Comparing the Steady State System Modeling Results of the Conceptual NIST Reactor with ANL-PARET and RELAP5-3D

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INTRODUCTION

A tank-in-pool type research reactor with an innovative horizontally split compact core was recently studied at National Institute of Standards and Technology (NIST) with the primary objective of delivering advanced neutron sources for scientific experiments [1]. The reactor concept considered a 20 MW thermal power and a 30-day operating cycle. A plate-type fuel element with low enriched uranium (LEU) - U_3Si_2-Al - was used. The core is cooled by a forced downward circulation of light water, and surrounded by a cylindrical heavy water tank, which is about 2.5 m diameter and 2.5 m height and placed in the center of a large light water pool that serves as thermal and biological shields. A 3-D cutaway view of the split core design is illustrated in Fig. 1, in which two vertical cold neutron source (CNS) assemblies were inserted to the center of the reactor from the top region. More detailed description of the core and reactor configurations of the new design can be found in Ref. 2-3.

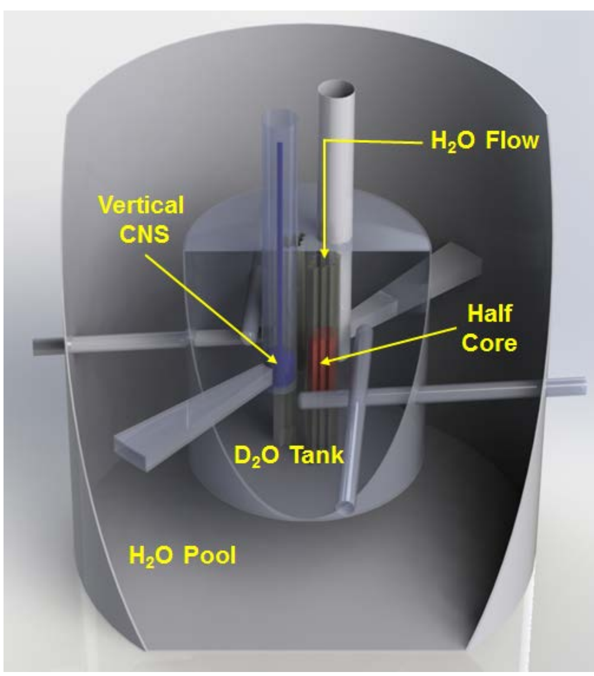

Fig. 1. A 3-D cutaway view of the reactor.

Preliminary neutronics and safety analyses have been performed for the proposed split core design to justify the physics feasibility of the neutron source capability of the design, and assess the thermal-hydraulics (T/H) safety features of the reactor [3]. In these studies, the neutronics calculations were carried out using the Monte Carlo code MCNP [4], and the T/H safety calculations were completed using the modular channel code PARET [5].

The PARET code is a computational T/H analyses tool developed by Argonne National Laboratory (ANL) with particular suitability for plate-type research reactor safety analyses. It consists of a one-dimensional (1-D) T/H model and a point-kinetics model to couple the neutronics and thermal hydrodynamics effects on reactor behavior during normal and off-normal conditions. A reactivity model is integrated in the code to provide proper thermal feedback from the T/H model to the neutronics model. PARET can be used to develop a multi-channel model to predict the transient behavior of the reactor during normal and off-normal design basis accidental scenarios. However, PARET is merely a channel analysis code and unable to model complete cooling loops in the reactor, and the limited accountability of the PARET results is generally recognized.

To overcome this computational modeling limit, research efforts have been extended to duplicate the system modeling results of PARET for the NIST new reactor using the more sophisticated system modeling code RELAP5-3D [6]. A nodalization of the core and other important components of the primary cooling system of the reactor, such as heat exchanger and primary loop pump, will be developed in the RELAP5-3D. A comparison study of the system behavior predicted by both codes will be carried out. The focus in the first stage of this study will be given to the steady-state performance. This summary presents the preliminary research outcome of the system performance characteristics of the reactor of RELAP5-3D, comparing against the previous results yielded from PARET predictions.

The rest of the summary is organized as follows. The modeling procedure and key parameters used in both computational models are described in the next section, followed with the presentation and discussion of the preliminary analysis results. A brief conclusion and future work will be provided at the end of the summary.

COMPUTATIONAL MODELS

The nodalization of the reactor core is established and developed with RELAP5-3D. The hydraulic components of the coolant flow channel including pipes and plenum are modeled by sets of single control volume and junctions. The entire coolant loop has not yet been completed, thus the boundary conditions to the core channel model are established through time-dependent control volumes and junctions. The heat structure components that counts the heat source from the reactor power is also established. The nodalization scheme of the reactor core is shown in Fig. 2.

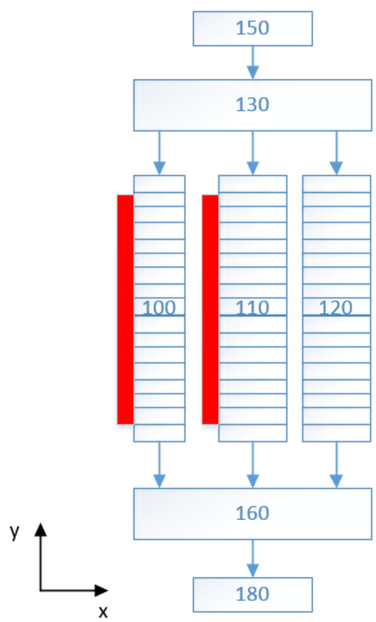

Fig. 2. Nodalization of the reactor core.

As shown in Fig. 2, the thermal-hydraulic system of the reactor core is represented by one hot channel (No.100), one average channel (No.110) and one bypass channel (No.120) using the built-in PIPE component in the RELAP5-3D code. The hot channel describes the flow channel with the hottest power peaking factor in the fuel assembly, the remaining flowing channels are lumped to one average channel. The bypass channel is developed to take into account the side flow that was stuck in the area between fuel assemblies. All three channels are divided into 17 control volumes along the flow direction. The upper plenum (No.130) and bottom plenum (No. 160) are modeled to connect and mix the flow at the entrance and exit point of the flow channels. The inlet condition (flow source) was provided using a time-dependent control volume (No.150) and its corresponding time dependent junction. Similarly, the outlet condition (flow sink) is defined by a single control volume (No.180) and the corresponding single junction.

The proposed NIST reactor core consists of 18 fuel assemblies, and each fuel assembly is composed of 17 MTR-type fuel plates. In RELAP5-3D, each fuel plate in a fuel assembly is lumped to one heat structure. To show more detailed temperature distribution in the fuel plate, the heat structure is divided into 15 volumes along the axial direction, and 10 mesh intervals along the transverse direction of the plate: two mesh intervals on each side of the cladding and six intervals in the middle for the fuel meat part. A nodalization scheme for the heat structure (i.e., the fuel plate) is shown in Fig. 3.

The key parameters of the core for RELAP5-3D input such as core material, fuel assembly geometry data, core thermal hydraulics, neutron kinetics, and boundary conditions are included in Table I.

Table I. Key Parameters for RELAP5-3D Model.

Materials	**Values**
Fuel meat material	U_3Si_2-Al
Fuel type	Plate type
Fuel density (g/cc)	6.53
Enrichment (wt%)	19.75
U-235 loading (g/plate)	391.47
Fuel assembly geometry	
Fuel assembly	18
Fuel plates per assembly	17
Aluminum plates	2
Fuel plate width (cm)	6.665
Fuel meat width (cm)	6.134
Fuel plate thickness (cm)	0.127
Fuel meat thickness (cm)	0.066
Cladding thickness (cm)	0.0305
Fuel plate length (cm)	60
Fuel meat length (cm)	67.28
Thermal-hydraulics	
Fuel thermal conductivity (W/m·K)	48
Cladding thermal conductivity (W/m·K)	180
Fuel volumetric heat capacity (J/m^3·K)	2.225E+6
Cladding volumetric heat capacity (J/m^3·K)	2.419E+6
Inlet coolant temperature (°C)	37
Core outlet pressure (kPa)	200
Total power (MW)	20
Inlet volumetric flow rate (gpm)	8000
Hydraulic diameter (cm)	0.56
Reactor kinetics	
Prompt neutron generation time (μs)	252.63
Effective delayed neutron fraction (β_{eff})	0.00718

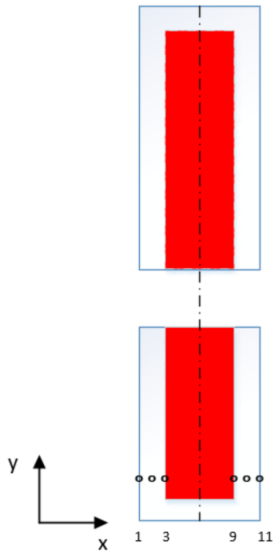

Fig. 3. Nodalization of the fuel plate.

The neutronics performance characteristics of the reactor were obtained with the Monte Carlo–based code MCNP6 [4]. A point reactor kinetics model is enabled in the model to count for the power variation with the reactivity coefficients and kinetics parameters provided by neutronics calculations. For the RELAP5-3D kinetic input, delayed neutron fraction and prompt neutron generation time are needed to calculate one of the important input β/Λ, the delayed neutron precursor information such as delayed neutron precursor yield ratio and delayed neutron decay constant for each delay group also necessary for the RELAP5-3D input. These kinetics parameters are provided in Table I. The axial power distributions for the hot and average channel (required by the RELAP input) for the start-up (SU) and end-of-cycle (EOC) core are obtained from the previous neutronics studies [2-3]. The SU core stands for the most reactive core status in the fuel cycle, thus the power peaking factor remains as most eminent.

RESULTS

Only the results for steady-state conditions at full power are available from RELAP-3D calculations at this moment. These results are compared against formerly published PARET results [7] to verify the correctness of the modeling procedure and outcome.

The temperature of fuel centerline, cladding surface temperature and coolant temperature in the hot channel and average channel of the SU and EOC core are presented in Fig. 4, 5, 6, and 7, respectively. As shown in the figures, the axial temperature distribution of the fuel, cladding and coolant predicted by both codes agree very well for both SU and EOC cores.

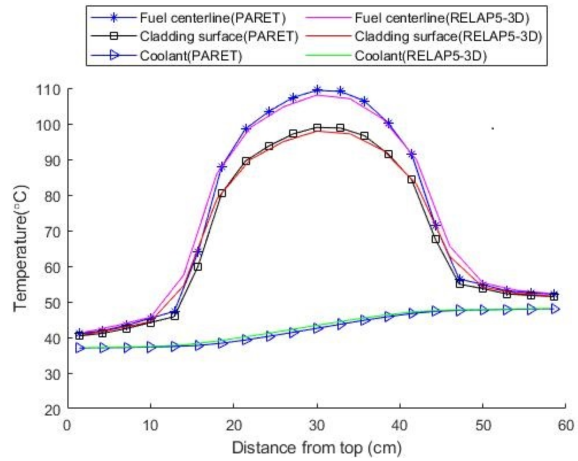

Fig. 4. Temperature in the hot channel (SU)

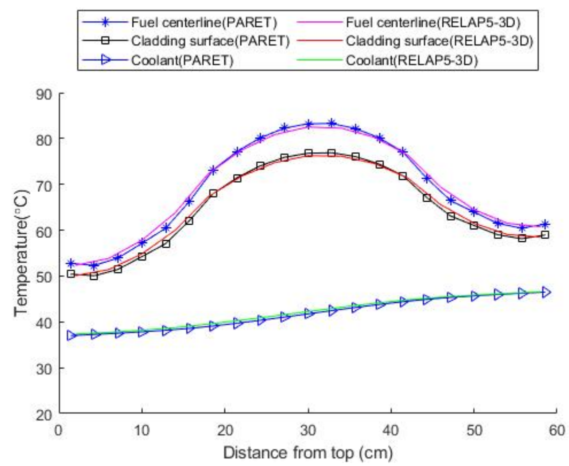

Fig. 5. Temperature in the average channel (SU)

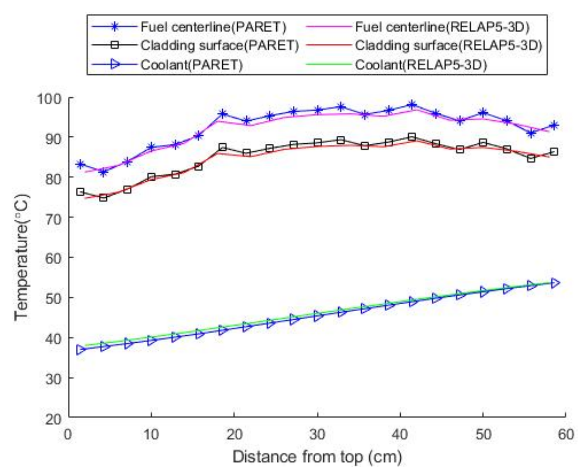

Fig. 6. Temperature in the hot channel (EOC)

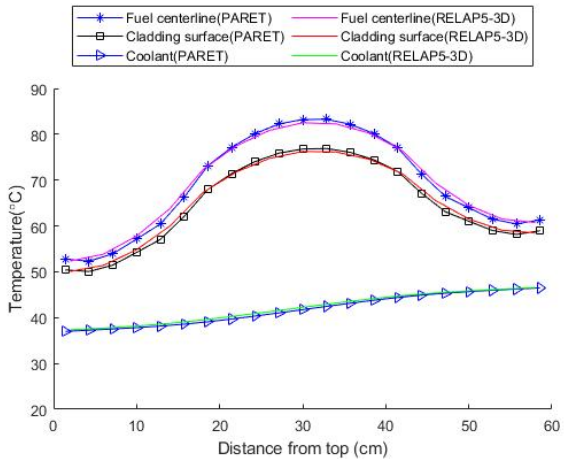

Fig. 7. Temperature in the average channel (EOC)

More quantitative comparison of the temperature predictions from both codes are summarized in Table II and Table III for SU and EOC cores, respectively. The maximum fuel centerline and cladding surface temperature in the hot channel of the SU core are 108.05°C and 97.91°C, respectively, while PARET predicts 109.38°C and 98.95°C for these two quantities. Coolant outlet temperature in RELAP5-3D is calculated as 48.11°C with respect to 48.01°C calculated by PARET. The similar agreement has been achieved for the EOC case.

Table II. Component Temperature Comparison in SU

°C	Hot Channel		Average Channel	
	RELAP5	PARET	RELAP5	PARET
T (Fuel)	108.05	109.38	82.52	83.27
T(Cladding)	97.91	98.95	76.25	76.90
T(Outlet)	48.11	48.01	46.56	46.49

Table III. Component Temperature Comparison in EOC

°C	Hot Channel		Average Channel	
	RELAP5	PARET	RELAP5	PARET
T (Fuel)	96.94	98.14	73.80	74.44
T(Cladding)	89.20	90.10	68.96	69.43
T(Outlet)	53.74	53.66	46.54	46.50

CONCLUSIONS

The RELAP5-3D model to simulate the thermal hydraulics properties of the conceptual NIST research reactor core was established and the steady state calculations were performed. Preliminary results produced by the RELAP5-3D have a good agreement with the ones from the PARET code, which verifies the validity of the current model in a certain degree.

In the next stage, some additional components in the primary cooling system of the reactor, such as heat exchanger and primary loop pump, will be developed in the RELAP5-3D model, and design basis accidental transient analyses will be performed using the RELAP5-3D models.

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