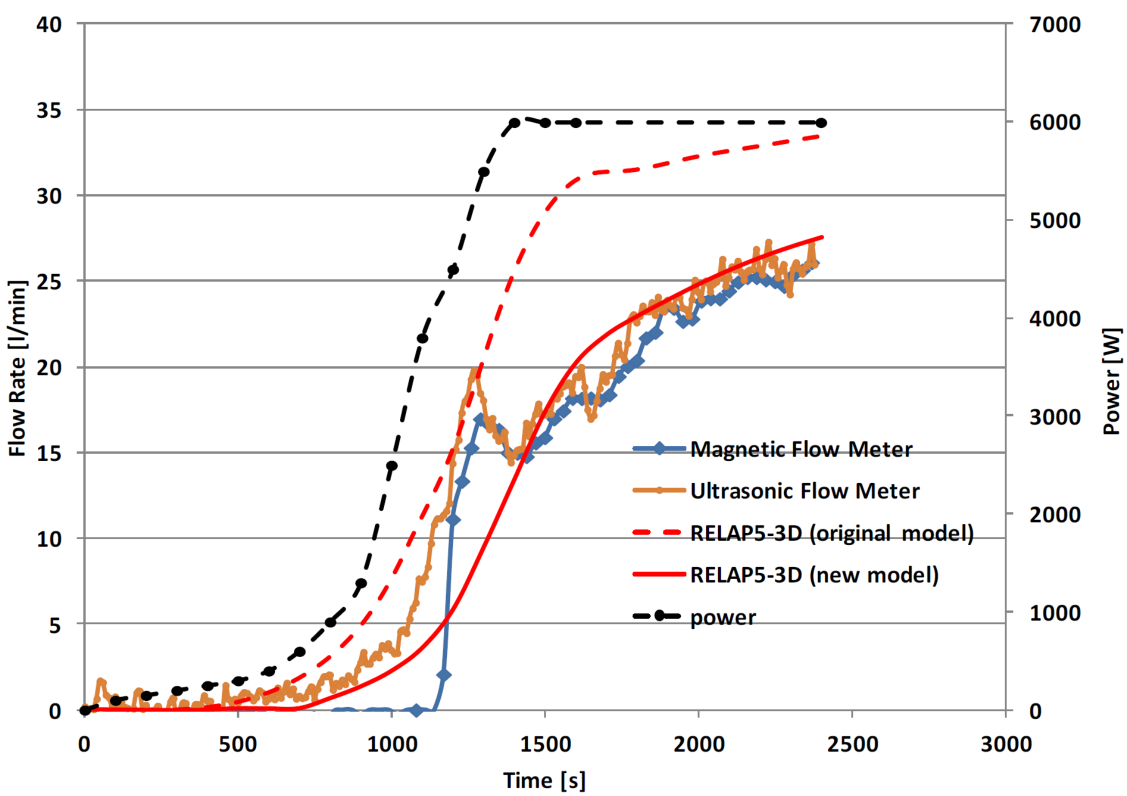


Figure 11. Facility shakedown experimental results compared to the RELAP5-3D simulations. Adapted from [30].

The experimental data resulting from this NEUP project is not yet publicly available. To perform code benchmark studies, researchers will need to digitize the final report data plots or contact the PI (Prof. Yassin Hassan) for more information.

3.1.2 NEUP No. 09-840: TAMU Bypass Flow Experimental Facility [31]

Based on the GA prismatic MHTGR design, the research group at TAMU constructed the flow visualization facility shown in Figure 12. The facility consists of two separate 500-mm-long blocks stacked upon one another and split into three sections with six 25.4-mm-diameter cooling channels each. The scaling is such that it matches the Reynolds number of 30,000 for the GA reactor design. The investigators use this facility to measure the bypass flow and crossflow that would happen in a prismatic reactor to ensure that the bypass flow was not large enough to significantly affect core cooling. The data was acquired using thermocouples along the test tube, and the nonintrusive laser doppler velocimetry technique was used to capture the velocity information inside the bypass gaps. The gap divider shown in the right side of Figure 12 is used to change the width of the bypass gap. The detailed test loop setup with the instruments and data acquisition systems were illustrated in Figure 13.

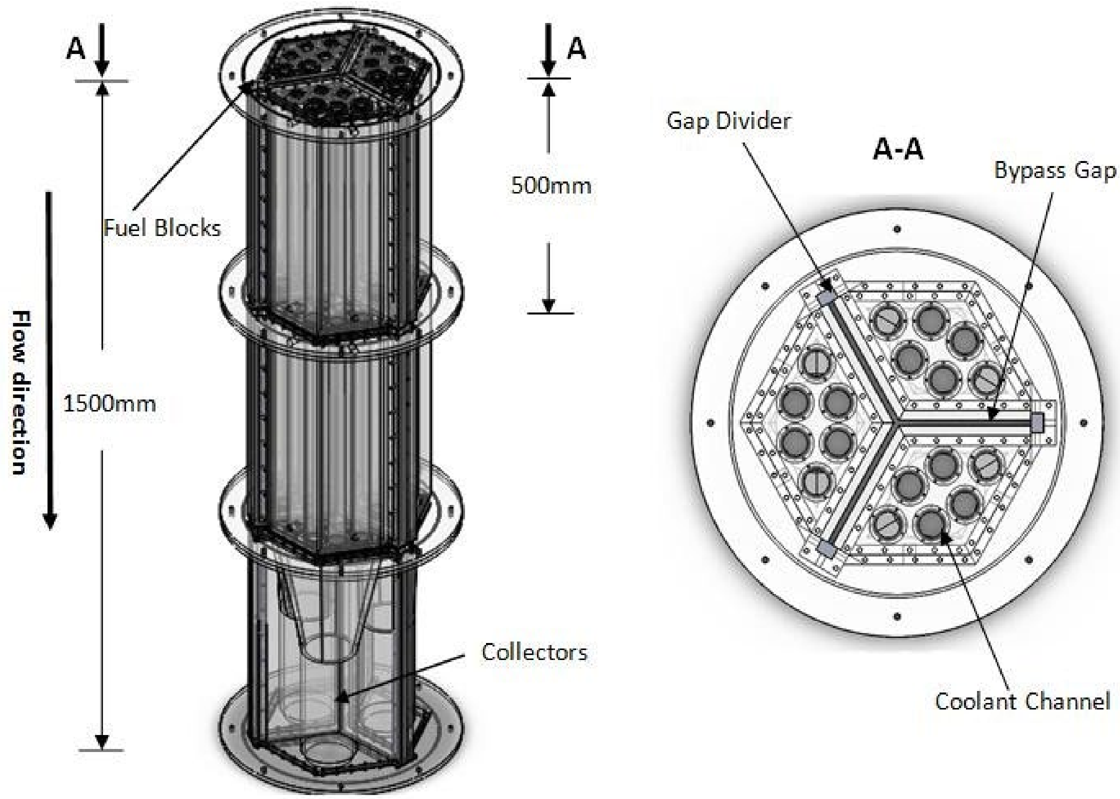


Figure 12. Diagram of the core bypass facility test section geometry. Adapted from [31].

Besides the experimental work, this project also investigated if CFD calculations could predict bypass flow fractions in a prismatic HTGR for different bypass gap widths and crossflow gap geometries. As shown in Figure 14, the geometry was treated with different cell counts and wall functions to determine the best fit. Both the near wall and wall function approaches were taken, along with different mesh geometries with a tweaked κ-ε approach. Most simulations assumed a uniform crossflow gap of 1 mm, though one simulation looked at a wedge-shaped crossflow gap between 1 and 2 mm.

The results of the CFD analyses as well as the experimental data have indicated that decreasing the bypass gap size will decrease bypass flow fraction, that is decreasing the crossflow gaps are confirmed to reduce the flow rate in the bypass gaps and increase flow to the coolant channels. This was expected, but for 6-mm gaps, the bypass flow fraction accounted for up to 12% of the total flow, whereas the smaller 2-mm gap only accounted for 1.7% of the flow. The Reynolds number was also directly proportional to the bypass flow. The presence of the crossflow gap and its shape had a significant effect on flow redistribution when the coolant passes through the crossflow gap. This study found a significant secondary flow moving from the bypass flow gap towards coolant holes, which resulted in up to a 28% reduction of the coolant mass flow rate in the bypass flow gap while the flow in the coolant channels increased by around 5%.

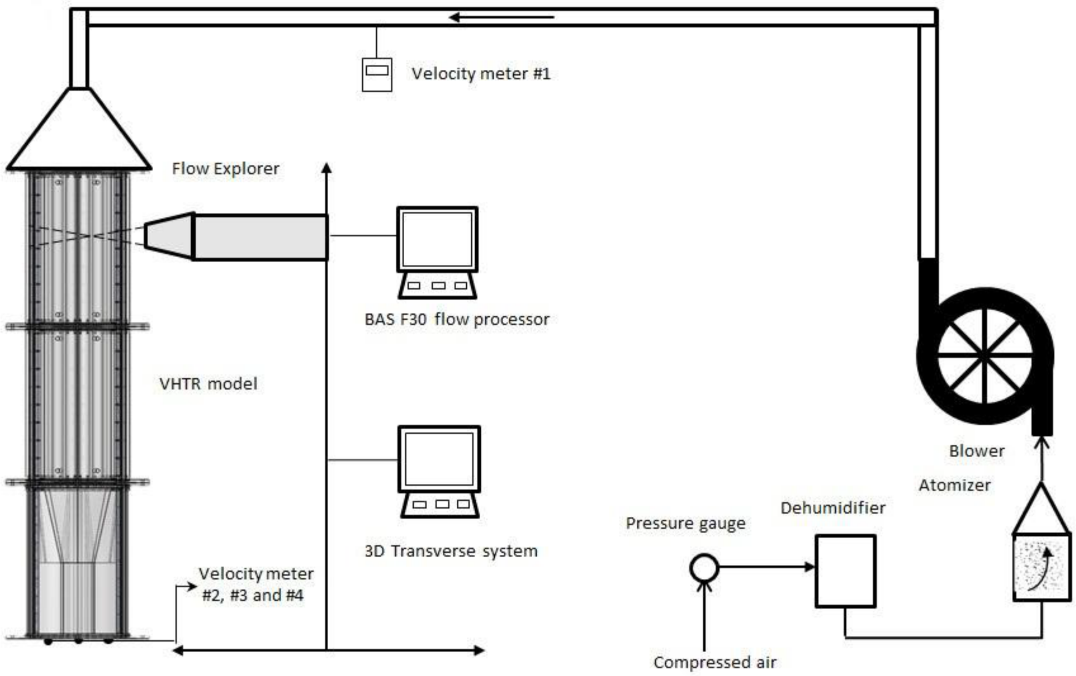


Figure 13. The experimental loop for the bypass flow investigations with the instrumentation locations. Adapted from [31].

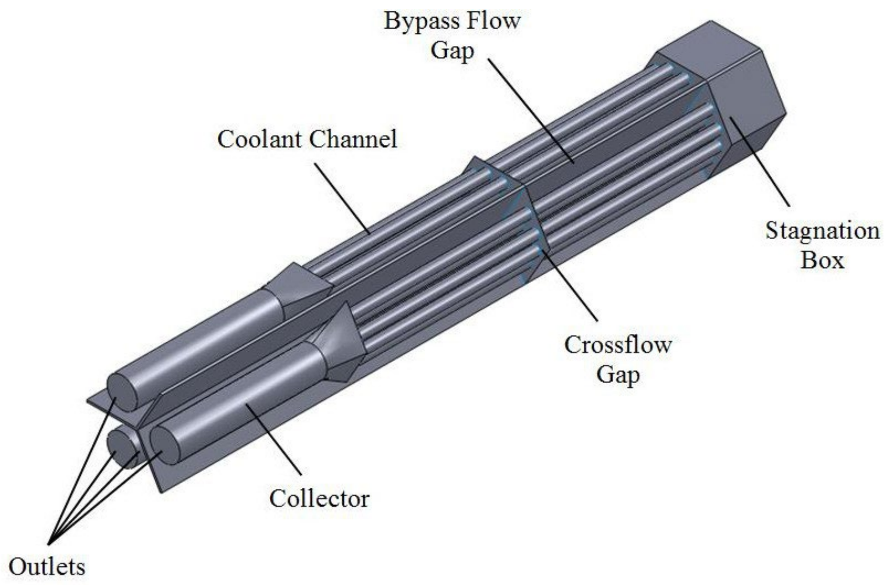


Figure 14. The geometry modeled for CFD calculations. Adapted from [31].

The experimental data resulting from this NEUP project is not yet publicly available. To perform code benchmark studies, researchers will need to digitize the data plots from the final report or contact the PI (Prof. Yassin Hassan) for more information.

3.1.3 NEUP No. 11-3081: Rotatable Buoyancy Tunnel Facility [32]

The NEUP project led by Prof. Barton Smith at Utah State University (USU) was focused on establishing very-high-quality validation data for heat convection stemming from a vertical plate with multiple system response quantity measurements. The experimental facility built for this NEUP project is the Rotatable Buoyancy Tunnel (RoBuT) Facility. This facility has a unique feature that it is located on a Ferris-wheel-like structure to allow for a full 180-degree rotation to study buoyancy opposed and aided convection. As shown in Figure 15, the geometry is straightforward, consisting of four vertical walls that are assumed to be flat for validation purposes.

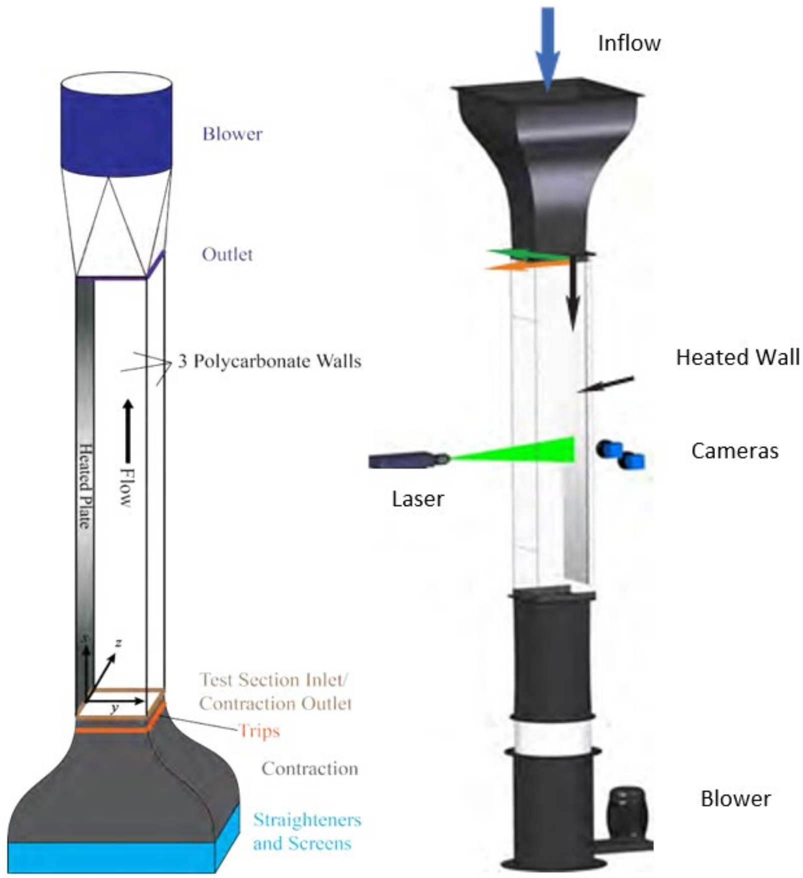


Figure 15. Both up (Left) and down (Right) configurations for the RoBuT test tunnel. Adapted from [32].

One of the walls shown in Figure 15 is heated, assumably through isothermal heating. The origin of the measurements as indicated in Figure 15 (left) is at the center of the leading edge of the heated plate. The x-coordinate runs the length of the plate, y is the coordinate away from the plate, and z is the coordinate normal to x and y. The test section is made of three optically clear, 0.5-in.-thick Lexan® polycarbonate walls. The top wall is split into three sections, with the two ends being fixed into the mounting brackets and the center section removable for cleaning purposes (olive oil tends to build up in the test section over time, making optical measurements less reliable). The bottom of the facility leading into the test section contains honeycomb-like flow straighteners to ensure laminar flow for the inlet velocity profile. RoBuT instrumentation measures conditions, such as temperature, humidity, and pressure, using an Omega HX92A and a Barometric Pressure Sensor SB100 from Apogee Instruments. The Omega sensor is accurate to 0.6°C and 2.5% relative humidity, while the SB100 is accurate to 1.5% of atmospheric conditions. The three nonheated wall temperatures are noncontrolled conditions measured with type K thermocouples to a 2.2-degree uncertainty. The heated plate is controlled using three heaters to help ensure isothermal heating. All the measurement and instrumentation uncertainties were clearly mentioned in the project final report [32].

The physical processes being measured in this NEUP are mixed convection and forced convection stemming from a heated vertical plate in a vertical rectangular tube. The working fluid being measured is air in different convection scenarios. Both mixed and forced convection for buoyancy aided and opposed convection were investigated. For each case, the full set of measurements conducted for this validation experiment are:

- Inlet velocity profiles for forced, mixed, buoyancy aided, and opposed convection each with a heated plate having constant heat flux
- Boundary layer velocity profiles at three streamwise locations for each case
- Heat flux from the wall at three streamwise locations
- Pressure drop through the test section for each case
- Atmospheric conditions for each case
- Experimental setup and settings for each case
- Geometric measurements of the test section
- Temperature of all four test section walls and wind tunnel inlet for each case
- PIV measurements for flow velocity field.

Although this NEUP was preceded by multiple experiments focusing on convection stemming from a heated plate and the resulting flow laminarization such as Wang et al. [33] and Hattori et al. [34], the results obtained by this research work could potentially qualify as a high-quality benchmark according to the standards published by Oberkampf [35]. The experimental conditions have been clearly documented with the detailed facility geometry, and the properties such as inlet velocity profiles have been measured thoroughly. To disseminate the data to a wider audience, a public website has been created by this NEUP team with all of the experimental data and necessary descriptions to perform validation work available for unlimited download [36]. Given the well-organized descriptions and experimental documentation, this is an excellent example of finalizing and preserving the data generated by a publicly funded DOE NEUP project.

3.1.4 NEUP No. 11-3218: CCNY Forced and Natural Convection Facility [37]

The main objective of this project is to identify and characterize the conditions under which abnormal heat transfer phenomena would occur in a VHTR with a prismatic core design. This facility is designed to obtain heat transfer coefficient data at high temperatures and pressures for gas flows with different Reynolds numbers. High-pressure and high-temperature experiments are conducted to obtain data that could be used for validation of VHTR design and safety analysis codes. The focus of these experiments is to generate benchmark data for design and off-design heat transfer for forced, mixed and natural circulation in a VHTR core. In particular, a flow laminarization phenomenon is intensely investigated since it could give rise to hot spots in the VHTR core.

The experimental facility at the City College of New York (CCNY) is designed to mimic the HTTF at OSU, so careful considerations ensured that the dimensionless parameters, such as the Reynolds, Prandtl, and Grashoff numbers, are matched the HTTF and Prototype MHTGR. As shown in Figure 16, the test section is made of thermally isotropic graphite G348 with a thermal conductivity of 128 W/mK at 298 K and is housed in a stainless-steel pressure vessel rated to 69 bar at 623.15 K with the American Society of Mechanical Engineers (ASME) standards. The diameter of the cooling channel in the graphite is 16.8 mm through which the gas is forced, with four smaller heating channels equally spaced around the cooling chamber with a diameter of 12.7 mm. The test section is equipped with instrumentation to provide high-fidelity validation data. The instruments included 40 type k thermocouples at 10 different locations along the test section to provide boundary conditions along with varying radial depths, hot-wire anemometry (HWA) for flow measurements, pressure transducers, and sensors. Later, a second test vessel identical to the first is added to the facility and connected on the top of the first test vessel to facilitate natural circulation experiments. A mass flow meter is added to the connective piping for natural circulation measurements as well as extra thermocouples to measure the inlet and outlet temperatures of each test tube.

(a) (b)

Figure 16. High-pressure and high-temperature test facility for (a) forced convection and (b) natural circulation experiments. Adapted from [37].

The experiments performed in this NEUP project are focused on heat transfer from graphite under different regimes of turbulence, whether the flow direction is upward or downward, and whether the flow is mixed or forced. The gases used in the experiments are nitrogen, air and helium. To conduct an experiment, the graphite is heated to the desired temperature, and then once the desired temperature is reached, the gas flow is adjusted to the regime being investigated. There is a wide range of different temperatures, Reynolds numbers, and gas flows measured, resulting in a total of 105 runs tested. These are all shown in Table 6. Each experiment is run until a steady state is reached, which is defined by graphite temperature variation under 3 K per hour. Natural circulation is carried about by allowing one vessel, the hot vessel (HV), to be hotter than the other one, the cold vessel. The density differences are created, allowing for natural circulation to develop, from which the heat transfer coefficient is measured. These experiments are conducted when natural circulation is measured with the HV at 373–800 K and 10–64 bar.

The experimental data resulting from this NEUP project is not yet publicly available. To perform code benchmark studies, researchers will need to digitize the final report data plots or contact the PI (Prof. Kawaji) for more information

Table 6. Range of experimental conditions for forced convection experiments. Adapted from [37].

Gas Type	N_2 (upward)	He (upward)	He (downward)
Number of runs	37	41	27
Flow rate [SLPM]	30–220	50–500	50–500
Heater power [W]	1,520–6,800	1,520–6,800	1,520–6,800
Pressure [bar]	34–64	34–64	34–64
Inlet temperature [K]	291–298	291–298	291–298
Graphite temperature [K]	353–965	353–813	353–813
Inlet Reynolds number	1,300–14,000	560–5,190	560–5,190

3.1.5 NEUP No. 13-4953: Plenum-to-Plenum Facility (P2PF) [38]

Missouri University of Science and Technology has constructed a Plenum-to-Plenum Facility (P2PF) under NEUP Project No. 13-4953 as a separate effects test facility to compliment the integral test facility, OSU's HTTF. The P2PF is designed to measure and replicate the plenum-to-plenum thermal-hydraulics flow for a prismatic HTGR, and the LOFC accident is one of the scenarios that the P2PF can produce validation data for. As shown in Figure 17, the facility has an electrically heated (riser) channel and a water-cooled coolant channel to simulate HTTF channels at different radial locations. As illustrated in Figure 18, the upper plenum is surrounded by a cooling jacket that can determine the upper plenum outer temperature. The upper plenum is also water cooled to simulate the RCCS heat removal system for the upper plenum and to calculate the total heat removal through natural circulation. The lower plenum remains adiabatic. The ¼-scale facility mimics the scale of the HTTF but is only half as long as the HTTF. This facility has HWA measurements for flow along with thermocouples and heat flux sensors for boundary condition measurements. Other instrumentation includes noninvasive microfoils for both local instantaneous temperature measurements and heat flux measurements. More detailed information for the P2PF setup can be found in the project final report [38].

The experiments done at this facility can be separated into two groups. The first includes natural circulation heat transfer experiments with differing temperatures and pressures as well as differing heat fluxes. These circulation experiments are done with both air and helium. Properties investigated in this first section include the dimensionless numbers, such as Nusselt, Gasthof, Rayleigh, and Prandtl numbers, for a range of temperatures and pressures. The effect of outer surface temperature on the inner heat transfer and gas dynamics is investigated to maximize inner heat transfer and minimize heat loss to the environment. The unique facility design allows for local property measurements to compare to the bulk property measurements also being taken. These numbers are calculated using the measurements taken from the instrumentation. The experiments are operated by filling the facility with dry helium to the desired temperature and adjusting the heat flux to the desired level for that test. These tests are kept running until the steady-state condition is reached, which is defined as a change under 0.5 K in 30 minutes.

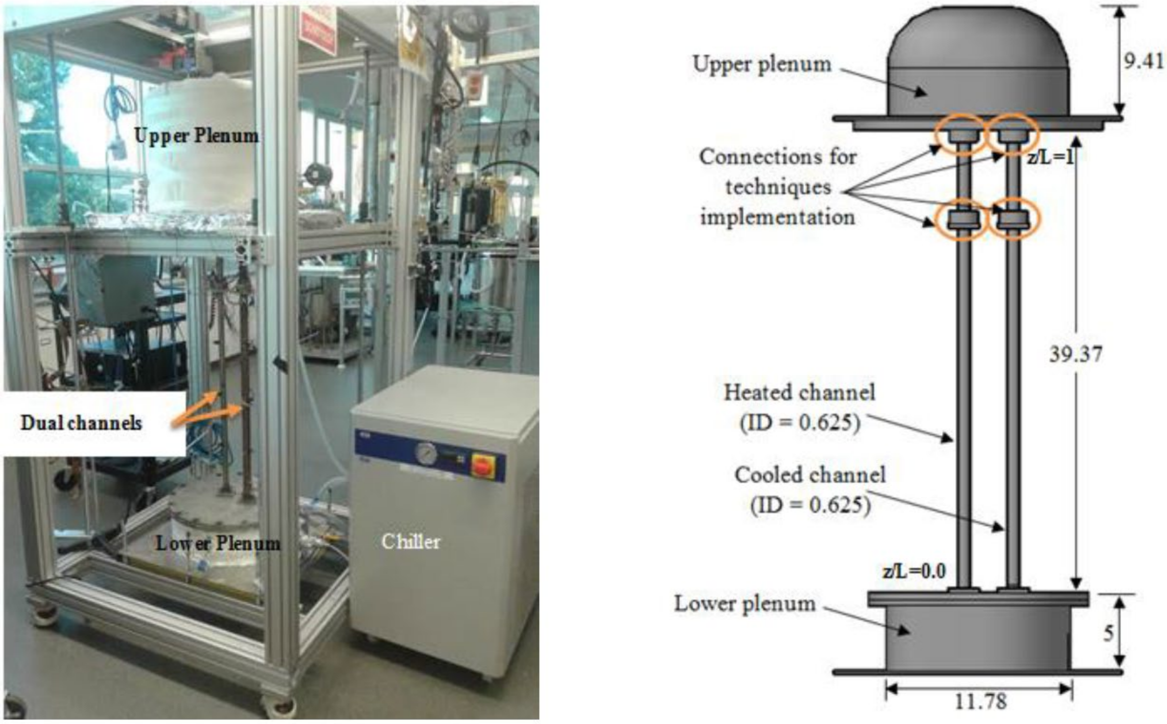


Figure 17. (a) Picture of P2PF preliminary design and (b) schematic diagram of P2PF preliminary design (all dimensions in inches). Adapted from [38].

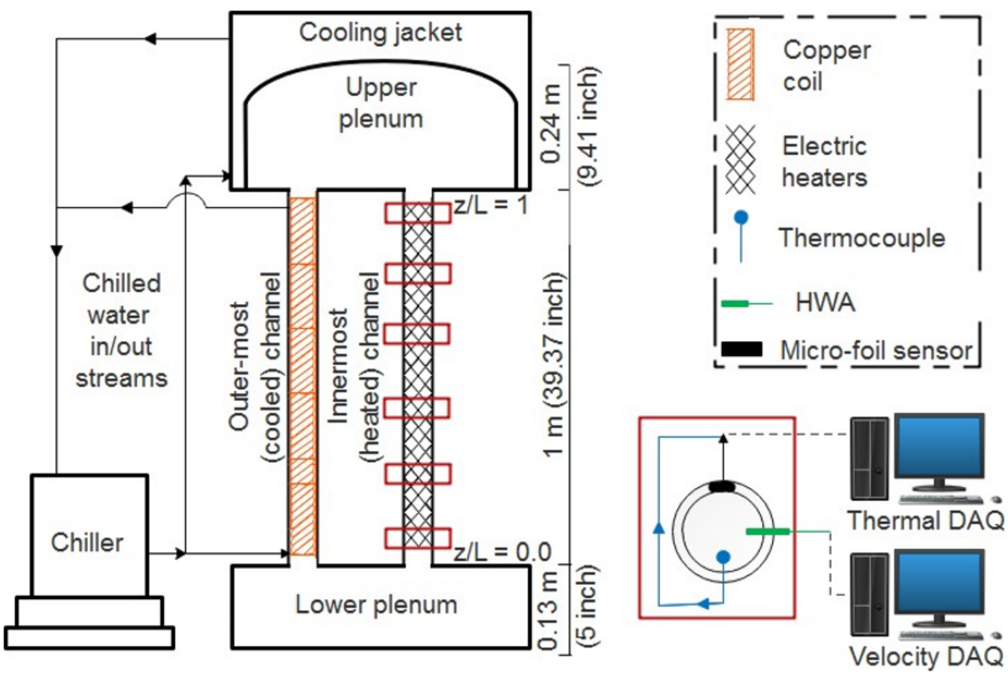


Figure 18. Flow diagram for the P2PF design. Adapted from [38].

The second type of experiments investigates the gaseous dynamics of the naturally circulating gases using the P2PF. The gaseous dynamics were investigated for differing upper plenum outer surface temperatures. The measurements are done using HWA placed at different axial locations with the ability to radially adjust the probe in 1-mm increments. The procedure for the experiment is to pump dehumidified air at atmospheric pressure into the test section and then turn on the chiller at the desired temperature and heat flux. The test section is left for 3 hours to reach a steady state, after which data is recorded.

Beside the experimental work, there are CFD investigations performed with the resultant experimental data. CFD simulations are carried out using air ranging from 15 to 75 psi, with differing heating and cooling values. Both COMSOL and STAR-CCM+ have been used to perform the CFD simulations, and the simulation setup and results can be found in detail in Ref. [39].

The experimental data resulting from this NEUP project is not yet publicly available. To perform code benchmark studies, researchers will need to digitize the final report data plots or contact the PI (Prof. Muthanna Al-Dahhan) for more information.

3.1.6 NEUP No. 13-5000: UW WRCCS Facility and UM RCCS Separate-Effect Test Facility [40]

NEUP Project No.13-5000 is led by the University of Michigan (UM) in Ann Arbor and the University of Wisconsin (UW) in Madison collaborated. This NEUP project had two parts. The first was focused on the RCCS facility at UW and making design recommendations for the RCCS based on data analysis. The UW Water-based RCCS (WRCCS) facility shown in Figure 19 is an open loop asymmetric natural circulation facility, including an elevated water storage tank, a downcomer for water supply to the heated test section, a heated enclosure test section to represent the outside of a reactor pressure vessel and cavity walls, and an adiabatic chimney. The system is completely closed with the exception of a steam outlet at the top of the tank, where the steam is condensed and stored as a final reference for water inventory losses. While the UW WRCCS facility is important and should be looked at for validation purposes for the HTGR's RCCS design, the final report of this NEUP project is focused on the second part where a RCCS separate effects test facility was constructed at UM for CFD validation purposes.

The UM RCCS facility was built to investigate and validate 3D natural thermal mixing in the upper plenum of the RCCS stemming from multiple riser ducts to examine possible thermal stresses on material in the RCCS system. As shown in Figure 20, the facility has been fabricated in transparent acrylics to allow access for optical laser measurements like PIV. The working fluid of the facility (i.e., water) is stored in two tanks of 400 liters and one tank of 2,000 liters. The upper plenum of the test section has the inner dimensions of length × width × height = 470.6 × 249.3 × 462 (mm^3) and two exhaust pipes with an inner diameter of 76.2 mm. A pack of six risers, whose distance between neighboring risers is 12.7 mm, was attached to the bottom of the plenum. Each riser has a 12.7 × 63.5 (mm^2) cross section. The water flow is driven through a conditioning section with honeycomb grids and long channels to obtain a uniform water jet flow and to minimize the turbulence level. The axial length L_{riser} of the risers is 800 mm, yielding a ratio of $L_{riser}/D_r = 38.1$ (D_r is the hydraulic diameter of the riser), which is sufficiently long for the flow to reach a fully developed state.

The experimental campaign for the UM RCCS is focused on different numbers of jet plumes mixing with various jet velocities in the upper plenum of the RCCS. The jet plumes are shot into the plenum in groups of two, four, or six plumes and measured to compare different configurations. This concludes in datasets for multiple different Reynolds numbers ranging from 9,000 to 14,000 at room temperature and under standard pressure condition. The experiments are repeated for statistically converged results and the measurement uncertainties are quantified. As for the CFD code V&V study, extensive analyses using STAR-CCM+ with a Reynolds-Averaged Navier-Stokes (RANS) Standard k-epsilon (k-ε) model are performed on the jet plumes mixing behaviors, and good agreements have been reached for velocity profiles on an area close to the inputs of the jet plumes. However, further away from the plumes near the exhaust where there exists recirculation still show some discrepancies between the CFD simulations and the experimental observations.

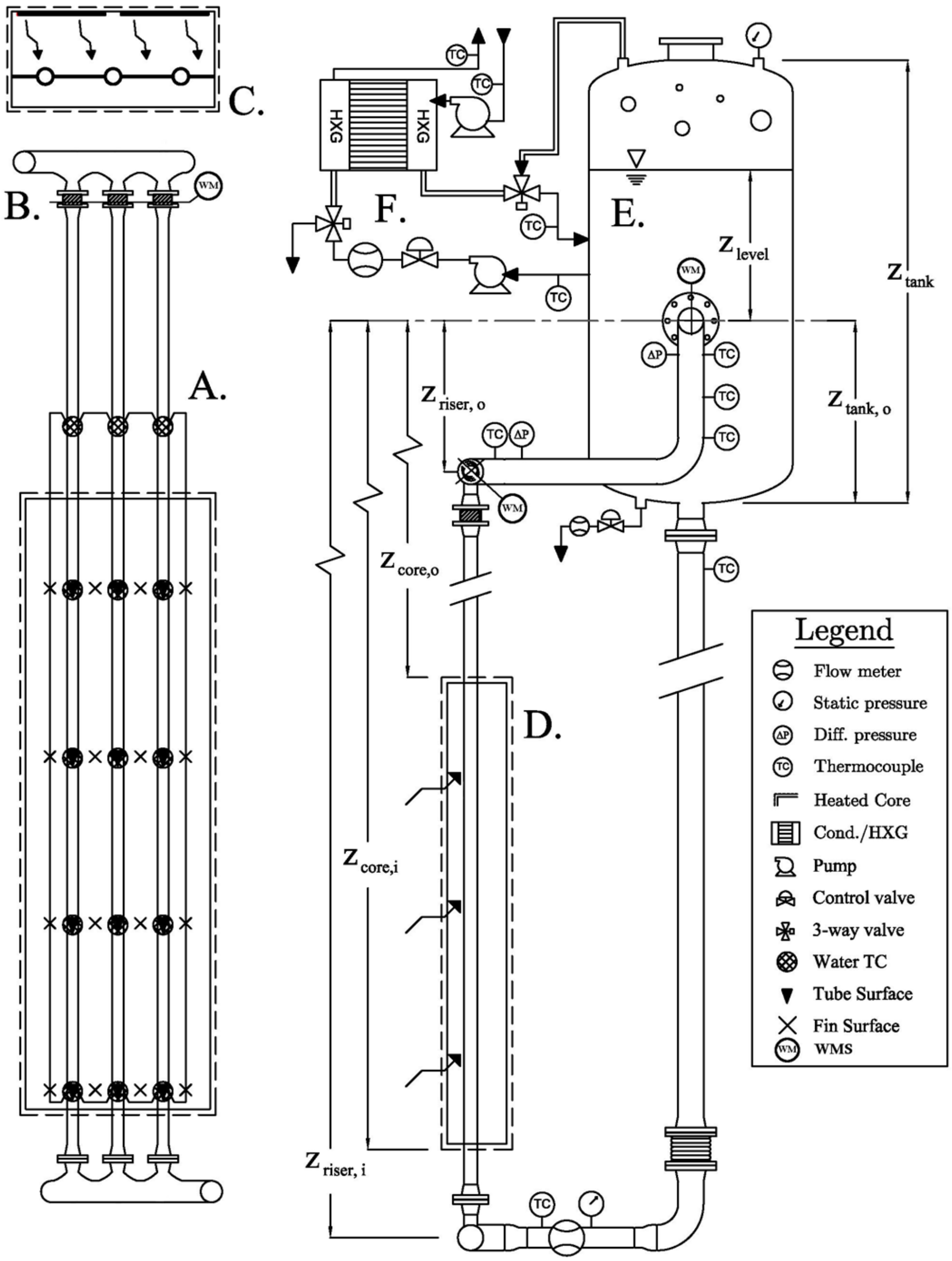


Figure 19. UW Water-cooled RCCS design schematic. A: Test section, B: Outlet header, C: Heated enclosure (top view), D: Heated cavity (front view), E: Water storage tank, F: Condenser/Heat removal loop. Adapted from [41].

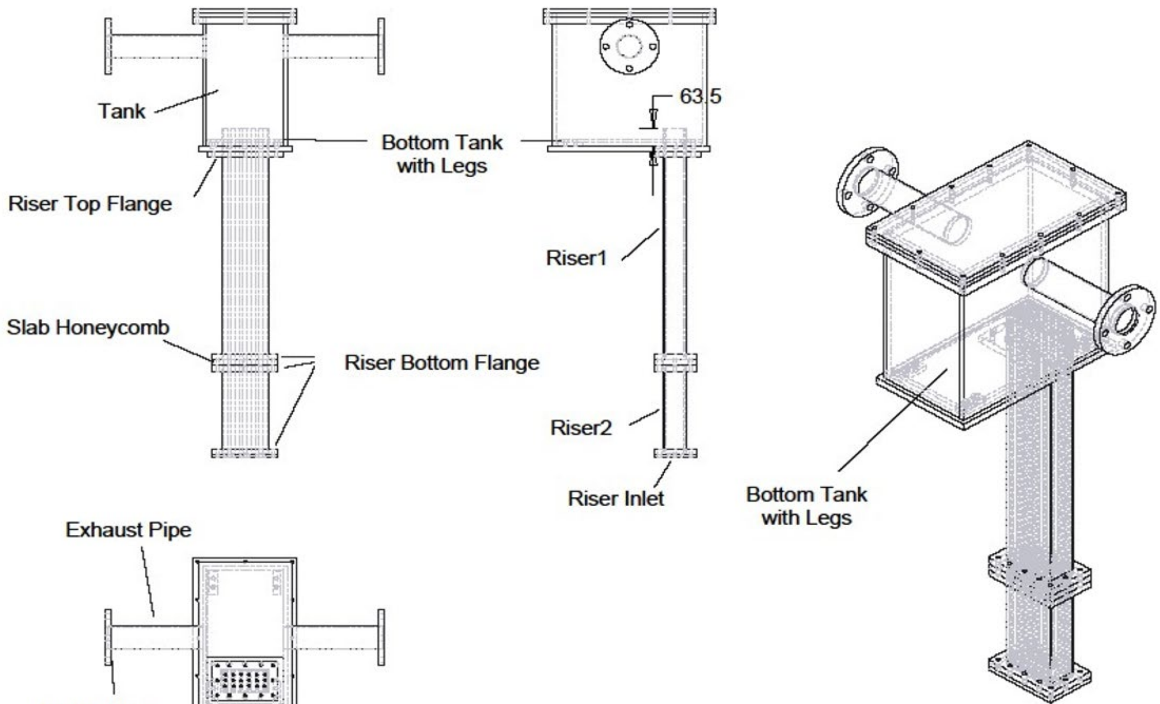


Figure 20. Drawings of the UM RCCS separate effects test facility. Adapted from [40].

All the data resulting from this NEUP project is available on request from Dr. Victor Petrov and Dr. Annalisa Manera at the University of Michigan. The data from this experiment has not been used for a technical benchmark, but was used to support the Integrated Research Project (IRP) under IRP-NEAMS-1.1 "Center of Excellence for Thermal-Fluids Applications in Nuclear Energy: Establishing the knowledgebase for thermal-hydraulic multiscale simulation to accelerate the deployment of advanced reactors." Further work was suggested to use a large eddy simulation to investigate those discrepancies.

3.1.7 NEUP No. 15-8205: CCNY Bypass Flow Test Facility [42]

This NEUP project made use of the same CCNY facility built for NEUP Project No. 11-3218 (Section 3.1.4) but was modified slightly to focus on the forced convection and bypass flow phenomena, natural circulation flow and heat transfer, and graphite oxidation due to air ingress that could occur in a prismatic HTGR design. As this project investigated the air ingress and bypass phenomena, the graphite test tubes were modified to have a small rectangular slot down the length for air to flow through. As shown in Figure 21, the bypass channel has a cross section of 40 × 3 mm in addition to the heater and cooler channels of the other test sections, along with a porous disc to ensure uniform flow into the two channels. The natural circulation facility was modified to allow for mixing experiments from helium and nitrogen by adding mass flow rate measurements to the tubing between the hot and cold vessels.

Using nitrogen as the working fluid, the test section is pressurized to a wide range of pressures and temperatures for the bypass flow experiment campaign. Data is then taken on the differing properties of nitrogen flowing through the cooling and bypass channels. This is done by installing more instrumentation, such as thermocouples and mass flow meters, on the bypass channel.

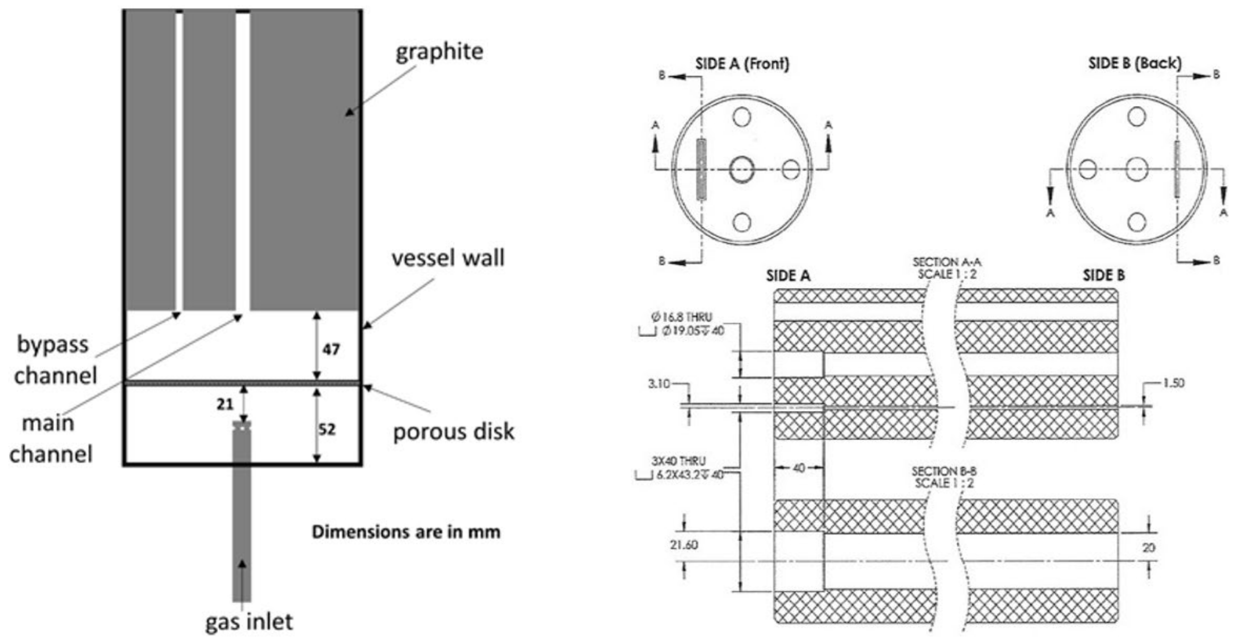


Figure 21. Modification to the Natural Circulation Test Facility to measure bypass flow. Adapted from [42].

The natural circulation mixing experiments with helium and nitrogen have been carried out using the natural circulation test loop illustrated in Figure 22, but with a modified gas input system to allow for air ingress experiments. The air ingress experiments are then carried out by filling the facility with helium to a desired pressure and then injecting nitrogen into the helium environment. Once the nitrogen is injected to its desired partial pressure, the facility is left for 10 minutes to allow the denser nitrogen to settle in the lower plenum. Once the gas has separated, the heater gets triggered, and the needle valve is closed to prevent circulation. Then the heaters are started until the graphite midpoint temperature reaches the desired value, where the needle valve is opened to allow natural circulation to begin. Connected to the lower and upper plena are gas port for sampling the gas to determine its volumetric components. The gas is then sampled at fixed time periods, and the experiment is set to be completed when the volumetric components reach a steady state in both the upper and lower plenum.

All data from this project has been transferred to INL, and will be made available as part of the online access (see Section 4).

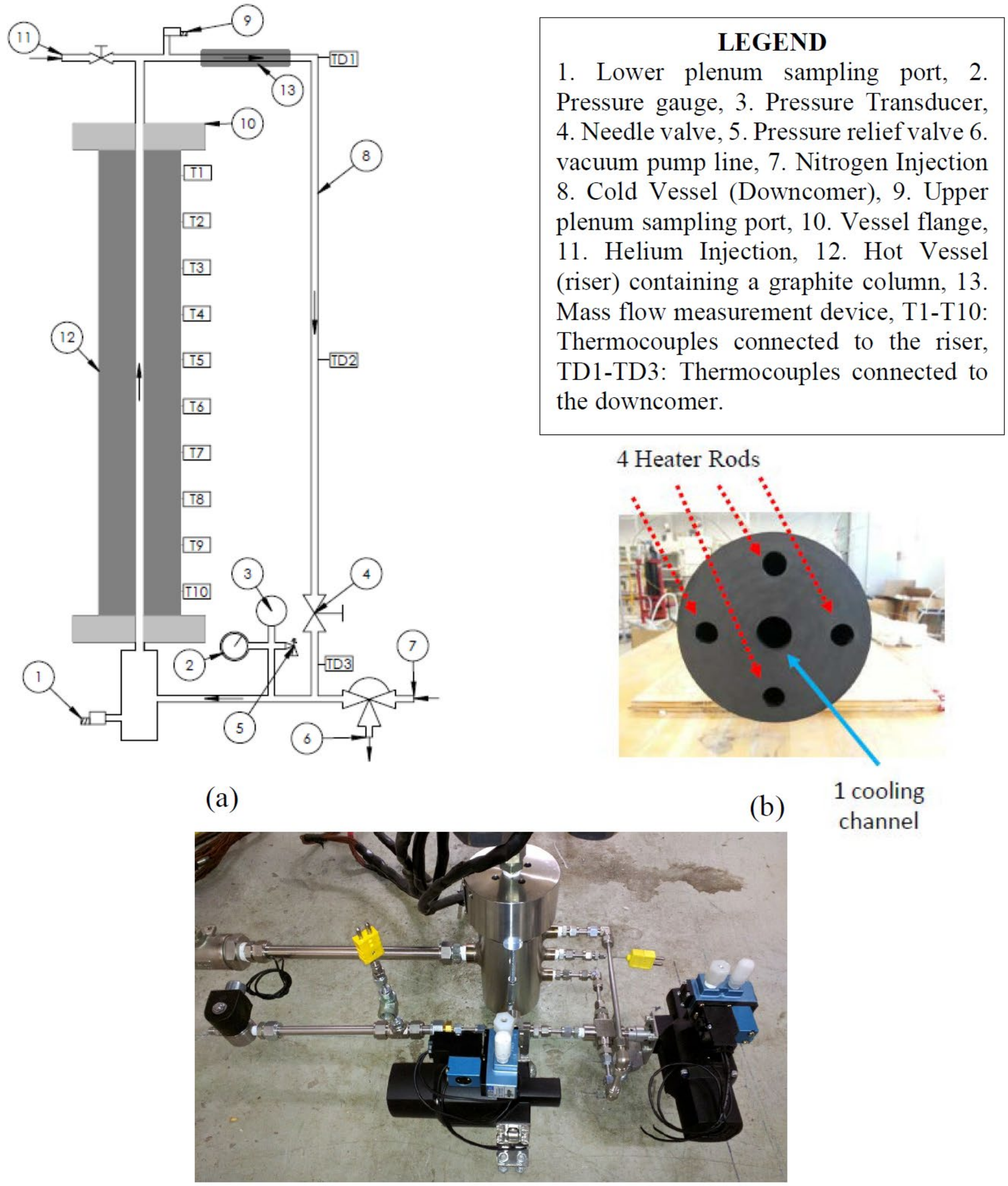


Figure 22. (a) Natural circulation experimental test loop, and (b) Cross-sectional view of the graphite column in HV, (c) lower plenum and gas sampling system. Adapted from [42].

3.1.8 NEUP No. 16-10244: High Temperature Test Facility [43]

The HTTF is an integral effect test facility at OSU that is a 1/4 scale facility in length and diameter compared to the GA MHTGR reference design. This facility is designed to validate codes for HTGR normal operations as well as accident scenarios, such as PLOFC, DLOFC, etc. The modular design allows testing different core types, even allowing the HTTF to be converted into a pebble-bed reactor design.

There are also a variety of different break locations to test different sizes and locations of potential breaks in the reactor [9]. The reactor is surrounded by the RCCS to control the boundary conditions along the outer edge for CFD analysis purposes. Because the facility is a large integral-level test facility, the instrumentation list is vast, including hundreds of thermocouples in various axial and radial locations, as well as pressure taps and gas concentration instruments. The facility core region consists of several ceramic blocks stacked on top of one another. As shown in Figure 23, the core sections are filled with several channels consisting of heater channels, three different sizes of cooling channels, bypass flow channels, and instrumentation channels. Given the fact that the facility description for the HTTF is very complex, there are multiple reports detailing various details, such as design updates and instrumentation packages [44-46].

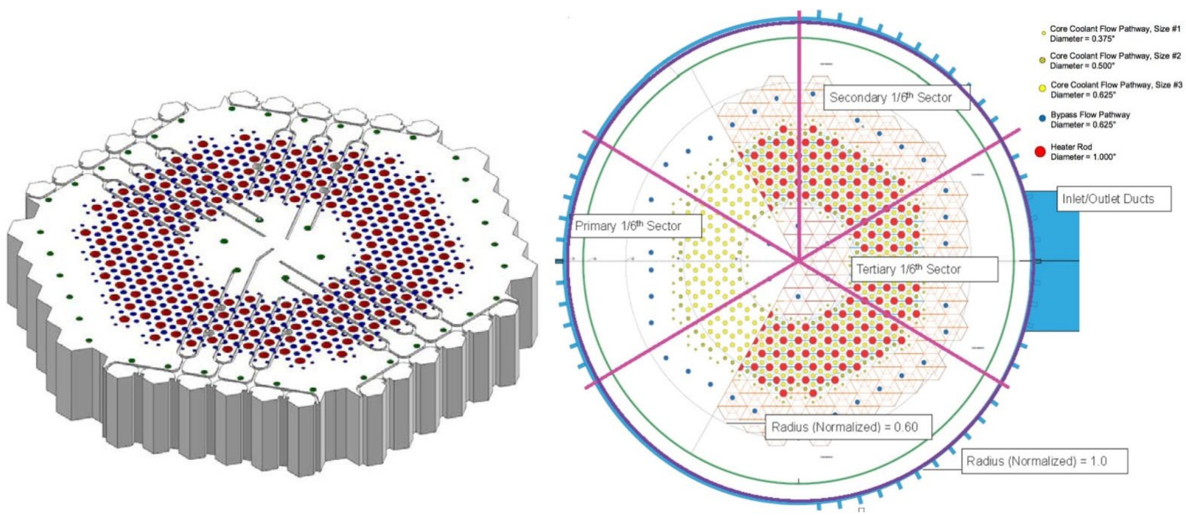


Figure 23. On the left is a simulated model of a single core block, which stack to form the main channels, while on the right is a drawing from the technical design report showing a cross section of the main core channels. Adapted from [9].

Being an integral effect test facility, the unique experiments conducted at the OSU HTTF facility are very important for validating the HTGR design and performing reactor safety analyses. In this NEUP project, eight experimental tests using the HTTF were completed with different scenarios and specifications. Table 7 provides a description of the tests completed during this NEUP work and their initial conditions for the RCCS and the primary coolant loop. The results of each experiment are presented in the final report [43]. Alongside the project final report, each experiment had a Test Acceptance Report issued that detailed features about the test, including a description of the test, test log, assessment of the test data, critical instrument list for that experiment, data qualification assessment, acceptance criteria, all data plots, test data file identification, test procedures, baseline configuration, change requests, and calibration records [47-54].

Since the detailed experimental setup and time-resolved data are available at OSU, several code validation options exist. Some of the HTTF's test experiments conducted using the HTTF setup have been modeled and performed the code validation, such as using RELAP5-3D to model a depressurized conduction cooldown experiment by Epiney [55]. This work is part of the larger OECD/NEA benchmark that has recently been launched based on the HTTF PG26 test data, but access to this data set will be limited to the participants. However, despite the vast instrument list, HTTF did not measure the primary coolant flow rate, which can present a challenge for some HTTF validation studies. The experimental data resulting from this NEUP project is not yet publicly available. To perform code benchmark studies, researchers will need to digitize the final report data plots or contact the PI (Prof. Izabella Gutowski or Brian Woods at OSU) for more information.

Table 7. Experiment cases conducted using HTTF at OSU during this NEUP study. Adapted from [43].

Test No.	Test Conditions	Test Purpose
PG-28	Lower plenum mixing test with low power (<350 kw)	Investigate impact that changes in mass flow rate have on thermal striping in the lower plenum
PG-29	Low-power double inlet-outlet crossover duct break	Perform a depressurized conduction cooldown test with the break coming from a double ended crossflow duct
PG-30	Low-power plenum mixing constant temperature test	Investigate impact mass flow rate has on thermal velocity profiles and mixing in the lower plenum
PG-31	Low-power pressure vessel bottom break with restored forced convection cooling test	Investigate pressure vessel bottom break transient with restored circulation
PG-32	Low-power asymmetric core heatup	Investigate the response after the core is heated up asymmetrically
PG-33	Zero-power long term cooldown test	To investigate passive cooling from natural circulation of the reactor core
PG-34	Low-power asymmetric core heatup with full hybrid heater	Another asymmetric core heat up with different initial conditions and heater parameters
PG-35	Zero-power crossover duct exchange flow and diffusion	To investigate the flow in the crossover duct and diffusion of gases

3.2 Knowledge Gaps and Benchmark Suggestions

It is clear from the preceding sections that there is more experimental data generated than code validation benchmark studies performed. This section will summarize current knowledge gaps that could be used for future NEUP scope calls, and provide suggestions for future benchmark studies that can be based on these NEUP projects.

3.2.1 Knowledge Gaps in Nuclear Energy University Program Projects

At least two high-importance with low-to-medium knowledge level areas have not yet been covered extensively with NEUP-funded projects as shown in Table 3 together with the NRC PIRT study for HTGR accident and thermal-fluid analysis [15]:

- Fluid stratification at air ingress accidents
- Plenum mixing behaviors during normal operations and accident scenarios.

Additionally, load change or transient operation scenarios should receive more attention for later NEUP proposal initiatives. After examining the completed and ongoing NEUP-funded HTGR projects, we would also like to point out that the following areas could be included in future NEUP investigations:

- Decay heat during LOFC scenarios
- Core heat transfer during LOFC scenarios

- Thermal fatigue analysis in HTGR's upper plenum structures.

3.2.2 Benchmark Opportunities

Besides the experimental investigations mentioned in Section 3.1, a few other NEUP projects generated high-quality experimental data but with limited benchmark studies. We would recommend reviewing the data generated by the NEUP projects listed below as sources of future CFD and system code validation efforts:

- 09-784: Investigation of Countercurrent Helium-air Flows in Air-ingress Accidents for VHTRs [56]
- 11-3079: Modeling and Test Validation of a Reactor Cavity Cooling System with Air [57]
- 12-3759: Experimental and CFD Studies of Coolant Flow Mixing Within Scaled Models of the Upper and Lower Plenum of a Prismatic Core VHTR [58]
- 14-6435: Fluid Stratification Separate Effects Analysis, Testing and Benchmark [59]
- 15-8627: Experimental Validation Data and Computational Models for Turbulent Mixing of Bypass and Coolant Jet Flows in Gas-Cooled Reactors [60]
- 17-13115: Experimental Determination of Helium/Air Mixing in Helium Cooled Reactor [61].

4. WEB PLATFORM FOR NUCLEAR ENERGY UNIVERSITY PROGRAM HIGH-TEMPERATURE GAS-COOLED REACTOR PROJECTS DATACENTER

Many of the HTGR-related NEUP projects have produced valuable computational and experimental results ([31, 40, 42]), but the actual data from these projects are still only available on request from the university PIs. One example of an optimal NEUP project close-out can be found in NEUP Project No. 11-3081. As a brief recap, the research group at USU conducted their experiments using a RoBuT facility for vertical plate heat convection, and at the conclusion of the project, set up a public online access website for the dataset and all supporting documentation that will be required to develop detailed validation models [36], as shown in Figure 24.

Given this example and the general requirements of a validation database, our vision of the INL HTGR NEUP data platform is to provide a comprehensive database for all completed NEUP projects, with links to the datasets and documentation for both the computational and experimental work achieved in the HTGR-related NEUP projects.

Experimental Validation Data for CFD of Steady and Transient Mixed Convection on a Vertical Flat Plate

Download

Blake W. Lance, Utah State University

Description

Simulations are becoming increasingly popular in science and engineering. One type of simulation is Computation Fluid Dynamics (CFD) that is used when closed forms solutions are impractical. The field of Verification & Validation emerged from the need to assess simulation accuracy as they often contain approximations and calibrations.

Validation involves the comparison of experimental data with simulation outputs and is the focus of this work. Errors in simulation predictions may be assessed in this way. Validation requires highly-detailed data and description to accompany these data, and uncertainties are very important.

The purpose of this work is to provide highly complete validation data to assess the accuracy of CFD simulations. This aim is fundamentally different from the typical discovery experiments common in research. The measurement of these physics was not necessarily original but performed with modern, high fidelity methods. Data were tabulated through an online database for direct use in Reynolds-Averaged Navier Stokes simulations. Detailed instrumentation and documentation were used to make the data more useful for validation. This work fills the validation data gap for steady and transient mixed convection.

The physics in this study included mixed convection on a vertical flat plate. Mixed convection is a condition where both forced and natural convection influence fluid momentum and heat transfer phenomena. Flow was forced over a vertical flat plate in a facility built for validation experiments. Thermal and velocity data were acquired for steady and transient flow conditions. The steady case included both buoyancy-aided and buoyancy-opposed mixed convection while the transient case was for buoyancy-opposed flow. The transient was a ramp-down flow transient, and results were ensemble-averaged for improved statistics. Uncertainty quantification was performed on all results with bias and random sources.

An independent method of measuring heat flux was devised to assess the accuracy of commercial heat flux sensors used in the heated wall. It measured the convective heat flux by the temperature gradient in air very near the plate surface. Its accuracy was assessed by error estimations and uncertainty quantification.

OCLC

985526239

Document Type

Dataset

DCMI Type

Dataset

File Format

.csv, .pdf

Publication Date

2015

ADDITIONAL FILES

Data.zip (13053 kB)
MD5: *0206abc512b00edfd182e3c30988678b*
Aid-BC-AtmCond.csv (1 kB)
MD5: *d4db291e275912d746eca81b9cfb13cd*
Aid-BC-HeatedWallTemp.csv (7 kB)
MD5: *5395739959ed9d4af353d8abdeddef98*
Aid-BC-InletTemp.csv (1 kB)
MD5: *44387a1b86698be28256022ea987bbe6*
Aid-BC-Inlet-Vel.csv (270 kB)
MD5: *c4388b1857a7273be3c0d7d3b81a2c53*
Aid-BC-LeftWallTemp.csv (1 kB)
MD5: *5065ed2943a7005e7345051664f7ed4d*
Aid-BC-RightWallTemp.csv (1 kB)
MD5: *a5107835afb25b406dae12bdf81a1fa8*
Aid-BC-TopWallTemp.csv (1 kB)
MD5: *5618c9b22c86dc70d6a3dade0f85105f*
Aid-SRQ-HeatFlux.csv (1 kB)
MD5: *950ec2d3f17a6fd9f7d76fb79edd9d5d*
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MD5: *829707f025c8babaa2b84fb293e100bf*
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MD5: *97795300beef4fe1fde3e3a528e6bef2*
Aid-SRQ-Vel_x2.csv (24 kB)
MD5: *6aa88863799c5e8288c0386721437c0f*

Figure 24. USU online database for NEUP Project No. 11-3081 using RoBuT facility. Adapt from [36]. Accessed on 8/21/22 at https://digitalcommons.usu.edu/all_datasets/8/.

4.1 Database Structure

Based on the review for the NEUP final reports and related publications, we have initialized the construction of a preliminary data platform, as shown in Figure 25, powered by Microsoft® SharePoint Platform to link the available technical reports, journal publications, and conference proceedings from the corresponding research projects. In FY 2023, datasets generated by CFD or system code models (and in best-case scenarios, the actual models themselves), experimental facility descriptions, and all academic products related to the project will be also included in the platform for each NEUP project as part of the future work scope. The ultimate goal is that access to this platform will be publicly available to the HTGR research community via the ART-GCR program website [62] by the end of FY23.

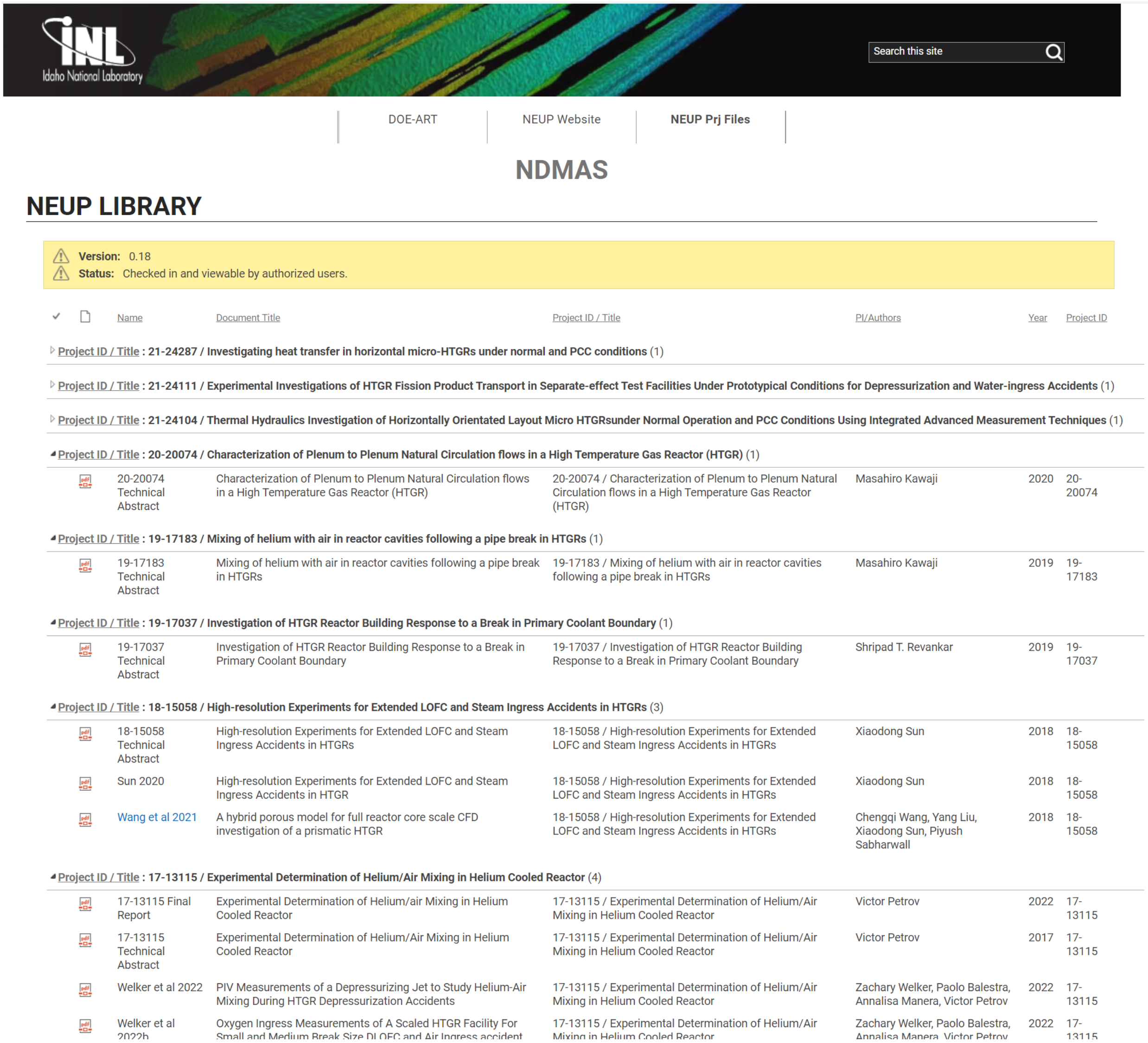

Figure 25. Preliminary NEUP library constructed on the ART-GCR NDMAS platform, containing the final report, abstract, and related publications for all 29 HTGR-related NEUP projects.

4.2 Plan for Data Transfer Process and Future Development

In order to collect the data from the NEUP-funded HTGR projects, we will propose setting up a cloud platform so that the PI can upload the experimental raw data and simulation files for each project accordingly. The details of this process will be developed, tested and disseminated to NEUP PIs in FY23.

The available experimental and computational data generated by these NEUP projects will then be gathered and disseminated on a publicly accessible online data platform, most likely via the ART-GCR website (www.art.inl.gov). This data platform will be continuously maintained to include newly funded HTGR-related NEUP projects in the future.

With regular maintenance and refinements, the ultimate goal of this work is to fully leverage the potential of these DOE-funded experimental datasets by providing a comprehensive single-entry access point for researchers in the HTGR industry, national laboratory, regulator, and academic communities.

5. CONCLUSION AND FUTURE WORK

A total of 29 NEUP-funded HTGR thermal-fluid projects were funded by the DOE Office of Nuclear Energy between FY 2009 and 2021, but each project has been conducted solely under the awarded university's supervision, and there is no comprehensive database or platform that identifies, collects, and stores the products of these projects for code V&V purposes by the HTGR community. Therefore, the ART-GCR program has identified the urgent need to perform an extensive literature survey to assess completed and ongoing NEUP-funded HTGR projects with the aim to develop a public-accessible data platform that can retrieve code validation data and guide future NEUP funded research as well.

After investigating the final report and other related publications for these awarded NEUP projects since FY 2009, the complete list shown in Table 2 has been created with each project title and PI's information. Based on the NRC PIRT study for HTGR thermal-hydraulics and accidents, the thermal-fluid phenomena and operating scenarios for these studies have been identified and summarized in a tabulated format (Table 3), enabling a direct view of the achievements as well as the knowledge gaps for these NEUP-funded HTGR projects. This comprehensive review also determined if the NEUP projects have conducted experimental investigations, performed computational , or both. Table 4 reviewed the experimental data types the NEUP project has collected and the simulation tools used in any available code benchmark study. Among these NEUP projects, eight projects that constructed experimental facilities have been selected in FY22 for review, and their experimental setups have been briefly discussed and summarized. Based on the data availability and quality, some NEUP experimental work has been identified for future code benchmark candidates. Additionally, potential phenomena and data knowledge gaps have been identified to support for future NEUP funding decisions.

Built upon the extensive survey we performed for all NEUP reports and related publications, we constructed a preliminary data platform (shown in Figure 25) with Microsoft® SharePoint, and summarized the completed project meta data, such as the project abstract, final reports, and resultant scientific publications. This data platform is still under development; therefore, it is not publicly available yet.

In FY23, the ART-GCR program will collaborate with university PIs to gather more detailed information on their NEUPs and incorporate the available data generated to finalize the NEUP-HTGR data platform. The intention is that the platform will be publicly accessible on the ART-GCR website by the end of FY23, and maintained with new NEUP-HTGR projects. The ultimate goal of this work is to leverage and disseminate these DOE-funded experimental datasets by providing a comprehensive single-entry access point for researchers in the HTGR industry, national laboratory, regulator, and academic communities.

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