Development of Critical Experiments to Provide Validation Data for Multiphysics Coupling Methods

Nuclear Energy Advanced Modeling and Simulation

Peter F. Caracappa
Rensselaer Polytechnic Institute

David Henderson, Federal POC
Timothy Valentine, Technical POC

Project No. 15-8101
Development of Critical Experiments to Provide Validation Data for Multiphysics Coupling Methods
Final Report

Principal Investigator
Peter F. Caracappa* (October 1, 2015 – August 15, 2018)
Wei Ji (August 15, 2018 – December 31, 2018)
Rensselaer Polytechnic Institute
*Now at Columbia University

Collaborators
Wei Ji (October 1, 2015 – August 15, 2018)
Peter F. Caracappa* (August 15, 2018 – December 31, 2018)
Michael Z. Podowski
Rensselaer Polytechnic Institute

Chang-ho Lee
Vijay S. Mahadevan
Argonne National Laboratory

Graduate Students
Mathieu Dupont
Matthew Eklund

NEUP Project: 15-8101
Grant Number: DE-NE0008439

March 31, 2019
Executive Summary

Computational simulation has become one of the most important tools in the modern toolbox for nuclear reactor design. Reactor simulations, like the reactors themselves, involve multiple physical processes including neutronics, fluid flow and heat transfer (thermal-hydraulics), thermo-mechanics, fuel behavior, chemistry, and balance of plant. Each process can affect the behavior of the others, and advanced computational tools have been developed that couple separate physics models together to simulate the feedback effects.

The objective of this work is to generate experimental data to support the validation of multi-physics reactor codes. It specifically targets the SHARP toolset from the NEAMS Reactor Product Line, but it is intended to be useful in any set of coded that couples thermal-hydraulics with neutronics simulations. The project takes advantage of the inherent flexibility of the RPI Reactor Critical Facility (RCF), a low power open-pool research reactor.

The standard configuration of the RCF is used to explore the effects of moderator temperature on neutronic behavior and reactivity. Because the low power of the reactor generates no appreciable temperature change in the moderator, the moderator temperature is artificially changed with two 18 kW electric immersion heaters. The temperature feedback effects can be observed by recording the change in the reactor power as the reactor goes from a supercritical state, to critical, to a subcritical state.

To expand the temperature range and temperature rate of change that can be investigated, an experimental apparatus consisting of a heated water loop was designed and installed in the RCF. In this setup, water from a heated reservoir tank is pumped through a pipe placed through the center of the reactor core. The water in the reservoir can be heated to around 70 °C and pumped through the system. The pump has a variable flow rate and electronic control, so a number of scenarios can be developed. A slug of water can be pushed through the system to generate a reasonably fast change in reactivity. The pump speed can be increased or decreased over the course of a measurement, or even oscillated at different periods, to observe the response of the reactor.

To complement the experimental data generated by the RCF, a set of computational models have been developed. They are intended to illustrate the usefulness of the experimental data generated in validating coupled multi-physics simulations. A model previously constructed in MCNP has been used in conjunction with safety analysis at the RCF for many years and was modified to support the design of the heated water loop experiment. This model was translated into Serpent 2, a similar Monte Carlo code, which provided the added advantage of being able to produce collapsed cross-section data files for use in the deterministic neutron transport codes that are part of the SHARP toolset. A complete reconstruction of the RCF model, both with and without the heated fluid loop experiment, were necessary for use with the SHARP codes, PROTEUS (neutronics) and Nek5000 (thermal hydraulics), as these require an unstructured mesh format to operate. The computational requirements for fully coupled three dimensional simulations are significant, and challenges to full implementation and comparison with RCF experiments remain.

A wide variety of experiments have been conducted for eventual benchmarking and validation of multi-physics reactor codes. A complete database of detailed time- and temperature-dependent measurements have been generated and stored. Several examples of the experimental results are presented here, with
refinement and analysis of experiments ongoing. While these experiments still only address a finite range of operating conditions, in particular being limited to water in the sub-boiling regime, they represent a unique (to date) look at the feedback between temperature and neutronic behavior suited for validation of the coupling routines in multi-physics codes. It is hoped that these will be the foundation of future benchmarks that will provide confidence in the tools that will be used to develop the next generation of nuclear reactor systems.
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I. Introduction

The overall purpose of numerical simulation in nuclear reactor engineering is to achieve an accurate prediction of the physical phenomena and behavioral outcome of the reactor under normal (steady-state or quasi-steady-state) and/or transient/accident conditions. The complex nature of the physical processes in the reactor has led to the development of a suite of high-fidelity codes, each specializing in particular physical processes. The interdependence of these different processes has driven an effort to couple these individual simulations together into multiphysics codes. While most single-scale/single-physics codes have been extensively validated and benchmarked, the continuing development of multi-scale and multi-physics codes through different coupling techniques are actively performed, which calls for new experimental data to validate the coupling methodologies [1].

Reactor simulations involve multiple physical processes including neutronics, fluid flow and heat transfer (thermal-hydraulics), thermo-mechanics, fuel behavior, chemistry, and balance of plant - all with effects that feedback to each other. To provide realistic and accurate models of these feedback effects, multiphysics coupling is needed to connect core neutronics to other important physical phenomena, such as those mentioned above. In the qualification framework for a multiple-code platform, the processes of code verification, solution verification, model validation, and predictive estimation of system responses with uncertainty quantification typically are first applied to separate physical models/codes for different applications, and then combined together using appropriate methodologies for the individual models of different physical phenomena [2]. This extended verification procedure involves testing the assumptions in the physical models, the data exchange between different code systems, and their numerical coupling in order to capture the nonlinear interaction effects between the models.

To validate single and multi-physics codes, databases of benchmark quality experiments and simulations are publicly available, as the International Handbook of Evaluated Criticality Safety Benchmark Experiments [3], the International Handbook of Evaluated Reactor Physics Benchmark Experiments [4], and the VERA Core Physics Benchmark Progression Problem Specifications [5]. The multi-physics coupling in these codes requires experimental validation that is in addition to the validation data that exists for the individual physics routines, with benchmark experiments specifically designed for that purpose [1]. The available benchmark data as of today is not sufficient, as most of it are single-physics. Experiments involving neutronics and thermal-hydraulics feedback mechanisms have to be designed and characterized to be used for these modern codes validation.

The NEAMS toolkit couples computational codes addressing neutronics, thermal mechanics, fluid dynamics, structural mechanics, and system response with the purpose of enabling the design of future nuclear power plants and reactor cores [6]. The physics and data coupling between these tools are facilitated through a series of supporting elements (CouPÉ, SIGMA, MOOSE). The integration of these physics codes and the coupled feedback mechanisms between the models must be addressed through experimental validation.

The integrated SHARP Toolset, which includes neutronics (PROTEUS) [7], thermal fluid (Nek5000) [8] and structural mechanics (Diablo) [9], is being developed to address the multiphysics simulation challenges for nuclear reactor design and safety analysis [10]. Much effort has been made in the code verification. The gap for experimental validation of the codes, however, still remains. Therefore, experimental data for validation of single-physics and the bi-lateral combination of the three components
in the SHARP Toolset are of great need. This work seeks to address this gap by taking advantage of unique features of the RPI Reactor Critical Facility (RCF), a low power research reactor, to generate a database of coupled-physics experiments that could be useful for validation of multi-physics tools. Computational simulations of the facility are prepared using the SHARP tools to compare experimental values with computational performance.

Facility Description

The RCF is a critical assembly fueled in a standard core configuration of 332 or 333 UO₂ fuel pins, each containing a total of 35.2 grams of U-235 (4.81 % enriched) within stainless steel cladding, originally manufactured for the Special Power Excursion Reactor Test (SPERT) reactor experiments. Each fuel pin is 106.45 cm in length, and 1.18 cm in diameter, with an active fuel length of 91.44 cm. Prior to reactor measurements, each fuel pin used was surveyed for mechanical defects (bending or warping) that may contribute to uncertainty in core behavior. These fuel pins are arranged in a regular lattice with a pitch of 1.6256 cm, as seen in Figure 1. The reactor thermal power is typically limited to no more than 15 W, which permits fuel handling with minimal controls. A more complete description of the reactor geometry and kinetics parameters are given in the Safety Analysis Report [11].

The reactor tank volume is approximately 13000 L, and light water is used as moderator/reflective. As the low thermal output of the reactor does not induce any measurable temperature changes, the temperature effect is observed by applying two 18-kW electric heaters to the water moderator, with mechanical agitators to maintain uniformity in moderator temperature. The heaters placed inside the reactor tank are used to heat the moderator water, and its temperature is recorded by thermocouples at different locations in the tank. As the heating rate is low (about 4 °C increase per hour), thermal equilibrium between all core components in contact with the moderator is assumed, including fuel pins. A submerged agitator is also operated to maintain a uniform temperature throughout the moderator. A 5-Ci PuBe neutron source is used to start the reactor, and 4 boron-impregnated steel control rods are used for reactor control, as seen on Figure 2.

In the standard lattice configuration of the RCF, there is no neutronics to thermal-hydraulics coupling due to no measurable thermal power effects on the moderator. The moderator temperature increase is achieved with electric heaters which allow the study of the thermal-hydraulics to neutronics coupling effects in a one-way coupling as described previously. The RCF is a classically under-moderated assembly in the standard lattice configuration [12]. With increasing moderator temperature, its density is decreasing, which in turn further reduces moderating power, and finally reduces the total excess reactivity of the reactor. With the current heating capacities and the volume of moderator water to be heated during normal operation, the maximum moderator temperature achievable is about 43 °C. The minimal operation temperature is 10 °C (50 °F) for safety reasons [11].

The RCF is equipped with several measurement instruments. There are three type J thermocouples recording the moderator water temperature at different locations in the tank, and the arithmetic mean of these 3 measurements is used as the moderator temperature value. One other type J thermocouple is measuring the air temperature outside the core. The type J thermocouples are connected to type J extension wires and to a video graphic recorder (Thermo Westronics SV100). Two compensated and one uncompensated boron-lined ionization chambers (LP1, LP2 for Linear Power 1 and 2 and PP2 for Period Power 2) measure current induced by the neutron flux, and are processed by picoammeters. (model RPI-
1718 and RPI-1717, custom designed by Circuit Equipment Corporation). The power supplies are ORTEC 710. The position and dimensions of the measurement instruments are shown in Figure 3 Figure 3 and Figure 4. All three neutron flux channels are displayed on graphic recorders in the control room and saved to a flat file with a 1 Hz sampling rate. The control rod position is recorded by optical encoders mechanically linked to the rod drive mechanism. The position of the 4 control rods is displayed in the control room, with a 0.01-inch resolution.

![Figure 1: Top view of the core in the standard configuration.](image1)

![Figure 2: RCF core general view](image2)
Figure 3: Position of the measurement instruments in the core (top view)

Figure 4: Position of the measurement instruments in the core (side view)
Report Organization

The following sections provide a summary of the experiments performed at the RCF, and the corresponding simulations that are used for comparison. Section III describes a set of experiments that were completed using the standard or previously operating configuration of the RCF reactor core. Section IV describes the design and implementation of an experimental configuration of the RCF consisting of a heated water loop, and several experimental measurements performed with it. Section V describes a set of computational simulations of the RCF experiments using various reactor analysis codes. The report concludes with a summary, suggestions for future work, and other assorted project information.

II. Standard Reactor Core Experiments

Positive Period and Excess Reactivity Measurements

The neutronic behavior of the reactor is characterized by the degree to which the fission chain reaction may be super-critical (or sub-critical, in the case of negative excess reactivity) in relation to a reference condition. A measurement of the reactor power/neutron flux rate of change without any included control mechanism (in the case of the RCF, when the control rods are fully withdrawn), provides what is referred to as the total excess reactivity. For a constant and small insertion of reactivity in a reactor, an asymptotic transient of purely exponential form appears, i.e. the reactor period $T$ is constant, as described in equation (1) [13]. In this equation, $P$ represents power, proportional to the reactor flux.

$$P(t) = P_0 e^{\frac{t}{T}} \quad (1)$$

The excess reactivity value is then obtained from the Inhour equation [13], a solution of the six-group point kinetics equations, and the kinetics parameters for the RCF, as shown in equation (2) and Table 1. Using this structure of the Inhour equation, we consider the ratios $\beta_i/\beta$ and $\beta_{eff}/\beta_{eff}$ being equal, for all precursor groups $i$. The kinetic parameters $\beta$ (delayed neutron fraction for $\text{U}^{235}$), $\beta_i$ (delayed neutron fraction for $\text{U}^{235}$ and the $i^{th}$ precursor group) and $\lambda_i$ (delayed neutron decay constant for the $i^{th}$ precursor group) are selected from literature [14]. The errors given on $\beta_i/\beta$ provided by Keepin [14] are probable errors. $\beta_{eff}$ (effective delayed neutron fraction for the RCF) and $l^*$ (prompt neutron lifetime), are calculated by a SERPENT 2 V2.1.27 [15] simulation of the RCF reactor in its standard 332 pins configuration, using the adjoint Nauchi’s method.

$$\rho = \frac{t}{T} + \frac{\beta_{eff}}{\beta} \sum_{i=1}^{\gamma^6} \frac{\begin{array}{c} \beta_i \\ \lambda_i \end{array}}{1 + T \lambda_i} \quad (2)$$

<table>
<thead>
<tr>
<th>$l^*$ (s)</th>
<th>$\beta_{eff}$</th>
<th>Group #</th>
<th>$\beta_i/\beta$</th>
<th>$\lambda_i$ (s$^{-1}$)</th>
</tr>
</thead>
<tbody>
<tr>
<td>2.67805E-05 ± 0.000012</td>
<td>0.00793470 ± 0.000056</td>
<td>1</td>
<td>0.033 ± 0.004</td>
<td>0.0126 ± 0.0004</td>
</tr>
<tr>
<td>2</td>
<td>0.219 ± 0.013</td>
<td>0.0305 ± 0.0015</td>
<td></td>
<td></td>
</tr>
<tr>
<td>3</td>
<td>0.196 ± 0.033</td>
<td>0.111 ± 0.006</td>
<td></td>
<td></td>
</tr>
<tr>
<td>4</td>
<td>0.395 ± 0.016</td>
<td>0.301 ± 0.0016</td>
<td></td>
<td></td>
</tr>
<tr>
<td>5</td>
<td>0.115 ± 0.013</td>
<td>1.14 ± 0.22</td>
<td></td>
<td></td>
</tr>
<tr>
<td>6</td>
<td>0.042 ± 0.012</td>
<td>3.01 ± 0.43</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
It is assumed that equation (1) remains valid locally for experiments where the moderator temperature is changing slowly due to heating. As the rate of water heating is low (about 4 °C per hour), the change in moderator temperature during even the longest-lived delayed neutron group lifetime is small enough to be ignored (on the order of the measurement precision).

The facility allows different core configuration to be tested, thereby characterizing the reactivity worth of those individual changes. As an example, excess reactivity of several reactor configurations was measured. The results of these reactivity measurements are shown in Table 2.

<table>
<thead>
<tr>
<th>Core configuration</th>
<th>Reactivity (cents)</th>
</tr>
</thead>
<tbody>
<tr>
<td>333 pins</td>
<td>20.36 ± 0.11</td>
</tr>
<tr>
<td>332 pins, -1 peripheral pin</td>
<td>8.60 ± 0.05</td>
</tr>
<tr>
<td>331 pins, -1 peripheral and -1 central pin</td>
<td>22.11 ± 0.13</td>
</tr>
<tr>
<td>332 pins, -1 central pin</td>
<td>32.58 ± 0.18</td>
</tr>
</tbody>
</table>

A few observations are made from this data. Removing peripheral pins effectively decreases the total excess reactivity of the core. This is due to the decreased fuel inventory in the reactor. Removing the central pin causes an increase in reactivity. Under the standard configuration, the core is “water-starved,” or under-moderated as compared to the optimal moderator to fuel ratio.

**Isothermal Coefficient of Reactivity**

The isothermal coefficient of reactivity is calculated as a function of the moderator temperature, for 332-pin (16 to 36 °C) and 333-pin (30 to 45 °C) standard core configurations with measurements at a series of static moderator temperatures. In the 332-pin configuration, a fuel pin is removed from the periphery of the lattice. To perform this set of measurements, the immersion heaters are turned on and moderator temperature increases until a desired temperature is reached. The heaters are then disengaged, and since the only heat losses are due to the ambient environment, the temperature remains stable for the duration of the measurement. The control rods are then fully removed, and the reactor period measured to determine the corresponding excess reactivity. The temperature changes being very small over the course of a measurement (<2 %), this set of experiments is considered steady-state. The heaters are reengaged, the moderator temperature rises to the next desired set point, and measurement procedure is repeated. This experiment shows the influence of the moderator temperature on the reactivity and allows the calculation of the isothermal coefficient of reactivity ($\alpha_T$) for a given reactor configuration.

Figure 5 Error! Reference source not found. shows the results of reactivity measurements with increasing moderator temperature and its associated uncertainty, for both core configurations. As expected, the reactivity is decreasing with moderator temperature increase (i.e. $\alpha_T<0$). This is primarily due to the decrease in water density, reducing the moderation of the neutron flux. This further demonstrates the under-moderation of the reactor in this configuration. Figure 6 shows the fitted reactivity evolution with time, using a degree 2 polynomial and a weighted least squares iterative approach, for both configurations. Figure 7 shows the corresponding reactivity derivative over temperature (i.e. $\alpha_T$) for both configurations.
Figure 5: Reactivity measurements at various temperatures for 332 and 333 pins configurations.

Figure 6: Evolution of the reactivity and fitted reactivity with temperature for 332 (left) and 333 pins (right) configurations.
Transient Temperature Reactivity Feedback

An additional set of experiments was performed using the standard reactor core to demonstrate temperature feedback effects from heating of the reactor moderator. The control rods are set at a height such that the reactor is slightly super-critical or has a small excess reactivity (thus the neutron flux measurements values are increasing slowly with time). Typical starting conditions for the measurements were about 1 cent of excess reactivity. The electric heaters are engaged, resulting in a temperature increase of about 4 °C per hour. As moderator temperature increases in this configuration, the influence of the negative isothermal coefficient of reactivity is shown, and the reactor period is observed to decrease until the reactor becomes critical, and then becomes negative resulting in neutron flux decrease. It is possible to define a parameter called the critical time $t_{\text{crit}}$, the time from an arbitrary start of an experimental measurement at which the reactor becomes exactly critical (i.e. the excess reactivity is equal to 0 cents). Correspondingly, the critical temperature $T_{\text{crit}}$ is defined as the moderator temperature at the critical time. This experiment is repeated for various initial moderator temperature values, for similar observations of reactor change of state.

Figure 8 and Figure 9, show the change in neutron flux (as a measure of fission rate or reactor power) and increasing moderator temperature, each as a function of measurement time, for 2 cases. As observed, the reactor begins super-critical, and loses reactivity as the moderator temperature increases. Figure 10 and Figure 11 transform the neutron flux measurements into a reactivity value (with an associate uncertainty band). For case 1, reactivity is slightly positive until around
$t = t_{\text{crit}} = 764 \pm 151 \text{ s}$, where excess reactivity changes sign and then becomes negative. The corresponding critical temperature, $T_{\text{crit}}$, in this configuration is $33.21 \pm 0.33 \, ^{\circ}\text{C}$. Due to the slow rate of temperature change, the critical time has a fairly large uncertainty, but the corresponding uncertainty in the critical temperature is reasonably small. For case 2, $t_{\text{crit}} = 568 \pm 135 \text{ s}$ and $T_{\text{crit}} = 37.11 \pm 0.33 ^{\circ}\text{C}$.

Figure 8: Neutron flux and temperature with 95% uncertainty band vs. time, case 1.

Figure 9: Neutron flux and temperature with 95% uncertainty band vs. time, case 2.
Figure 10: Evolution of the temperature and reactivity with time, case 1.

Figure 11: Evolution of the temperature and reactivity with time, case 2.
III. Heated Fluid Loop Experiments

While the temperature feedback experiments using the standard reactor core provide an interesting coupled physics case, the experiments are necessarily slow and cover a limited temperature range. To expand the capabilities and generate a more diverse data set, an experimental configuration has been designed and commissioned, consisting of a heated water loop placed in the center of the RCF fuel lattice. The low power and flexibly defined safety constraints of the RCF make such a radical redesign of the reactor core possible.

Design and Implementation of Experimental Apparatus

The addition of this loop to the core provides a reduced volume where moderator temperature can be controlled, which increases the moderator temperature range that can be achieved and increases the speed at which the temperature change can occur. This configuration also results in overall changes to the behavior of the RCF operating conditions, which must be characterized. The design of this loop considered both neutronics and thermal hydraulics aspects, as well as practical functional requirements. The ultimate constraints on the design are the technical specifications incorporated in the RCF, operating license, with the key requirement being a maximum limit of 60 cents excess reactivity under any condition [11].

The feasibility of the experimental design was assessed through a scoping study using a previously established MCNP model of the RCF to simulate the reactivity worth of a water-filled pipe fill at the center of the core. The changes in water temperature were accounted for with adjustments to water density and cross-section library settings, including $S(\alpha,\beta)$ treatment. Since the purpose of these simulations was scoping and feasibility, the geometry is simplified, without considering water feed and return. A limiting factor in the experimental dimensions is that the apparatus was required to fit within the confines of an insert in the reactor top plate, the equivalent size of a 5 by 5 fuel pins grid (8.128 cm). For the following calculations, a steel cylinder of thickness 0.2 cm and height 171.77 cm, filled with water is modeled. The results of the different simulations are summarized in Table 3.

<table>
<thead>
<tr>
<th>Core configuration (moderator temperature 20 °C)</th>
<th>Test section water temperature (°C)</th>
<th>$k_{eff}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reference core - 333 pins (Figure 12)</td>
<td>-</td>
<td>1.00711 ± 0.00019</td>
</tr>
<tr>
<td>Test section inserted - 308 pins (Figure 13)</td>
<td>20</td>
<td>0.95264 ± 0.00020</td>
</tr>
<tr>
<td>Test section inserted - 308 pins (Figure 14)</td>
<td>76.85</td>
<td>0.95661 ± 0.00020</td>
</tr>
</tbody>
</table>
Figure 12: Simulated lattice - 333 pins reference core with water at ambient temperature - top view

Figure 13: Simulated lattice - 308 pins test section inserted with water at ambient temperature – side view on the left, top view on the right

Figure 14: Simulated lattice - 308 pins test section inserted with water at 76.85 C – side view on the left, top view on the right
The first clear lesson of these scoping calculations is the dramatic effect of exchanging 25 fuel pins for a steel pipe and water column, with $k_{\text{eff}}$ decreasing by nearly 5500 pcm. As expected, a significant number of fuel pins must be added at the periphery to counteract this change. In fact, the size of the reactivity change suggests that there may need to be some fuel placed within the water loop, which is explored further below. Replacing the water column in the pipe with water at the upper design limit, an increase in $k_{\text{eff}}$ of about 400 pcm is observed. This change is large enough to suggest that it could be effectively observed with the reactor instrumentation during an experimental run. It is also observed that the addition of the steel pipe filled with water in the center results in an over-moderated region. In an over-moderated region, increasing the temperature results in increasing reactivity. In this configuration, the core is formed by 2 distinct regions, as the outer regions of the fuel lattice would still be under-moderated. This kind of 2-region core has also been studied in a separate work [16].

When hot water passes through the submerged test section, it loses temperature through heat transfer to the surrounding medium. The rate of heat loss from the top of the test section to the bottom is approximated through a simplified axial heat transfer model, as seen in Figure 15.

![Simple axial heat transfer model](image)

**Figure 15: Simple axial heat transfer model**

In this model, it is assumed that:

- Only conduction heat transfer between water and pipe is considered
- Surface heat transfer coefficient $h_1$ is assumed to be large
- $T_{\text{out,pipe}}$, the temperature at the outer boundary of the pipe is equal to $T_{\text{out}}$, and both are constants
- $C_p$ of water is constant over the temperature range (20-80 °C)
- The water inside the test section is uniform temperature in the radial direction

Several pipe materials, dimensions and pumps were considered during the design process. Stainless steel was selected as the design material, which produced some challenges in implementation of the experiment. Stainless steel is a not insignificant neutron absorber, so the fuel loading needs are exacerbated. The heat conduction through the pipe wall is also large enough to predict a large temperature decrease along the axial length of the test section. The predicted temperature change along the length of the test section, as a function of the water flow rate for an assumed 60 °C inlet temperature, is shown in Figure 16.
Ultimately, the material was chosen for its mechanical properties, to withstand the design temperature range and pressures.

![Figure 16: Predicted axial temperature evolution with water flow-rate](image)

The RCF reactor top plate was previously modified to include a removable insert in the central 7x7 lattice locations. To install and secure the test section assembly in the core, a replacement insert has been fabricated. The central section of the insert holds the fluid loop assembly, and the edges consist of a perimeter row of lattice locations with the same pitch and diameter as the rest of the reactor core. A CAD representation and photograph of the top plate insert are shown in Figure 17. Ideally, the fluid loop would pass through the reactor core in a single direction. However, the bottom lattice plate of the RCF does not have the same removable insert as the top plate and modification of the bottom plate was considered infeasible. Therefore, the loop was designed to include a manifold at the base of the test section connecting to four small pipes for the loop return. The space for the return pipes in the lattice plate insert is evident in Figure 17.

![Figure 17: CAD drawing and photograph of top plate insert](image)
The final test section design consists of a 310 L reservoir tank connected by flexible hoses to the test section. A CAD model of the reservoir assembly and the test section are shown in Figure 18 (the connecting hoses are omitted). Photographs of the completed assemblies are shown in Figure 19 and Figure 20. Not shown is an expansion tank added to the assembly to accommodate changes in volume with temperature change. The expansion tank is open to atmospheric pressure, so pressurization within the reservoir tank does not occur. Also missing is a welded metal cradle in which the reservoir tank sits, which was not completed at the time of this photograph.

Figure 18: final CAD design of the test section elements

Figure 19: Test section assembly
The circulating pump is a variable speed pump capable of flow rates between 0.2 and 0.7 Liters/s, equivalent to a linear speed of water between 0.02 and 0.30 m/s. It is ethernet controlled, which allows for remote control from the RCF control room, as well as performing experiments with pump speeds that vary over the course of the experiment. The reservoir tank is equipped with two 8 kW immersion heaters. These heaters are wired to a pair of thermostats, each with a temperature probe, which will set the temperature point. The thermostat has a set range from 60 to 160 °F (up to 75 °C).

The test section is monitored from the reactor control room, with 13 type T thermocouples and one pressure transducer. There are 4 sets of 3 thermocouples at varying radial depths and axial position on the inlet part of the central pipe, and one thermocouple on the outlet part of the loop, as described on Figure 21.

At each of the 4 axial positions of the inlet section, each set of 3 thermocouples are inserted at 3 different depths, to measure a radial water temperature from the pipe edge to the center, as shown on Figure 22. A pressure transducer is also installed on the inlet leg of the loop at the top of the central pipe, used to measure an equivalent pressure applied by the water column and the flowrate on the internal pipe edge. In addition, there are 2 thermocouples inserted inside the loop water tank to monitor the reservoir temperature while water is being heated.

Figure 20: Water reservoir and pump assembly
Scoping calculations with MCNP [17] demonstrated that criticality could not be achieved only by adding fuel elements to the periphery of the core. To provide sufficient reactivity, a 2x2 assembly of fuel
pins was inserted within the large-diameter pipe. The pins are held in place by an assembly (shown in Figure 23) such that the pins are on the same pitch as the fuel in the rest of the reactor, although they sit slightly offset in the axial direction. A cross-sectional view of the completed assembly as inserted into the RCF core generated from MCNP VisEd is shown in Figure 24. A photograph of the assembly installed in the RCF is shown in Figure 25.

Figure 23: Pin assembly holder within large-diameter pipe

Figure 24: Cross-section of the RCF core with the experimental apparatus inserted
The installation of the test section was completed and fuel was loaded in a 1/M approach to critical procedure. It was found that more fuel was required for critical than had been predicted, as shown by Figure 26.

For the measured data, the positive excess reactivity values were determined by a measurement of the positive period of the reactor. The core load with 400 pins appeared to be exactly critical, and so the value of zero cents excess reactivity is assumed. The values for the subcritical configurations were extrapolated
from the subcritical multiplication measurements, assuming that the pin added above 400 pins had the same
worth as the last pin added to get to 400 pins. Disregarding the subcritical measurements for 392 and 394
pins due to reasonably poor counting statistics, the measured excess reactivity is below the predicted value
by an average of $26.3 \pm 2.1$ cents. This suggests the composition of the steel in the test section differs more
from nominal than was considered in the simulation.

**Test Section Experiments**

**Temperature Profile**

In one set of experiments performed with the test section, varying inlet temperatures and flow speeds
are selected, resulting in different temperature profiles along the axial length. Here the purpose is not to
observe temperature feedback effects, but to characterize the temperature profile effects under effectively
steady-state conditions. The loop reservoir water is heated until the desired temperature is reached, and the
reactor control rods are set at the critical height for the bulk moderator temperature (i.e. power is constant
with zero excess reactivity). The pump is then activated at the desired flowrate. Due to the over-moderated
conditions near the test section, the increased temperature in the loop results in an insertion of positive
excess reactivity. Outside the test section, the submerged agitators are operated as noted above to maintain
uniformity in the moderator temperature. The heat-transfer through the test section results in a gradual
increase in the moderator temperature, but the magnitude is small over the course of a single measurements
(less than $0.5 \, ^\circ\text{C}$ per minute). Data in which moderator and inlet temperatures are effectively constant is
isolated for analysis. A similar experiment is also performed, where the moderator water is heated up, and
cooler water is circulated through the test section, resulting in a different set of reactivity effects.

As heated water circulates through the loop, heat transfer through the loop wall to the surrounding
reactor water results in a drop in temperature along the test section. The magnitude of the temperature drop
depends on the loop water inlet temperature, the loop water flow-rate, and the moderator temperature.
Considering radial symmetry, the 3 radial temperature measurements are plotted at 4 different axial regions,
shown as blocks rather than interpolating the temperature profile between them. To get a clearer spatial
representation with the moderator water level and the fuel regions, a comparison between the lowest and
highest flow-rates possible ($0.22 \, \text{L/s}$ and $0.75 \, \text{L/s}$) are shown in Figure 27 and Figure 28.
As predicted, the axial temperature is decreasing by heat transfer to the surrounding moderator water. However, over the course of the axial length, a significant change is seen in the radial distribution of temperature. There is very little radial temperature gradient when the test section is outside of the moderator (small heat transfer rate to the surrounding air). Over the length within the moderator, a significant radial temperature gradient is observed, reaching more than a 30 °C difference between center and edge) as the loop water gets closer to the bottom of the test section. The flow-rate also has an influence on the temperature profile. A higher water flow-rate means a faster speed of the water through the loop, and less
heat transfer to the surrounding moderator water. A difference of about 5 °C average outlet water temperature is observed, for equivalent average inlet water temperature.

Figure 29 illustrates the measured effect of loop inlet temperature and loop flowrate on excess reactivity. In each case, the surrounding moderator temperature is about 20 °C. As loop temperature increases, reactor excess reactivity increases due to the over-moderation in the test section region. At higher pump flowrates, the delta T between inlet and outlet decreases (test section water remains hotter) and the reactor excess reactivity likewise increases. The effect becomes clearer as the difference between the test section and the surrounding moderator becomes larger.

![Figure 29: Reactor excess reactivity as a function of loop inlet temperature and pump flowrate.](image)

**Insertion of hot or cold slug into test section**

The configuration of these experiments is similar to those described above, but rather than analyzing the equilibrium conditions of a steady flow rate, the response of the reaction to a slug of heated water injected into the test section is observed. In this case, with the pump off, the water in the test section is permitted to come into thermal equilibrium with the surrounding moderator, while the water in the reservoir is heated to a desired temperature. The pump is then engaged at a set speed and the heated water begins moving through the test section, and a change in the reactor period results. A similar experiment can be completed, where the reactor moderator is heated and the reservoir remains unheated, resulting in a cold slug entering the test section when the pump is engaged. These are directly coupled neutronics and thermal
The process is repeated for various initial moderator and loop temperature values, as well as loop flow-rates.

A sample experimental run is provided for a cold slug experiment. The moderator is brought to a temperature of 37.7 °C, while the reservoir temperature remains at room ambient (24.8 °C). The reactor state is supercritical (ρ=15 cents). The pump is engaged, and the slug makes its way through the system. As can be seen in Figure 30, the reactor does not respond immediately to the pump, as it takes a period of time for the slug to reach the test section from the reservoir. As the slug traverses the test section, displacing the water in equilibrium with the surroundings, a complex reaction is observed. Finally, when a steady test section temperature is achieved, the reactor period stabilizes to a constant rate, different from the initial rate, equivalent to about 10 cents excess reactivity. These short transient during the slug flow may provide interesting feedback results for validation.

The effect of this transient is further illustrated in Figure 31. When the cold slug passes the inlet thermocouple, the difference between the inlet temperature becomes suddenly large. During this time, there is a steep decrease in excess reactivity. As the slug completes its journey through the test section, the temperature difference between the inlet and outlet reaches a steady value (representing the temperature gain while passing through the test section) and the excess reactivity stabilizes.

![Figure 30: Relative power and pump flowrate in a cold slug experiment](image-url)
In a corollary to the slug experiments, after the steady state reactivity worth is observed, the pump is disengaged and the water in the test section again becomes static. The heat transfer will continue, and the test section will return to thermal equilibrium with the surrounding moderator. Figure 32 shows an example of this behavior. Since the reactor was at critical before the pump was first engaged, it is expected that it will return to critical as the test section returns to equilibrium.

Figure 31: Reactivity and inlet-outlet temperature difference with time in a cold slug experiment.

Figure 32: Relative power and reactivity in return to equilibrium
Varying pump speed experiments

The pump selected for the test section assembly is capable of control over ethernet using a web interface. The provided control interface is fairly rudimentary, but because it uses standard communication protocols, a script was written in python that could provide much finer control over the pump speed, including allowing for smoothly varying the speed with time. Firstly, the pump either begins at a minimal flow rate that is linearly increased to the maximum flow rate (ramp-up), or begins at the maximum flow rate and is linearly decreased to the minimum (ramp-down). For these experiments, the reactor starts at critical, and the change in reactor period with changing flow rate is observed. Measurements are completed at various ramp-up and ramp-down periods. In an additional set of experiments, the pump speed is varied sinusoidally. For these experiments, the critical control rod height is estimated for the median flow rate, so that the reactor moves from sub-critical to super-critical as the flow oscillates.

Figure 33 demonstrates the response to a ramp-up procedure. The pump is set up to increase from minimal to maximal flowrate in 200s. The reactor is super-critical, and the power rate of increase accelerates as the pump speed increases. A dynamic increase of loop water flowrate decreases the inlet-outlet temperature gradient, increasing the reactor period with time which is equivalent to the reactor excess reactivity.

![Graph showing pump flowrate and relative power during a 200s ramp-up.](image)

*Figure 33: Relative power and pump flowrate during a 200s ramp-up.*
Figure 34 illustrates a similar reaction to the ramp-down experiment. The pump engages at maximum speed and decreases to minimal flowrate in 200s. The reactor becomes super-critical, and the period keeps decreasing as the pump speed decreases. A few seconds after the pump is stopped, the power becomes constant and the reactor becomes critical again.

**Figure 34: Relative power and pump flowrate during a 200s ramp-down.**

The oscillating pump speed case is illustrated in Figure 35. In this case, the oscillation is set to a 60s period. The power change is very small, but observable. The period of oscillation of the reactor power is the same as the pump, but shifted in phase.
IV. Reactor Simulations

To complement the experimental data generated by the RCF, a set of computational models have been developed. They are intended to illustrate the usefulness of the experimental data generated in validating coupled multi-physics simulations. Although fully coupled simulations have not yet been achieved, these models can form the basis for future validation work.

Monte Carlo Neutronics Simulations

The primary software used for modeling and simulating the behavior of the RCF during its operation has been the Monte Carlo N-Particle (MCNP) neutronics code. As shown previously, the system has been modeled for each of its configurations and used for safety analysis reports as well as student and reactor operator training courses. The MCNP model allows students and researchers to learn through both simulation and experiential learning how the reactor is to operate. Therefore, there is a great deal of experience in using MCNP for analysis of the RCF.

It is known that there is some bias in the MCNP simulations compared to measurements at the RCF. This is generally managed by defining reactivity calculations for the analysis are in reference to a $k_{crit}$ value for the model determined from a benchmark reactivity measurement. The value of $k_{crit}$ is determined from:

$$ \rho(\$) = \frac{k_{MCNP} - k_{crit}}{k_{MCNP} \cdot k_{crit} \cdot \beta_{eff}} $$

Using a 333 pin fuel load and a moderator temperature of 56 °F, the measured reactivity was found to be 0.2057. A simulation of the reactor under the same conditions results in a value for $k_{MCNP}$ of 1.00940 ± 0.00014, and therefore a $k_{crit}$ value of 1.00780. From there, the reactivity worth in $\$ \$ is calculated using this reference value of $k_{crit}$ by:
Δρ(\$) = \frac{k_{eff} - k_{crit}}{k_{eff} \cdot k_{crit} \cdot \beta_{eff}}

A parallel model to the MCNP model was generated for use in Serpent 2, a 3D Monte Carlo-based neutronics code similar in function to MCNP, is specifically made for nuclear reactor core simulation of neutronics behavior [15]. Most importantly, Serpent 2 allows for customization in geometry for use of creating collapsed multigroup cross sections for use in other deterministic neutronics codes. The macroscopic cross section generation feature is used as the basis for further simulation with the SHARP neutronics model, Proteus [18]. Images of the RCF geometry as output by Serpent 2 for a standard 333 pin configuration are shown in Figure 36.

![Images of RCF reactor from Serpent 2. Left: side view. Right: top-down view.](image)

**Figure 36: Images of RCF reactor from Serpent 2. Left: side view. Right: top-down view.**

**SHARP Toolset Code Modeling**

The central goal of the modeling tasks is to generate models within the SHARP tools, specifically PROTEUS (neutronics) and Nek5000 (thermal-hydraulics), which could be used to perform coupled physics simulations. Unlike MCNP and Serpent 2 which both model reactor geometry using constructed solid geometries (CSG), PROTEUS requires a 2D or 3D mesh for the neutron transport solvers. These conformal meshes must be provided from an outside software package, such as Cubit [19] or its academic and industrial version, Trelis [20]. PROTEUS can use first-order, second-order, or higher-order elements for its solution package, although it is much more suited for first-order elements (particularly where reactor geometry is highly heterogeneous).

The model of the RCF was built from components using the Trelis software. The simplest geometry component developed was the single pin cell, which was generated in both 2D and 3D models. Figure 37
illustrates the structure of the 3D mesh generated for the single pin cell. The single pin cell is useful for understanding the performance of the model with minimal geometric uncertainties. An example of the PROTEUS output of the thermal neutron flux and fast neutron flux is shown in Figure 38.

A 3D solid geometry of the RCF reactor was further developed using Cubit 15.0 for full-core simulations. The structural model is illustrated in Figure 39. A calculation using the core model in 2D with one-fourth symmetry is shown in Figure 40, with plots of the thermal neutron flux and fast neutron flux presented. Three-dimensional simulations provide resultant distributions in both the axial and radial dimensions, as can be seen in Figure 41 and Figure 42.
Figure 39: Image of three-dimensional RCF reactor geometry as constructed in Cubit 15.0.

Figure 40: 2D RCF reactor with 333 pins and control rods. Left: geometry and mesh as produced by Cubit 15.0. Middle: thermal neutron flux from Proteus-SN. Right: fast neutron flux from Proteus-SN.
Figure 41: Results of 3D Proteus-SN calculations. Left: thermal neutron flux Right: fast neutron flux

Figure 42: Results of 3D Proteus-SN calculations. Left: neutron absorption rates Right: nu-fission rates
The Cubit model of the RCF has been modified to include the heated water loop experiment. Figure 43 shows the full view of the RCF with the experiment installed, and Figure 44 shows the detail of the central pipe of the test section with the 2x2 fuel array installed. This model was developed to implement the coupled-physics simulations in SHARP. Significant obstacles to completing the mesh without errors were encountered, and numerous webcuts were required to define the different regions. The model exceeded the computing resource available and coupled simulations have not yet been completed. Development of the test section model is ongoing.

Figure 43: Trelis model of the RCF core with fluid loop inserted

Figure 44: Detail of fluid loop model mesh
Comparisons Between Codes

With parallel models constructed in MCNP, Serpent, and Proteus, quantitative inter-comparisons between them can be made. Specifically, the calculated eigenvalues for models under the same conditions are compared.

The pin cell model was simulated in Serpent 2 as a two-dimensional geometry for comparison of eigenvalues, along with a representative pin in MCNP6 with all reflective boundary conditions. MCNP6 and Serpent 2 used over 100,000 particles per generation and at least 5,000 active cycles. Proteus-SN used a 2D mesh with 4813 vertices, 11 energy groups, and 40 angular moments (7 polar angles, 9 azimuthal angles) with the Legendre-Tchebychev cubature scheme. The eigenvalues are compared in Table 4.

<table>
<thead>
<tr>
<th>Table 4: Calculated eigenvalues for single RCF Pin</th>
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<tbody>
<tr>
<td>Eigenvalue ($k_{inf}$)</td>
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The results of this simulation show excellent agreement between MCNP and Serpent 2, within 6 pcm difference of one another, and a difference within 160 pcm of Proteus-SN. The possibility of the difference in eigenvalue solutions may be due to the difference in method of solution or the need to further refine the energy and spatial discretization of the solution in Proteus-SN.

The 2D full core simulations of the RCF have been performed for a two-dimensional plane with the control rods inserted. This has also been simulated in MCNP6 and Serpent 2 for comparison. There are two meshes that were used in Proteus-SN; one is a coarser mesh with a total of approximately 39 thousand vertices, and a finer mesh with approximately 59 thousand vertices. MCNP6 and Serpent 2 used over 100,000 particles per generation and at least 5000 active cycles. The Proteus-SN simulations used two 2D meshes of differing refinement with 39,142 vertices for the coarser mesh and 59,348 vertices for the finer mesh, 11 energy groups, and 48 angular moments (5 polar angles, 7 azimuthal angles) with the Legendre-Tchebychev cubature scheme. The outer boundary conditions on the four outer edges were set to reflective. This 2D simulation implies an axially infinite reactor core configuration, essentially eliminating neutron leakage. For this reason, the 2D results are for comparing between the neutronics codes and not directly against experimental values. The eigenvalue results are included in Table 5.

<table>
<thead>
<tr>
<th>Table 5: Table of simulation eigenvalues for RCF 2D full-core</th>
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</thead>
<tbody>
<tr>
<td>Eigenvalue ($k_{inf}$)</td>
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As with the 2D pin example, the 2D full core example shows excellent agreement between both Monte Carlo codes (MCNP and Serpent 2) with a larger difference in Proteus-SN for the coarse and finer meshes. The difference being less than 300 pcm for the finer mesh and less than 400 for the coarser mesh suggest that the Proteus-SN solution may not yet be fully resolved and an even finer mesh may give more accurate results.
Along with two-dimensional simulations of the RCF core, three-dimensional unstructured meshes were created and simulated in Proteus-SN. MCNP6 and Serpent 2 used over 100,000 particles per generation and at least 5,000 active cycles. Proteus-SN used 3D meshes of 147,778 vertices and 377,363 vertices for the coarser and finer meshes, respectively, 11 energy groups, and 120 angular moments (9 polar angles, 11 azimuthal angles) using the Legendre-Tchebychev cubature scheme. Once again, the eigenvalue results were compared against Serpent 2. The eigenvalue results are shown in Table 6.

Table 6: Table of eigenvalues comparing full-core 3D Proteus-SN

<table>
<thead>
<tr>
<th></th>
<th>MCNP 6.1</th>
<th>Serpent 2</th>
<th>Proteus-SN (Coarser Mesh)</th>
<th>Proteus-SN (Finer Mesh)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Eigenvalue ($k_{eff}$)</td>
<td>0.99922(3)</td>
<td>0.99982(3)</td>
<td>1.03594</td>
<td>1.03483</td>
</tr>
</tbody>
</table>

The eigenvalue differences between the full 3D core in Proteus-SN and Serpent and MCNP are approximately 3500 pcm, a significant amount. As with the 2D full core example, the higher scale models require a significant number of elements to provide an accurate solution. Due to the computational limitations at the time, a model numbering in the millions of elements that would be required to reach a satisfactory solution was not possible. The full 3D case as described in Table 6 was run on 300 CPUs and required 9 hours of computation time for the coarse mesh and 240 CPUs and over 15 hours for the finer mesh case. Increasing the mesh size to tenfold more vertices or higher would not be possible for this particular computer cluster due to memory and other constraints. With access to computational clusters at and significantly more memory and computational power, future simulations may be able to be performed with higher fidelity.

Comparisons with Experiments

The simulations noted in the previous section were all performed with the $S_N$ solver in PROTEUS (PROTEUS-SN). This is the traditional solver used, and is the one that was integrated with the other tools in the SHARP multi-physics routines. However, there is a different solver available in PROTEUS which uses the method of characteristics (PROTEUS-MOC). This solver has the advantage of being much faster, and with less constraints on computing resources, and can therefore be expected to be able to produce more fully converged results in reasonable time frames. However, this solver has not (to date) been integrated into the SHARP engine. Since the initial project direction was intended to parallel the multi-physics experimental development, this pathway was initially ignored.

The MOC solver is also limited in that it can only accept extruded geometries, which are 2D meshes projected in the direction perpendicular to the plane into a block. The block must therefore be axially homogeneous. Several blocks can then be stacked to construct an axially heterogenous model. An example of this process is shown in Figure 45.
Simulations with PROTEUS-MOC were compared with primary measurements of reactor power previously performed at the RCF. Local power was determined experimentally by attaching gold foils to the exterior surface of fuel pins at various locations, accumulating sufficient integrated power (minimum of 100 Ws) and measuring the decay of the activation products [21]. Of interest is the relative shape of the power distribution, so all values are normalized to the maximum power indicated. Figure 46 shows the distribution of power along the axial length of the center fuel pin. Figure 47 and Figure 48 represent the power distribution at the central plane of the fuel region, at various radial pin positions along the perpendicular axes (in line with the control rod positions) and along axis diagonal to those, respectively.
Figure 46: Normalized axial power profile from experimental data and PROTEUS-MOC simulation for RCF 333-pin configuration from the bottom to the top of the active fuel region.

Figure 47: Normalized radial power profile from experimental data and PROTEUS-MOC simulation for RCF 333-pin configuration at the fuel centerline in perpendicular direction.
As observed in Figure 46, the shape of the axial power distributions are very similar, although there appears to be a shift of a few cm, suggesting the model performs well, but that the location of the reference point for distance (the position of the bottom of the fuel) be confirmed. In the radial power distribution figures, the experimental point is marked at the center of the fuel pin, whereas in reality the measurement is taken around the circumference of the fuel pin, representing the flux at the surface. Considering the magnitude of the variation in power density seen across the diameter of the fuel pin, the experimental data tracks very well with the PROTEUS-MOC results.
V. Conclusions

This project has expanded the available data for use in validation of reactor simulation codes. In particular, experiments have been developed which include transient temperature changes that feed back reactivity effects that can be observed. Several temperature dependent effects were measured using the standard core configuration of the RCF. The slow transient induced by heating the moderator with electric immersion heaters when the reactor was very slightly supercritical resulted in a measurable change in excess reactivity, changing the reactor to a subcritical state.

To expand the temperature range and conditions that could be obtained, a new experimental setup designed and installed in the RCF. The apparatus consisted of a water loop circulating heated water from a reservoir tank through the center of the reactor. These experiments included the effects of local temperature changes, as well as faster temperature transients with the injection of a hot or cold (relative to the surroundings) slug of water. Based on the analysis thus far, these may be the most promising data for future applications.

The RCF standard core and the heated water loop experiments have been modeled using mesh generation tools for application in the SHARP toolset codes. The complexity of the geometric construction challenges the computational resources necessary for the convergence of solutions using the PROTEUS-SN code. The PROTEUS-MOC solver is more promising, but to date has not been integrated into the coupled multi-physics routines of the SHARP toolset.

Future Work

While the experiments performed under this project have produced data for validation of multi-physics simulations that has been otherwise lacking in the literature, there are still a number of limitations and areas where additional benchmark experiments will still be needed. A few areas where future investigation may be addressed are noted.

First, The experiment was conducted under atmospheric pressure, thus the system was limited to sub-boiling temperatures, and in practicality, the maximum inlet temperature is limited to about 70 °C. A similar experiment may be commissioned under pressure may further expand the temperature range over which feedback conditions could be measured. The feasibility of a pressurized system in an experimental facility would need to be determined.

Second, the artificial moderator temperature change due to heating in an external reservoir introduces a temperature feedback into the critical reactor system. However, this remains effectively a one-way coupling, with the thermal conditions affecting the neutronic behavior, but not vice-versa. One of the benefits of these benchmark experiments is the controlled nature of the temperature changes, which permit more precisely defined input conditions for multi-physics modelling. An alternative design of this kind of experiment could be to replace fuel elements with electric heating elements. The heat output of these elements could be controlled to “respond” to the changes in the reactivity rate through the rest of the reactor to simulation the neutronic-> thermal-hydraulic feedback mechanism.

Third, the RPI Reactor Critical Facility is instrumented only with ex-core detectors. These provide adequate response to the overall condition of the reactor system. However, they do not provide a direct indication of radial or axial power distributions, nor can the propagation of power changes through the core
as moderator temperature shifts be measured. The test section experiments, particularly those involving the hot and cold slugs and oscillating flow rates, could have benefitted from in-core detectors in the vicinity of the test section that could respond to local power changes.

Fourth, an interesting problem where temperature effects on neutronics are important in the low-temperature, low-pressure regime are for subcritical systems, such as found in shipping containers and dry-cask storage systems. Some further experimentation is likely necessary to extend the application of these benchmarks to subcritical systems.

Finally, there are computational uncertainties that have been identified gaps for multi-physics benchmarks. The characteristics and effects of different computational libraries must be more fully understood. Significant differences in the behavior of the system have been observed when using the ENDF vs JEFF nuclear data libraries. The libraries themselves have different sets of limitations, such as the availability and temperature range of $S(\alpha,\beta)$ scattering data. The effects of these differences should be more fully explored.

**Publications List**

**Peer Reviewed Publications**


**Conference Proceedings and Presentations**


Additional publications are in progress and will be forthcoming.
Purpose: The purpose of this project is to develop a set of benchmark validation experiments for multiphysics coupling by taking advantage of the inherent flexibility of low-power reactor critical assembly. This project will use targeted experiments to validate the coupling between neutronics, thermal hydraulics, and structural mechanics routines present in the NEAMS Reactor Product Line toolkit.

Objectives:
• Construction and validation of model of experimental facility in SHARP toolkit codes
• Demonstration of neutronic-thermal hydraulic feedback coupling mechanisms with temperature and void, using a uniform reactor core setup
• Design and construction of an experimental test section to increase experimental benchmark capabilities
• The design of multi-physics coupling benchmark experiments using reactor feedback data from local perturbations

Logical Path:
• The model of the reactor facility is implemented in the SHARP toolkit codes
• The computational model is validated against a set of standard measurements of reactor physics at the facility
• An experimental apparatus is constructed to extend the capability to introduce localized perturbations in the reactor
• Benchmark data from experiments is generated
• Analysis of uncertainty and uncertainty quantification for propagation of errors through multi-physics simulations is performed

Outcomes: The project will provide evaluated data providing for comparison of multiphysics experimental datasets with toolkit simulations, an analysis of uncertainty and uncertainty quantification for propagation of errors through multi-physics simulations, and the identification of gaps in available data and definition data requirements for future experiments.

Details:
Principal Investigator: Peter F. Caracappa
Institution: Rensselaer Polytechnic Institute
Collaborators: Wei Ji (RPI), Michael Z. Podowski (RPI), Chang-ho Lee (ANL), Vijay Mahadevan (ANL)
Duration: 3 years Total Funding Level: $799,993
TPOC: Justin Thomas
Federal Manager: Dan Funk
Workscope: NEAMS-1
PICSNE Workpackage #: 15-1801

Results:
• Single Pin and 2D Partial Core, 3D core eigenvalue results compare closely with results in Serpent and MCNP6
• Moderator temperature/reactivity curves are generated for 332-pin core (60 to 95 °F) and 333-pin core (85 to 115 °F)
• PROTEUS simulations of uniform-core experiments and impact of mesh quality and simulation parameters have been characterized
• Several sets of test section benchmark data, as function of inlet temperature and flow speed have been generated
• Data organized into accessible database for publishing and use

Accomplishments:
• Experimental data for static moderator temperature/reactivity measurements collected
• Dynamic coupled moderator temperature/reactivity experiments completed
• Installation and validation of test section is complete – now have a fully-functioning temperature-heterogenous critical assembly, a new and unique capability
• Extensive sets of coupled test section benchmark data completed
• Temperature characterization of test section fluid flow completed
VI. References


