VALIDATION OF REACTOR PHYSICS-THERMAL HYDRAULICS COUPLED CALCULATION IN WATER-COOLED RESEARCH REACTORS WITH LAMINAR FLOW REGIMES

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ABSTRACT

A collaboration between the University of Florida and the Swiss Federal Institute of Technology, Lausanne (EPFL) has been formed to develop and validate detailed coupled multiphysics models of the zero-power (100 W) CROCUS reactor at EPFL and the 100 kW University of Florida Training Reactor, for the comprehensive analysis of the reactor behavior under transient (neutronic or thermal-hydraulic induced) conditions. These two reactors differ significantly in the core design and thermal power output, but share unique heat transfer and flow characteristics. They are characterized by single-phase laminar water flow at near-atmospheric pressures in complex geometries with the possibility of mechanically entrained air bubbles. Validation experiments will be designed to expand the validation domain of these existing models, computational codes and techniques. In this process, emphasis will be placed on validation of the coupled models developed to gain confidence in their applicability for safety analysis. EPFL is responsible for the design and implementation of transient experiments to generate a database of reactor parameters (flow distribution, power profile, and power evolution) to be used to validate against code predictions. The transient experiments performed at EPFL will be simulated on the basis of developed models for these tasks. Comparative analysis will be performed with SERPENT and MCNPX reference core models. UF focuses on the generation of the coupled neutron kinetics and thermal-hydraulic models, including implementation of a TRACE/PARCS reactor simulator model, a PARET model, and development of full-field computational fluid dynamics models (using OpenFOAM) for refined thermal-hydraulics physics treatments. In this subtask of the project, the aim is to verify by means of CFD the validity of TRACE predictions for near-atmospheric pressure water flow in the presence of mechanically entrained air bubbles. The scientific understanding of these multiphysics domains will be expanded and the validation base of commonly-used calculation methods will be expanded to cover a new range of research reactor types. From a practical perspective, CROCUS and the UFTR will have fully validated reactor dynamic and transient models for dynamic and accident analysis. With these validated models, both facilities will have improved capabilities and flexibility for extended operations in the future. CROCUS and the UFTR will be able to make future reactor modifications with reduced regulatory resistance. A feasibility analysis of future power uprates at

these facilities will also result.

1. Introduction

Reactor transient behavior analyses are necessary for the safety assessment, and ultimately operation, of nuclear reactors. Unfortunately, experiments that involve reactor behavior under transient conditions are often cost-prohibitive, dangerous, or forbidden per the exploitation rules, in particular those involving severe accident conditions. As a result, computational codes were developed to simulate reactor transient behavior. These codes allowed reactors to be modeled and transient behavior predicted with some degree of accuracy. However, to ensure the accuracy and reliability of the computational codes, reactor transient experiments must be conducted to validate the computational models. For that purpose, a collaboration between the University of Florida (UF) and the Swiss Federal Institute of Technology in Lausanne (EPFL) will develop coupled multiphysics models of their research reactors using several benchmarked codes. The results of the models will then be compared to the results of experiments conducted under this collaboration in order to validate the applicability of the multiphysics models to the research reactors.

The two research reactors being modeled are the University of Florida Training Reactor (UFTR, 100 kW) and the zero-power (100 W) CROCUS reactor at EPFL. While the two reactors differ significantly in design and power output, they share a common trait in the coolant flow: single-phase water in laminar flow through complex geometries at near-atmospheric conditions with the possibility of mechanically entrained air bubbles. The entrained air bubbles may introduce turbulence in an otherwise laminar flow, questioning the accuracy of current heat transfer correlations and pressure drop models. Since this flow characteristic is so unique and virtually nonexistent in industry, there is no significant literature exploring the resulting heat transfer and fluid motion from said flow characteristic.

The 3D dynamic analysis of a nuclear system requires a specific methodology: the static and kinetic analysis of the neutronics of the core, the thermal-hydraulics evaluation of the core/system, and to solve the time dependent reactor equation with reactivity feedbacks. The latter implies the codes to be coupled. In order to model the complex phenomena taking place both at the neutronics and thermal-hydraulics sides, a specific coupling scheme is used between the individual codes in order to predict the time-dependent evolution of parameters such as core power, fuel or coolant temperatures. These parameters are of prime importance to characterize the safety of a nuclear system.

The static analysis of the neutronics of the core will be handled by MCNPX and SERPENT. MCNPX is a Monte Carlo based reactor physics simulator developed at Los Alamos National Labs. It has been extensively benchmarked and validated, leading to results that are commonly accepted by the scientific community. To offer models that serve as a comparison, SERPENT will also be used due to its practical approach to derive the macroscopic cross-sections needed for the neutron kinetics code. SERPENT is also a Monte Carlo based reactor physics and burnup calculation code developed at

VTT Technical Research Centre of Finland. While not as mature as MCNPX, SERPENT offers burnup capabilities that result in more reliable data than do MCNPX burnup capabilities.

Purdue Advanced Reactor Core Simulator (PARCS) will be used to model the reactor kinetics aspect of the core. It solves the time-dependent neutron diffusion equation in 1D and 3D. PARCS, coupled with TRACE, a thermal hydraulics code capable of analyzing the core behavior during transients and LOCAs, provide a fully coupled reactor transient analysis tool backed by the US Nuclear Regulatory Commission (NRC). PARET is a transient analysis code that was designed specifically for research and test reactors that use pin and plate type fuel, fuel configurations found in CROCUS and the UFTR, respectively. OpenFOAM is a free, open source computational fluid dynamics (CFD) software package. OpenFOAM will be used to develop full-field CFD models of both reactors for refined thermal-hydraulics physics treatments. Since it is unknown the effect the entrained air bubbles will have on the heat transfer and fluid motion nor the deviation in the results from TRACE or PARET from the experimental data due to the air bubbles, it is necessary to have a CFD model that accounts for the air bubbles. OpenFOAM will also be coupled to MCNPX in order to model the multiphysics in the core with high fidelity.

Fully coupled multiphysics models will be created for both the UFTR and CROCUS. However, all reactor transient experiments will be done on the CROCUS reactor ((Is it not thought to perform tests as well at UFTR ?)). Therefore, the experiments will validate the CROCUS models and will determine the transferability of those results to the UFTR and other subcooled research reactors. With validated reactor transient models, safety analyses required for future upgrades of the facilities can be reliably done.

2. UFTR and CROCUS Reactors

The UFTR is a research and training reactor at the University of Florida since May 1959. The reactor is an Argonaut type reactor, water-cooled using U_3Si_2-Al fuel enriched to 19.75 wt% in the shape of plates (MTR type). The reactor is currently licensed to deliver 100 kW of thermal power. The core (partial view) is shown in Fig. 1.

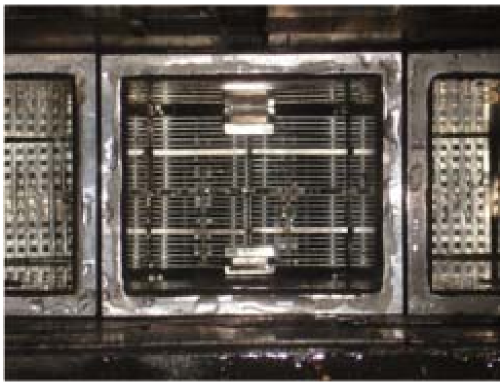

Fig. 1 UFTR Reactor element at UF

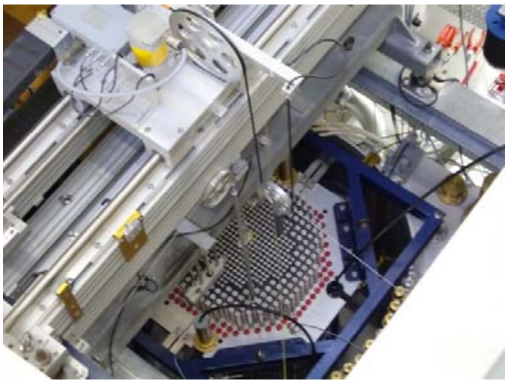

Fig. 2 CROCUS Reactor at EPFL

The CROCUS zero-power teaching and research reactor (Fig. 2) is a light-water moderated facility limited to a fission power of 100 W. The core is approximately 60 cm in diameter and 100 cm in height. First criticality was achieved in July 1983 and the reactor is in normal operation at low power since that date for teaching and research purposes. The core contains two zones of different fuel rods and consists of uranium ceramic fuel rods (with pure aluminum cladding) with an enrichment of 1.806 wt% ^{235}U and metallic uranium fuel rods with an enrichment of 0.947 wt%.

3. Thermal Hydraulic Phenomena of Interest for Validation

Neither CROCUS nor the UFTR have the closed-circuit piping typical of a PWR/BWR primary system. Instead, each admits a direct interface between the reactor primary coolant and air in the room. This gas-liquid interface allows for the introduction of a non-condensable gas (air) into the reactor piping system.

The potential for introduction of air is particularly strong in the UFTR. At the top of each fuel box, the water is simultaneously exposed to air and undergoing a 90 degree turn to begin its flow down the return pipes. To ensure safe operation, the reactor operates at a lower flow rate than the return pipes could support. As a result, the return pipes are not consistently completely flooded with water. This configuration allows for mechanical entrainment of air through these return pipes and into the flow system.

Flow disturbances (characterized as turbulence) have been noticed by UFTR operators in the fuel box nearest to the entrance pipes. Since the Reynolds number in the UFTR is approximately a factor of 2 too low to produce turbulent flow in single-phase, another explanation is required. Bubble-induced turbulence is proposed as the explanation for this phenomenon. A disproportionate fraction of entrained air bubbles can be expected flow through this first box, as it presents the first opportunity for the air bubbles to rise through the reactor core. The minimum bubble concentration to produce turbulence establishes a minimum for the level of air entrainment. Conversely, high concentrations of air bubbles would be expected to lead to pump cavitation and failure (which is not seen). This establishes a maximum concentration of air bubbles as well.

The introduction of a second, non-condensable component (phase) to the flow can change the heat transfer properties significantly. The effects of a second phase have been studied experimentally and theoretically for both boiling and condensing heat transfer. The familiar boiling correlations of Schrock and Grossman (1962) and of Chen (1966), among others, divide boiling heat transfer coefficients into two terms: one related to nucleation (expected to be zero for non-condensable vapor) and one related to single-phase convection, enhanced (or deteriorated) by the presence of the second phase. At extremely small vapor flow qualities, these correlations typically predict a reduction of turbulence and a deterioration of single-phase convection heat transfer.

Condensation heat transfer in the presence of a non-condensable has been studied in considerable detail. Typical results indicate a reduction in the heat transfer due to the non-condensable gas, due in part to its lower thermal conductivity. One formative paper (that of Minkowycz and Sparrow [1966]) predicts a reduction in heat transfer coefficient of up to 50 percent for an air mass fraction of just 0.5 percent.

One challenge to extending these correlations and models to the UFTR and CROCUS is the lack of single-phase turbulence in the research reactors. Studies of heat transfer in a nominally laminar flow, with the presence of a non-condensable, are quite rare. Since the literature is suggestive of a reduction in heat transfer (*i.e.;* reduction in margin to fuel failure, this is especially important for future power uprates), it is important to explicitly study the effects of non-condensable gas to confirm the UFTR and CROCUS safety bases.

In CROCUS, the water circulation through the fuel is quite peculiar. The water is pumped first in the core with a feeding pump from an auxiliary tank located outside the reactor shielding. The water goes through the core from the bottom side of an open vertical vessel and then overflows back in an open collector located on the top of the vessel but at the opposite side of the inlet and is mixed again outside the core in the open auxiliary tank. It means that the coolant does not flow solely with a vertical direction, but cross the fuel with a radial direction as well. During the mixing process, there is plenty of opportunity to have air bubbles inserted into the water.

4. Thermal Hydraulic and Neutronic Work in progress

Three major areas of work are being performed at this time.

a. *TRACE/PARCS and PARET Modeling*. Both CROCUS and the UFTR are modeled through conventional nuclear codes to address coupled thermal hydraulic and reactor physics modeling. Work focuses on nominally single-phase heat transfer (no local or bulk boiling) with non-condensable models activated for an assume concentration of non-condensable gas (air). Since the concentration of entrained air is itself an unknown, a parametric study is planned. At present, the modeling of the CROCUS reactor is under good progress for the static neutronic part, i.e. SERPENT and MCNPX.
b. *OpenFOAM Modeling*. Canonical nuclear thermal hydraulic codes (such as TRACE and PARET) are based primarily on experimental data, which are lacking for the

specific flow conditions in the UFTR and CROCUS. Full-field simulations, including non-condensable gas and heat transfer, will be performed in OpenFOAM to confirm or refute the applicability of the models in existing codes. The University of Florida team members have access to this software and a functioning bubble transport model, as recently presented by Owoeye and Schubring (2013). This model will be extended to the air-water case, including heat transfer, and apply to the reactor-specific geometries.

c. *Experimental Validation*. The EPFL team members will develop appropriate experimental techniques to measure pressure drop and heat transfer coefficients (through bulk and fuel pin temperature measurements) to verify the results from the OpenFOAM and thermal hydraulic code efforts. Appropriate modifications to the OpenFOAM simulations and TRACE/PARET models will be made based on the experimental results. At present, a PhD student has recently joined the EPFL team and will start by reviewing existing literature on experimental measurement techniques that could be used for the purpose. An additional challenge will be to measure the needed parameters for validation of the codes in a nuclear context, where adding instrumentation in the core will be to some extend considered by the Regulatory Body as a modification of the nuclear installation. Licensing questions and activities might slow down the project.

5. Conclusions

The UFTR and CROCUS reactors at the University of Florida and the Swiss Federal Institute of Technology in Lausanne are in the process of performing a new comprehensive safety analysis of their respective reactor. The ultimate goal of the approach taken is to renew the safety reports and have in the future the possibility to extend the usage of the reactors, possibly via power uprates. In practice, a collaboration between the two universities has been established to share the competence, experience and it allows for synergies in the development effort that will be needed.

For safety analyses, it is important to predict accurately the core behavior under transient conditions. The 3D dynamic analysis will be made using the following codes: SERPENT, MCNPX, PARET, PARCS, and TRACE. At present, the static and kinetics models of both reactors are under good progress. The major difficulty consists on validating the developed models. For this, specific experimental techniques will be developed and used at EPFL to obtain experimental data to be used to verify the code predictions. This will be the first step of the model validation.

Since both reactors are single-phase water-cooled at near atmospheric pressure conditions with a direct mixing of water and air at the core inlet, it is of particular importance to verify the heat transfer coefficients and pressure drop obtained with PARET and TRACE are adequate. It is good to recall that very little validation effort has been made in the past for the peculiar core geometries and operating coolant pressure of the UFTR and CROCUS reactors since most of commercial power reactors are working in turbulent regimes. To address this particularity, CFD modeling with the OpenFOAM software is currently underway at the University of Florida. In OpenFOAM, a new model for mechanically transported bubbles has been recently implemented and will be further extended to allow the modeling of geometries needed in UFTR and

CROCUS with water-air interfaces.

At EPFL, preparation work recently started to develop experimental techniques to verify the OpenFOAM predictions. This will be the second step of the model validation. By combining the results obtained by the thermal hydraulic codes and refined CFD treatment, it is expected to better understand the multiphysics phenomena occurring at the UFTR and CROCUS reactors.

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