Hypothetical Accident Analyses on the Conceptual NIST Reactor with a Split Core Using RELAP5-3D

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ABSTRACT

This paper performs steady-state and transient analyses for the recently proposed NIST research reactor. The primary purpose of this work is to examine the RELAP5-3D models for the study of the thermal-hydraulics (T/H) safety characteristics of the new reactor configured with a horizontally split core. Previously, the multi-channel T/H safety analysis code ANL-PARET was employed to model the transient behavior of the reactor during normal and off-normal design basis accidental conditions. These analyses were restricted to the reactor core portion due to limited modeling features of the PARET code. This investigation extends the reactor safety analyses using the more widely known systematic safety analysis code – RELAP5-3D. A nodalization of the core consisting important heat structures and hydraulics components in the primary cooling system of the reactor is realized in the RELAP5-3D. Both steady state and reactivity insertion accidental transient calculations are performed. For verification purpose, the RELAP5-3D simulation results are compared to the previous PARET results. These comparisons show that the RELAP5-3D outcomes have a very good agreement with the ones from the PARET code, which verifies the feasibility of the current model in a certain degree.

KEYWORDS
Research Reactor, RELAP5-3D, Design Basis Safety Analyses

1. INTRODUCTION

A tank-in-pool type research reactor with a novelty horizontally split core arrangement was recently investigated at National Institute of Standards and Technology (NIST) with the primary design purpose of producing high-quality neutron sources for scientific experiments [1, 2]. The reactor concept considers 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 in the new design. The reactor core is cooled by a forced downward circulation of light water and surrounded by heavy water in a cylindrical tank. The reflector tank is about 2.5 m in diameter and 2.5 m in height and placed in the center of a larger light water pool that serves as thermal and biological shields. A three-dimensional (3-D) cutaway view of the split core design is illustrated in Fig. 1, in which two cold neutron source (CNS) assemblies are vertically inserted to the center of the reactor from the top region. A more detailed description of the core and reactor configurations of the new NIST reactor design can be found in Ref. 2.

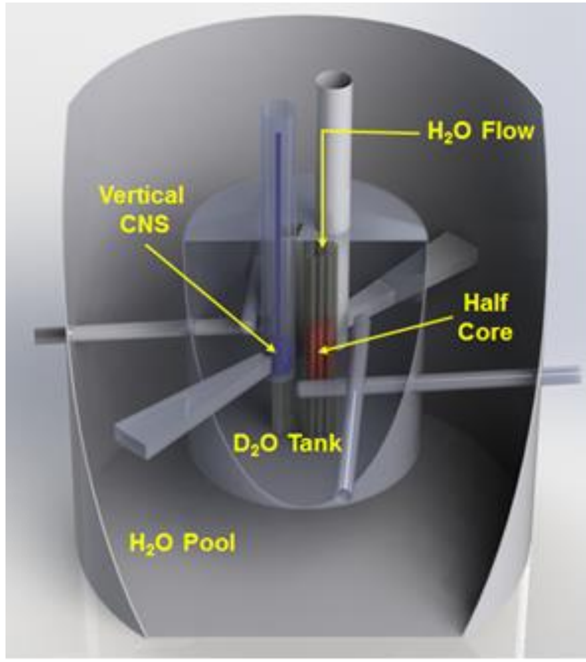


Figure 1. A 3-D cutaway view of the newly proposed NIST reactor [2].

Preliminary neutronics and safety analyses have been performed for the proposed split core design to justify the physical feasibility of the design and preliminarily assess the thermal-hydraulics (T/H) safety features of the reactor [2]. In these studies, the neutronics calculations were performed using the Monte Carlo code MCNP6 [3], and the T/H safety calculations were completed using the modular channel code PARET [4].

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. An integral reactivity model is integrated in the code to provide proper thermal feedback from the T/H model to the neutronics model. PARET can develop a multi-channel model to predict the transient behavior of the reactor in various design basis accidental scenarios. However, PARET is merely a channel code and unable to model full cooling loops in the reactor. Because of this, limited accountability of the PARET results is generally recognized.

To remove the computational modeling limitations of PARET, research efforts have been extended to analyze the NIST new reactor using the more sophisticated system modeling code RELAP5-3D. As part of these efforts, research focus was paid to reproduce the system modeling results of PARET in both steady-state and transient conditions. Nodalization of the core and other important components of the primary cooling system of the reactor are being developed in the RELAP5-3D. A close comparison study of the system behavior predicted by both codes are carried out. This paper presents the preliminary research outcome of the system performance characteristics of the reactor calculated with the RELAP5-3D code, comparing to the previous results predicted by the PARET [7].

The remainder of the paper is organized as follows. The modeling procedure and key parameters used in the computational models are described in Section 2, followed with the presentation and discussion of the preliminary analysis results. Some concluding remarks and future perspectives of this project will be offered at the end of the paper.

2. COMPUTATIONAL MODELS

Fig. 2(a) illustrates the nodalization of the reactor core in RELAP5-3D. The hydraulic components of the coolant flow channel including pipes and plena are modeled by sets of single control volume and junctions. The coolant loop has not yet been fully completed at this moment. Therefore, the core channel model is bounded with inlet and outlet components, which are established with time-dependent control volumes and junctions. Proper boundary conditions are provided with the ones consistent to the PARET model. The heat structure components are also developed to accommodate the proper heat power profiles in the core. A schematic scheme for the heat structure (i.e., the fuel plate) is shown in Fig. 2(b).

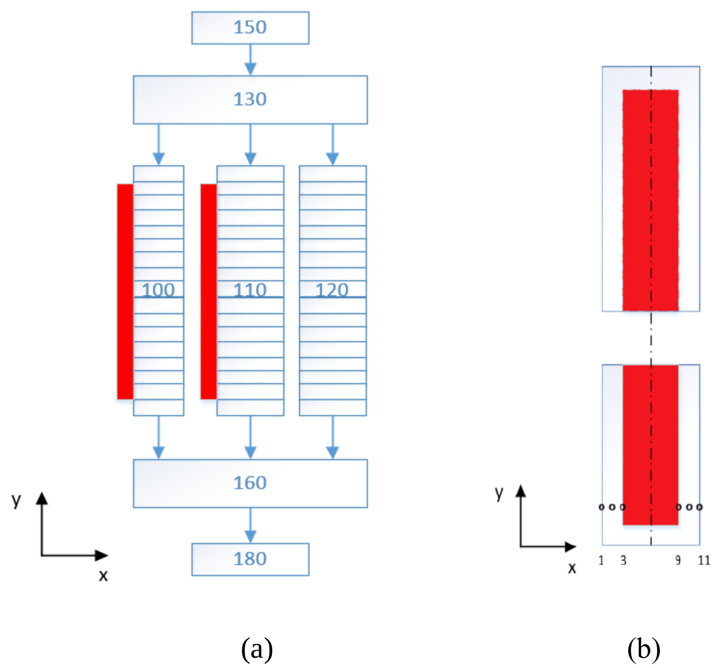


Fig. 2. Nodalization of the reactor core (a) and schematics of the fuel plate (b).

As seen 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 flow channels are lumped to one average channel. The bypass channel is developed to consider the side flow that is 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 element, and each element is composed of 17 MTR-type fuel plates and 2 non-fueled end plates (See Fig. 3). To accurately capture the geometric and physics features of the fuel element, each fuel plate in the fuel element is lumped to one heat structure in the RELAP5-3D model. In order 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. The computational setting for the fuel plate is more clearly illustrated in Fig. 2(b).

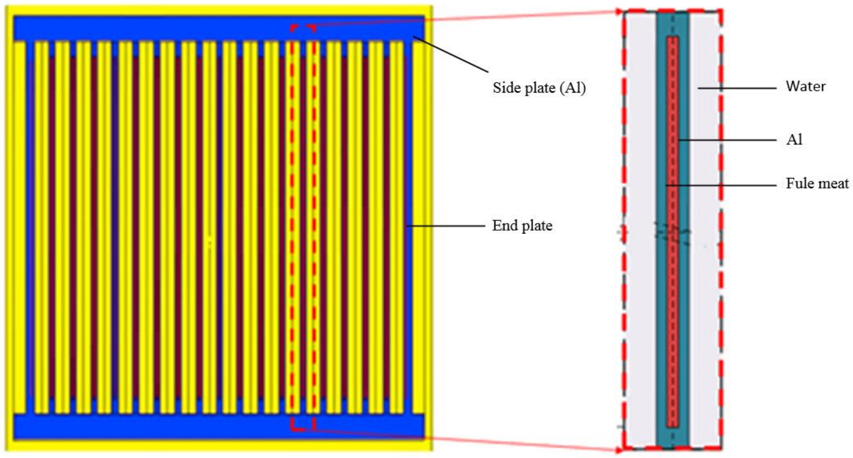


Fig. 3. A radial view of fuel element of the reactor.

The neutronics performance characteristics of the reactor were calculated by MCNP6. A point reactor kinetics model is enabled in the Relap5-3D model to account for the power variation with kinetics parameters provided by neutronics calculations. Reactivity coefficients were calculated but not used in the kinetics model. For the RELAP5-3D kinetic input, delayed neutron fraction and prompt neutron generation time are needed to calculate one of the important inputs, 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 shown in Table I. The axial power distributions for the hot and average channel (required by the RELAP) for the start-up (SU) and end-of-cycle (EOC) core, as shown in Fig. 4, are also obtained from the previous neutronics studies [2]. The SU core stands for the most reactive core status in the fuel cycle, thus the power peaking factor remains as the most eminent.

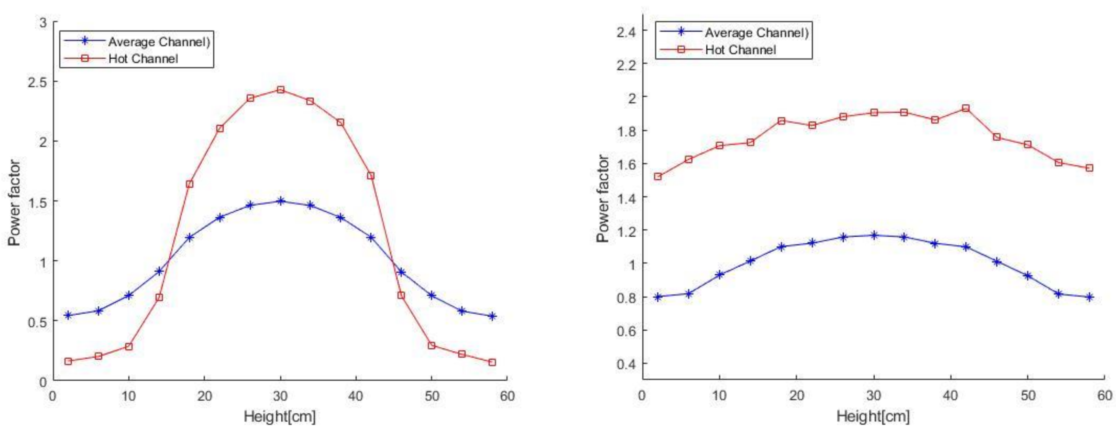


Fig. 4. Power Distribution in average and hot channels of the SU (left) and EOC (right) cores.

Table I summarizes key parameters of the core for RELAP5-3D inputs including materials, fuel assembly geometry data, thermal hydraulics, neutron kinetics parameters, and boundary conditions.

Table I. Key Parameters Used in the 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

Dittus–Boelter correlation [5] was used to predict the heat transfer coefficient in the single phase heat transfer condition and Mirshak critical heat flux (CHF) correlation [6] was chosen to calculate the departure from nuclear boiling ratio (DNBR). This Mirshak correlation applies to the situation that coolant velocity is between 1.52 to 13.72m/s, the subcooling at DNB is between 5 to 75 °C, the pressure is between 0.17 to 0.58MPa, and the hydraulic diameter between is 0.00533 to 0.0117m. The operation conditions of the proposed NIST research reactor fall into these requirements.

3. RESULTS

3.1. Steady-State Conditions

The steady-state results are compared against formerly published PARET results [7] to verify the correctness of the modeling procedure and outcome. Table II and Table III present a quantitative comparison of the temperature predictions from both codes for both SU and EOC cores, respectively.

Table II. Component Temperature Comparison in SU

°C	Hot Channel			Average Channel		
	RELAP5	PARET	Deviation	RELAP5	PARET	Deviation
T (Fuel)	108.05	109.38	1.22%	82.52	83.27	0.90%
T(Cladding)	97.91	98.95	1.05%	76.25	76.90	0.85%
T(Outlet)	48.11	48.01	0.21%	46.56	46.49	0.15%
MCHFR	3.300	3.473	4.98%	5.379	5.654	4.86%

Table III. Component Temperature Comparison in EOC

°C	Hot Channel			Average Channel		
	RELAP5	PARET	Deviation	RELAP5	PARET	Deviation
T (Fuel)	96.94	98.14	1.22%	73.80	74.44	0.86%
T(Cladding)	89.20	90.10	1.00%	68.96	69.43	0.68%
T(Outlet)	53.74	53.66	0.15%	46.54	46.50	0.09%
MCHFR	4.060	4.319	6.00%	6.894	7.245	4.84%

As can be seen in the tables, most of the predictions are within 1% deviation for both codes, except the minimum critical heat flux ratio (MCHFR) estimations. As RELAP5-3D really does not have the option to estimate the CHF based on the Mirshak correlation, hand calculations are performed to obtain the MCHFR using the outputs from RELAP5-3D, and larger errors are under expectation. Nevertheless, the largest devaluation in MCHFR comparison is within 6%. More detailed comparisons of the temperature distribution for fuel centerline, cladding surface temperature and coolant temperature in the hot channel and average channel of the SU core and EOC core can be found in the Ref. [8], which were recently presented in the ANS student conference.

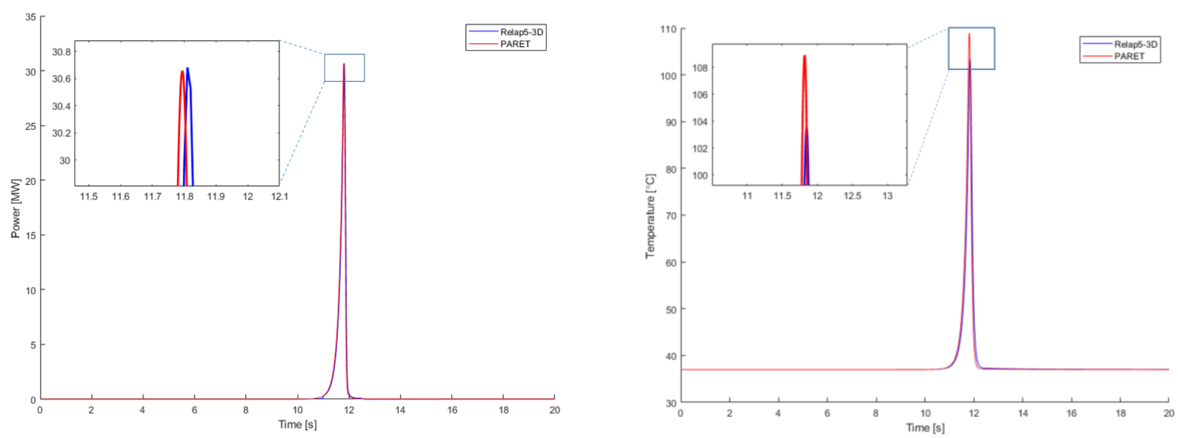
3.2. Transient Scenarios

In order to verify the transient modeling capability of the developed model, a couple of hypothetical reactivity insertion accident analyses are performed to mimic a slow-ramp reactivity-insertion and a fast-step reactivity-insertion accident in the proposed NIST reactor using the RELAP5-3D code. The transient analyses results are compared to the ones produced by PARET code earlier [7].

3.2.1. Slow Reactivity Insertion Accident (SRIA)

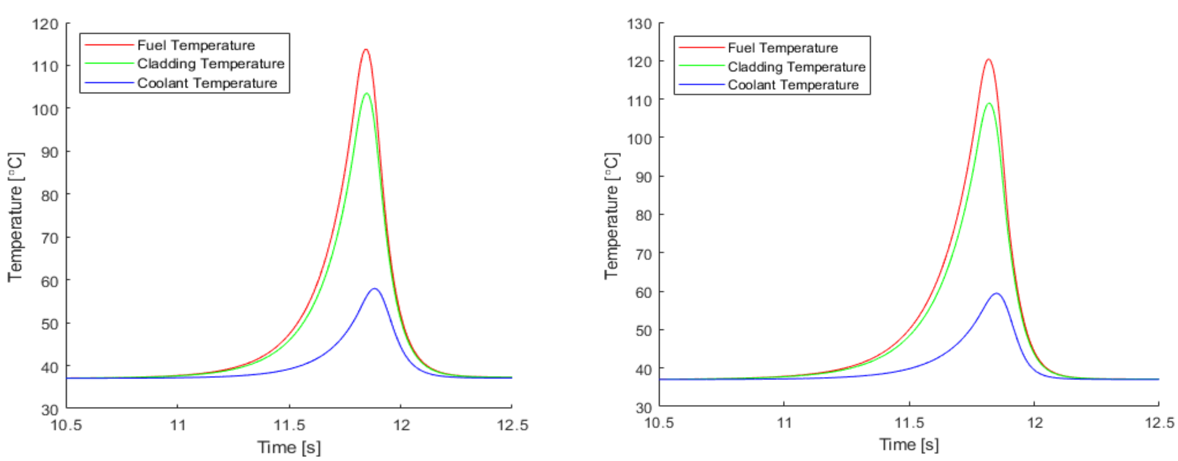
The slow reactivity insertion accident assumes that a positive reactivity is gradually inserted into the initially critical reactor at the low power of 2 Watts (0.01% of full power). The reactivity insertion rate is assumed to be 0.1$/s to mimic a ramp reactivity insertion condition during the reactor start-up. The reactor scram occurs at the power of 24 MW (120% of full power) with a high reactor power trip signal in Relap5-3D. To take into the account of the operation time delay due to the mechanical and electronic circuit effects, a delay of 25 ms is imposed to the control rod reaction after the trip. The control rods are assumed to be inserted with a speed of 1.2 m/s for the reactor trip. All reactivity feedback effects and period trip are neglected in the analyses. The core is considered at the status of the end of cycle (EOC), thus the control rods were fully withdrawn from the reactor at the initial time of the accident.

Fig. 5 shows transient behaviors of the power and peak cladding temperature in the first 20 s into the accidental scenario. Fig. 6 shows the changes of the fuel, cladding and coolant temperature predicted by RELAP5-3D and PARET, respectively.



(a) (b)

Fig. 5. Comparison of power (a) and peak cladding temperature (b) changes in SRIA.



(a) (b)

Fig. 6. Comparison of temperature changes by RELAP5-3D (a) and PARET (b) in SRIA.

As can be seen in both figures, the predictions of Relap5-3D and PARET agree well. The reactor power reaches the maximum power around 30 MW in about 12 s and then rapidly decreases to the decay power level because of the scram. The temperatures exhibit similar behavior trend and reach the highest values at around 12 s. Table IV presents a quantitative comparison of the peak power, peak clad temperature, peak fuel temperature and the corresponding time of occurrence during the SRIA.

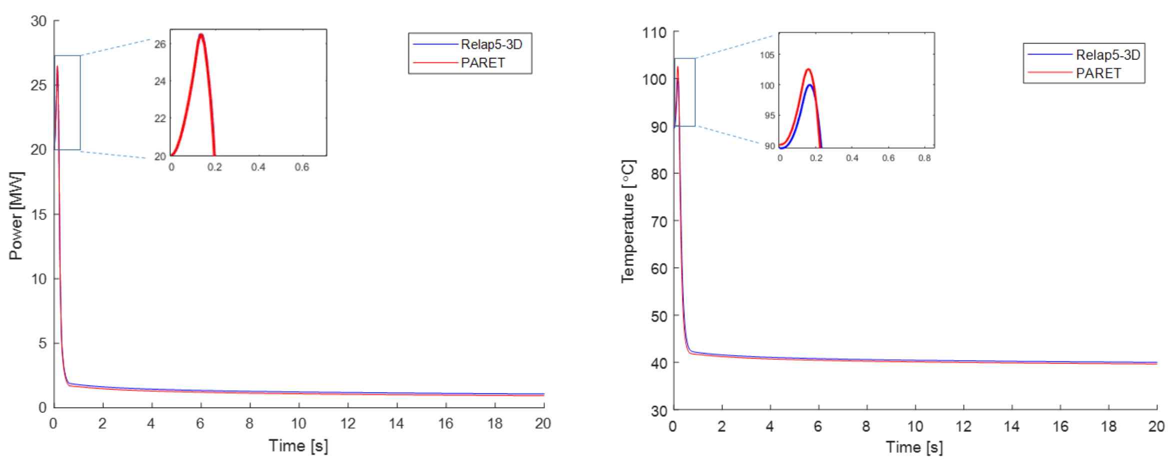
Table IV. Peak Temperature Quantities and Corresponding Occurring Time SRIA

Core Status	RELAP5-3D	PARET	Deviation
Peak power [MW]	30.68	30.66	0.07%
Peak power time [s]	11.81	11.79	0.17%
Reactor trip time [s]	11.79	11.75	0.34%
Peak clad temperature [°C]	103.58	108.93	4.91%
PCT time [s]	11.85	11.82	0.25%
Peak fuel temperature [°C]	113.73	120.41	5.55%
PFT time [s]	11.85	11.82	0.25%

3.2.2. Large Reactivity Insertion Accident (LRIA)

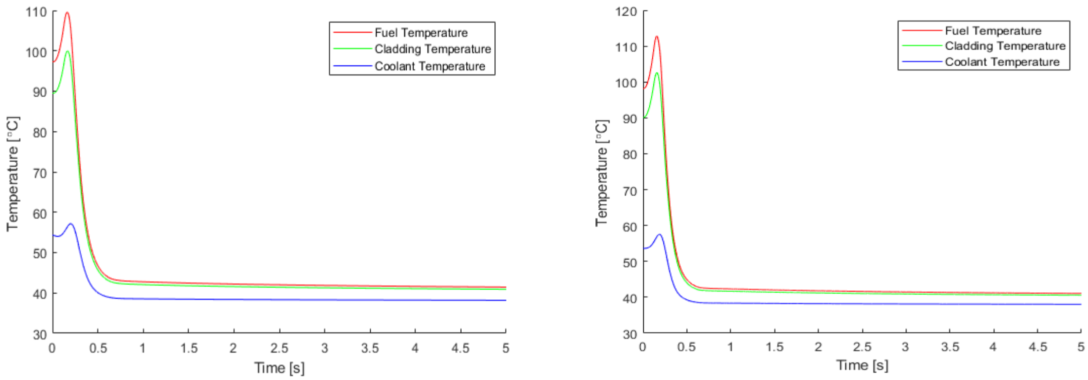
The large reactivity insertion accident assumes a step positive reactivity is inserted into an initially critical core at the power of 20 MW (full power) in a short time period to mimic the control rod ejection accident during the reactor normal operation. A large reactivity 1.5 $ is dumped into the reactor in 0.5 s. The reactor scram occurs at the power of 24 MW (120% of full power) with the high reactor power trip signal. A time delay of 25 ms is considered and the control rods are assumed to be inserted with a speed of 1.2 m/s for reactor trip. All reactivity feedback effects and period trip are neglected in the analyses. The core is considered at the end of cycle (EOC) and the control rods are assumed to be fully withdrawn.

Fig. 7 shows the power variation and peak cladding temperature variation in the first 20s of slow reactivity insertion accident. Fig. 8 shows the temperature changes of the fuel, cladding and coolant predicted by RELAP5-3D and PARET, respectively.



(a) (b)

Fig. 7. Comparison of power variation (a) and peak cladding temperature variation (b) in LRIA.



(a) (b)

Fig. 8. Comparison of temperature variation by RELAP5-3D (a) and PARET (b) in LRIA.

The results are again shown in a good agreement from both codes. The reactor power reaches the maximum power around 26 MW in about 0.1 s and then rapidly decreases to the decay power level following with the scram. Table V presents a quantitative comparison of the peak power, peak clad temperature, peak fuel temperature and the corresponding time of occurrence during the LRIA.

Table V. Peak Temperature Quantities and Corresponding Occurring Time LRIA

Core Status	RELAP5-3D	PARET	Deviation
Peak power [MW]	26.47	26.51	0.15%
Peak power time [s]	0.13	0.13	0.00%
Reactor trip time [s]	0.01	0.01	0.00%
Peak clad temperature [°C]	99.96	102.58	2.55%
PCT time [s]	0.17	0.16	6.25%
Peak fuel temperature [°C]	109.57	112.68	2.76%
PFT time [s]	0.16	0.16	0.00%

4. CONCLUSIONS

The RELAP5-3D model to predict the thermal hydraulics properties of the conceptual NIST research reactor core was established. Both steady-state and reactivity insertion transient calculations were performed. Preliminary results produced by the RELAP5-3D have a good agreement with the ones from the PARET code, which verifies the correctness 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 more design basis accident analyses such as the loss of flow accident (LOF) will be performed using the RELAP5-3D model.

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