COMPARISON OF THERMAL HYDRAULIC ANALYSIS METHODS (CODES) FOR THE UNIVERSITY OF FLORIDA TRAINING REACTOR (UFTR)

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ABSTRACT

Research reactors, such as those sited at universities and national laboratories around the world, operate in a significantly different regime than do light water reactors for power production. Research reactors typically operate at or near atmospheric pressure, compared to the 70+ atmospheres of pressure in power reactors. Further, boiling (including sub-cooled boiling) is generally avoided in the research reactor field, in stark contrast to LWRs, particularly BWRs. As a consequence, research reactors often operate with wider safety margins, allowing for more flexibility in their siting and use as training and educational tools. One drawback of these large safety margins is the reduced impetus for modern code development. The thermal hydraulics of power reactors is often analyzed using modern tools such as RELAP5/MOD3.3 and TRACE V5.0 (NRC), RELAP5-3D (INL), and RELAP7 (in development). Meanwhile, research reactors are often analyzed with older tools such as PLTEMP (ANL, 1980) and PARET (INL, 1969).

The University of Florida Training Reactor (UFTR) is a 100 kW Argonaut- type research reactor going through its 20-year NRC relicensing process. Supporting a comprehensive reworking of the safety analysis, a detailed review of the thermal hydraulic hydraulic analysis has been performed evaluating the best methodologies for modeling in this reactor. Comparisons are made between code results from research reactor specific codes (PLTEMP and PARET) and results from modern, validated, power reactor codes (RELAP) and validated against operational data from the UFTR. A recommendations for preferred codes for research reactors with similar flow regimes is made.

KEYWORDS
Research Reactor, RELAP5, PLTEMP/ANL, PARET/ANL

1. Introduction

The University of Florida Training Reactor (UFTR) is a 100 kW ARGONAUT type research reactor, and serves as a resource for education on reactor physics, control, operations, nuclear regulations, and safety culture. Over its nearly six decades of existence, the reactor has been used for applications such as nuclear medicine isotope production, reactor design benchmarking, nuclear data measurements, and neutron activation analysis supporting research in agriculture, biology, and geology.

Research reactors, such as those sited at universities and national laboratories around the world, operate in a significantly different regime than do light water reactors for power production. Research reactors typically operate at or near atmospheric pressure, compared to the 70+ atmospheres of pressure in power reactors. Further, boiling (including sub-cooled boiling) is generally avoided research reactors field, in stark contrast to LWRs and particularly BWRs. As a consequence, research reactors often operate with wider safety margins, allowing for more flexibility in their siting and use as research, training and educational tools. One drawback of these large safety margins is the reduced impetus for modern code development. The thermal hydraulics of power reactors is often analyzed using modern tools such as RELAP5/MOD3.3[1] and TRACE V5.0 (NRC), RELAP5-3D (INL), and RELAP7 (in development). Meanwhile, research reactors are often analyzed with older tools such as PLTEMP[2] and PARET [3].

As a part of a facility renovation and supporting the NRC 20-year relicensing effort, the UFTR has converted from HEU to LEU fuel and primary and secondary piping systems have been replaced. This has motivated a detailed review of the thermal hydraulic analysis. To demonstrate that a particular core loading using LEU fuel assemblies is safe, it is necessary to understand the technical bases of these requirements and verify that they remain applicable. As part of the safety basis update, an effort was undertaken to modernize the codes and methodologies used to perform safety analyses through reanalysis of the current LEU core. In the present work, comparisons will be made among the following approaches to modeling this sytem:

1. Code results from research reactor specific codes PLTEMP and PARET
2. Code results from modern, validated, power reactor codes, RELAP and TRACE
3. Operational data from the UFTR

By comparison of these three sources of information, recommendations can be made for preferred codes for future UFTR analysis and analysis of other reactors operating similar thermal hydraulic regimes. Previous work [4] sought to prove the inherent safety of the UFTR relative to standard regulatory requirements. This work extends those analysis to compare relevant industry codes and methods to determine optimal analysis techniques.

2. Description of Reactor

ARGONAUT type reactors, such as the UFTR, are especially effective training tools due their similarity to a pressurized water reactor (PWR) [5]. In both designs, primary coolant flows up through and around cladded fuel assemblies. Further, the ARGONAUT design, like a PWR, uses forced convection coolant, in contrast to most other research reactors use a standing pool design (free convection). The main difference is in the operating pressure.

The primary coolant (demineralized water) is pumped upward past the fuel plates at slightly above atmospheric pressure. The exiting warm coolant (approximately 50°C at 100 kW) is gravity-fed to a heat exchanger, where the heat is transferred to the secondary coolant through the heat exchanger. This secondary coolant is pumped in from a well and returned to the storm sewer.

The UFTR licensed rated thermal power level is 100kW giving a power density of 17.88 $kW \cdot L^{-1}$. The thermal flux at the Central Vertical Port (CVP) is approximately 1.5 $\times 10^{12}$ $n \cdot cm^{-2} \cdot s^{-1}$. At rated thermal power the fuel depletion rate is approximately $0.35\ g - UO_2 \cdot 1000EFPH^{-1}$ (EFPH: Effective Full Power Hours at rated thermal power). The core is divided into six fuel boxes and each box is sub-divided into four fuel bundle locations. Each fuel bundle consists of 14 plates of U_3Si_2-Al LEU enriched to 19.75%. Water serves as the both the coolant and partial moderator. Graphite blocks surrounding the fuel boxes provide further moderation and neutron reflection. Dummy plates and dummy bundles are loaded as necessary to limit the installed excess reactivity.

3. Neutronics Modeling

MCNP models have been developed for the UFTR with each operational core loading [4]. These models are used to: 1) estimate the control rods position at BOC, 2) perform whole-core depletion calculation to predict fission product inventory, 3) calculate power distributions for thermal hydraulics calculations, 4) evaluate the control rods reactivity worth, and 4) estimate the neutron fluxes in-core and at experimental ports. A full-core MCNP model was created to perform calculations of core physics parameters using the ENDF/B-VII.1[6] cross-section library. Core physics parameters evaluated include reactivity coefficients, the delayed neutron fraction, and generation time.

The MCNP model was also used to calculate the core reactivity coefficients for the beginning of life (BOL) and end of life (EOL) conditions. As shown in Table I, the UFTR design has negative coefficients for fuel temperature, coolant temperature, and void. (In agreement with much of the literature on reactor physics, these coefficients are indicated in terms of PCM, which is one part in one-hundred thousand.) This ensures that an increase in reactor power will result in a decrease in core reactivity at any core burnup level, which is a precondition to demonstrating that the UFTR is an inherently safe, negligible risk reactor.

Table I: UFTR Reactivity Coefficients for BOL and EOL

Coefficient		BOL	EOL
α_{void} (PCM/%void)	0-5% Void	-125 ± 4	-94 ± 4
α_{void} (PCM/%void)	5-10% Void	-140 ± 4	-106 ± 4
α_{water} (PCM/°C)	21-99°C	-6.7 ± 0.3	-4.8 ± 0.3
α_{fuel} (PCM/°C)	21-127°C	-1.9 ± 0.2	-1.7 ± 0.2
α_{fuel} (PCM/°C)	21-127°C	-1.7 ± 0.1	-1.6 ± 0.1

For transient analysis, a safety-limiting condition can be simulated wherein a fuel bundle with a higher power than in normal operation is created. This bundle is the most dangerous for the case of fuel failure, as it has higher concentrations of radionuclides than will occur in normal operation.

4. Thermal Hydraulic Modeling

Fuel is loaded into six fuel boxes each containing up to four identical fuel assemblies. In these thermal-hydraulic analyses, only the fuel box containing the fuel assembly with the maximum power was considered. In addition, each of the four fuel assemblies in the box was assumed to produce that same maximum power. The axial power distribution for the hottest plate was obtained from the results of the criticality calculations and applies to all fuel plates in an assembly.

The UFTR fuel design uses two semi-circular wedge pins to position the fuel assemblies in each fuel box, as shown in Figure 1 (left). The two-pin LEU configuration with the smallest assemblies in the largest box produces two wide East-West channels of width 0.3675” (9.33 mm). The two-pin configuration is hydraulically superior to the single pin configuration in that it causes more flow into the narrower coolant channels where it is needed most. In both the HEU and the two-pin LEU designs the 0.445” central North-South channel is maintained, as shown in Figure 1 (right).

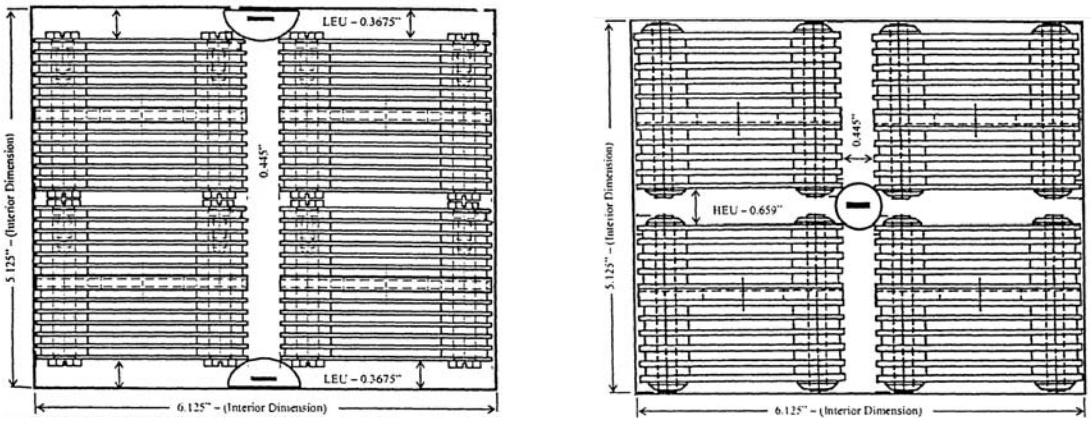

Figure 1: UFTR Fuel and wedge pin locations; LEU (left), HEU (right)

The grid plate, which supports the four fuel assemblies in each fuel box, is included in the hydraulic analysis because it makes the velocity distribution in each fuel box more uniform. The hydraulic model in the code assumes that the hydraulic resistance for each coolant path, from the bottom of the grid plate to the region above the fuel plates, has two components, a form or k-loss and a frictional loss. For each of these parallel paths or channels the pressure drop, ΔP, is given by;

$$\Delta P = (K + \frac{fL}{D}) \cdot \frac{\rho V^2}{2} \tag{1}$$

where K is the k-loss value, f is the friction factor for smooth-walled channels, L is the channel length, D is the channel hydraulic diameter, ρ is the coolant density, and V is the average coolant velocity in the channel. In laminar flow, the value of f is strongly affected by the shape of the channel. The K value includes the form losses at the inlet and exit to the fuel plates and hydraulic resistance due to the grid plate. The total flow area in the grid plate is smaller than the total flow area in the fuel region. A value of 5.0 was assumed for the K-loss value.[7] A larger value of K would result in larger margins to the limiting conditions, such as the onset of nucleate boiling, by causing the thinner channels to have more flow.

Since the ends of the side edges of the fuel plates are open where they abut the side channel, in theory there can be some flow between the fueled channels and the side channel through the center of the fuel box. However, in general, this lateral flow is expected to be small since the local pressure is expected to be essentially uniform at each axial level. The higher vertical flow velocities in the bigger channels, which have the larger hydraulic diameters, tend to keep the axial pressure drops through each of the parallel paths equal and the pressures uniform at each axial level. When the pressure is uniform at each axial level, i.e. during steady state conditions, there is no mechanism for redistribution of laminar flow among adjacent open channels. Detailed studies of cross flow between the UFTR fuel and bypass channels during transient conditions have not been conducted. The hot channel factors include a 20% uncertainty in channel flow distribution.

5. Codes and Methods

5.1 PLTEMP V4.1

The PLTEMP V4.1 is a code focused on research reactors. This code calculates steady-state thermal-hydraulic conditions for fuel assemblies with plate-type or tube-type geometries.[2] The code accounts for pressure drops axially in one dimension including bypass flows, and accounts for thermal effects in two dimensions. PLTEMP determines the friction factors and coolant mass flow rates in each assembly and fuel channel, then calculates the steady-state temperature distribution in the meat, clad, and coolant at each axial node. Parameters such as the Onset of Nucleate Boiling Ratio (ONBR) and Departure from Nucleate Boiling Ratio (DNBR) are calculated along with fuel, clad, and coolant temperatures in each channel.

5.2 PARET/ANL

Like PLTEMP, PARET/ANL Code is intended primarily for the analysis of research and test reactors that use plate-type (flat) fuel elements, or round fuel pins.[3] PARET has been subjected to extensive comparisons with the SPERT I and SPERT II experiments as well as other transient codes. These comparisons were quite favorable for a wide range of transients.[8][9]

5.3 RELAP5-3D

The RELAP series of codes is primarily focused on power reactors. These codes have been developed with a focus on specific applications such as simulations of transients in light water reactor (LWR) systems, loss of coolant accidents, anticipated transients without scram (ATWS), and operational transients such as loss of feedwater, loss of offsite power (LOOP), station blackout, and turbine trip. A version of RELAP5 may be obtained by universities from the NRC. For the present work, a more recent version, RELAP5-3D, is selected. This code, developed at INL is a more generic tool than others in the RELAP family that, in addition to calculating the behavior of a reactor coolant system during a transient, can be used for simulation of a wide variety of hydraulic and thermal transients. [1]. Future codes in the RELAP series include RELAP7; however, this code is not ready for use in UFTR analysis.

6. Steady State Analysis

During steady-state operation, the heat removal must maintain the cladding and fuel temperatures well below the coolant saturation (bubble nucleation) temperature of the coolant. This is achieved by ensuring sufficient margins to Onset of Nucleate Boiling (ONB) which, in turn, ensures that flow instability and departure from nucleate boiling will not occur and damage the fuel.

PLTEMP/ANL & RELAP5-3D were used to determine the thermal hydraulic conditions of UFTR at full power of 100 kW. The codes are also used to determine the safety limits for the core, considering the existing Limiting Safety System Setting (LSSS) (or trip points). Full power and over power conditions were modeled at several inlet flow conditions.

Nominal conditions for the UFTR are; average inlet temperature to the core is 30°C, at a flow rate of 43 gallons per minute (2.69 kg/s) . Inlet coolant flow rate and tempeperatures were selected based on the nominal steady state reactor conditions. LSSS state-points were based on current reactor trip settings in the UFTR Technical Specifications. The steady state conditions which were evaluated using PLTEMP/ANL and RELAP5-3D are shown in Table II

Table II: UFTR Modeled Steady State Conditions (all temperatures in °C)

Parameters			PLTEMP/ANL				RELAP5-3D	
Power [kW]	Inlet Temp.	Inlet Flow Rate [gpm]	$T_{clad,max}$	$T_{coolant,max}$	$ONBR_{min}$	$CHFR_{min}$	$T_{clad,max}$	$T_{coolant,max}$
100	30	30	70.7	67.2	1.687	479.5	69.41	63.20
100	30	43	58.5	54.8	2.407	563.3	60.64	54.35
100	45	30	83.7	80.2	1.387	392.4	83.21	77.12
100	45	43	72.4	68.8	1.954	476.6	75.02	68.86
125	30	30	78.5	74.3	1.418	377.7	72.20	64.46
125	30	43	64.8	60.3	1.972	457.0	67.50	59.71
125	45	30	91.5	87.4	1.155	303.5	86.34	78.75
125	45	43	77.9	73.4	1.634	382.8	81.78	74.14

6.1 Comparison of Critical Heat Flux Correlations

Kolev has reported that there are more than 500 empirical correlations for CHF in forced convection in heated tubes and channels, demonstrating that the final understanding of this phenomenon is not yet reached. The normal RELAP CHF calculation using the Groeneveld 2006[10] lookup table is used for plate type fuel adjacent to narrow channels for medium/low flow conditions and the Gambill-Weatherhead[11] model is used for plate type fuel adjacent to narrow channels for high flow conditions. PLTEMP provides the option to choose which correlation can be implemented based on system conditions. Table III compares CHF correlations for the steady state nominal condition. The large standard deviation is expected due to the varied ranges of experimental data used in the correlations. However, the most conservative CHF correlation still predicts a CHFR greater than 10.

6.2 Comparison of Heat Transfer Correlations

Eight correlations for heat transfer coefficient were used to calculate steady state conditions for the UFTR to establish sensitivity of the PLTEMP/ANL Model to correlation selection. Table IV lists the correlations and subsequent effect on $T_{clad,max}$, $T_{coolant,max}$, and $CHFR_{max}$. Only negligible variation is seen due to the low heat flux, single phase behavior, and good agreement among the correlations.

Table III: Comparison of Critical Heat Flux Correlations at Steady State Condition

Correlation	$CHFR_{min}$
Mirshak-Durant-Towell [12]	381.2
Bernath [13]	98.32
Labuntsov [14]	529.4
Mishima lower bound [15]	12.87
Weatherhead [16]	1103.9
Groeneveld 2006 [17] [10]	563.3
Mishima-Mirshak-Labuntsov [15]	21.28
M. M. Shah [18]	12.94
Sudo-Kaminaga 1998 [19]	17.084

Table IV: Comparison of Heat Transfer Correlations at Steady State Condition

Correlation	$T_{clad,max}[^\circ C]$	$T_{coolant,max}[^\circ C]$	$ONBR_{min}$	$CHFR_{min}$
Sieder-Tate [20]	59.347	55.634	2.338	563.3
Dittus-Boelter [21]	59.347	55.634	2.338	563.3
Colburn [22]	59.331	55.630	2.339	561.2
Petukhov & Popov[23]	59.347	55.634	2.338	563.4
Russian [14]	59.347	55.634	2.338	563.4
Sleicher-Rouse [24]	59.343	55.628	2.338	513.1

7. Transient Analysis

This analysis examines a large addition of positive reactivity. All automatic protective functions are ignored and no operator action is assumed to provide the most conservative analysis. The rise in fuel element temperature is used as a metric to measure the severity of the accident, since the integrity of the fuel will be maintained as long as the fuel maximum temperature remains below the 530°C safety limit.

The PARET-ANL [3] and RELAP5 codes were used for this analysis. Previous work has validated PARET/ANL using experimental data and quantified the resulting uncertainties [9]. Combined with the uncertainties associated with the core input parameters, a total uncertainty of 70% on temperature differences (increase during the accident) must be included when analyzing the final temperature results.

PARET-ANL uses the Martinelli-Nelson method to predict two-phase pressure drop in the case of saturated boiling. WCAP-1997 developed by Westinghouse contains correlations used by PARET-ANL for evaluating subcooled two-phase and single-phase pressure drop[3].

Obenchain[3] validates the PARET-ANL code against experimental data from tests on the high-enriched, plate-type SPERT III C-core. Additionally, Chatzidakis et al.[9], in a more extensive analysis of the DNB correlations used in PARET, show that agreement within 50% can be expected when using the Tong correlation for DNB heat transfer. The uncertainties associated with the input parameters (i.e. reactivity coefficients) allows for an additional 50% error. This implies that for this analysis, using the propagation of uncertainties method, an error of up to 70% for the increase in fuel temperature must be accounted for[4]. That is;

$$T_{max,withuncertainty} - T_{coolant} = 1.7 * (T_{max,calculated} - T_{coolant}) \tag{2}$$

The UFTR model for PARET/ANL was constructed by using the hottest channel and nominal initial conditions of maximum 100 kW power operation with the minimum coolant flow rate of 43 gpm. PARET is limited in its prediction of a thermal crisis by the fact that it employs a steady state DNB correlation. Thus, PARET is limited in its accuracy in the description of hydrodynamic instabilities.

The RELAP nodalization was modeled as two fuel-plate channels. One channel represented the fuel plate and associated coolant for the plate with peak power in the core, and the other channel represented the average of the remainder of the core. The fuel power densities for the peak and average channels were taken from MCNP results for the core operating at 100 kW under steady-state conditions. Given the hydraulically separated nature of the RELAP model, the PARET/ANL and RELAP Hot channel models are equivalent.

7.1 Slow Reactivity Insertion

A reactivity ramp insertion of 0.06% $\Delta k \cdot k^{-1} \cdot sec^{-1}$. This analysis replicates the insertion of reactivity due to control blade withdrawal at the maximum rate allowed by the Technical Specifications. All automatic protective functions are disabled and no operator action is assumed to provide the most conservative analysis. The rise in fuel element temperature is used as a metric to measure the severity of the accident, since the integrity of the fuel will be maintained as long as the fuel maximum temperature remains below the safety limit.

Figure 2 shows the comparison of results between PARET/ANL and RELAP. Table V compares the results for the concerned variables.

Table V: Predicted Values for Slow Reactivity Insertion

Parameter	RELAP	PARET/ANL
$Power_{max}$ [MW]	1.01	1.11
$T_{coolant,max}[°C]$	78.84	74.74
$T_{clad,max}[°C]$	90.02	77.28
$T_{fuel,max}[°C]$	109.85	77.30

7.2 Fast Reactivity Insertion

The largest reactivity insertion studied was $1.2 inserted in 0.5 seconds. Any larger or faster insertion of reactivity would produce unverifiable results based on stated code limitations.

Figure 3 shows the comparison of results between PARET/ANL and RELAP. Table VI compares the results for the concerned variables. Discrepancies between the two models during the fast reactivity insertions can be accounted for by the modeling differences, nodal compositions, and code intricacies. Decreases seen in the PARET/ANL clad and fuel temperature prior to the transient are due to differences between the input and code calculated steady state parameters.

The 70% uncertainty on temperature difference means that the maximum fuel temperature is 117.9°C, which is still well below the 530°C safety limit. The PARET-ANL code therefore shows that a positive reactivity insertion of $1.2 agrees with the RELAP results and that a small transient is incapable of causing damage to the fuel, even in the event that no operator action is taken nor automatic protective

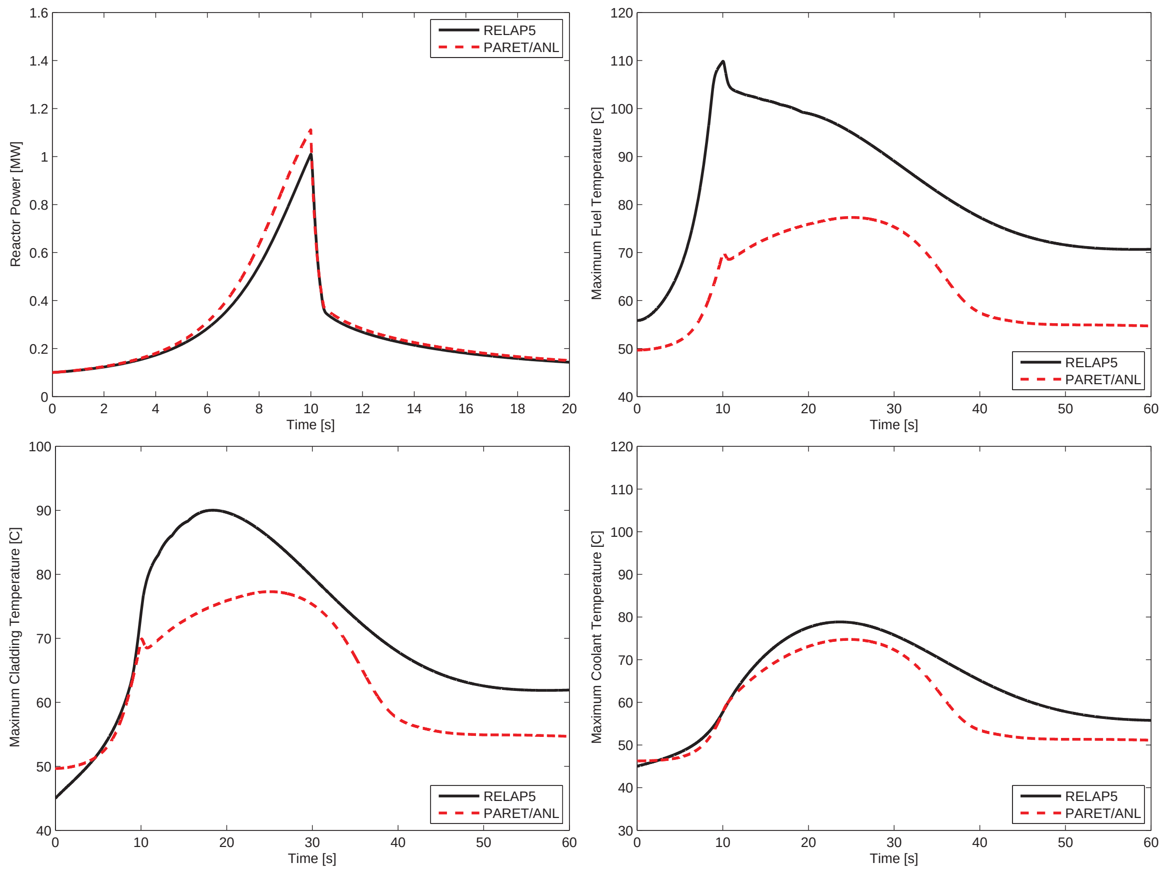

Figure 2: Comparison of RELAP & PARET/ANL for slow reactivity insertion – (Top Left) Reactor Power vs. Time, (Top Right) $T_{fuel,max}$ vs. Time, (Bottom Left) $T_{clad,max}$ vs Time, (Bottom Right) $T_{coolant,max}$ vs Time

features activated. The UFTR core design therefore maintains an adequate margin to safety in the event of a reactivity insertion accident.

8. Discussion

8.1 Engineering Hot Channel Factors

The hot channel factor for bulk water temperature rise F_b, hot channel factor for heat flux F_q, and hot channel factor for heat transfer F_h take into account fuel fabrication tolerances, power density distribution measurement error, coolant flow rate, and heat transfer coefficients. These effects can limit reactor operation due to calculated thermal safety margins. Hot channel factors described by [25] and recommended by [2] were used. Bulk temperature uncertainty (F_b) of 45% predicts nominal steady state exit temperatures of 79.6°C.

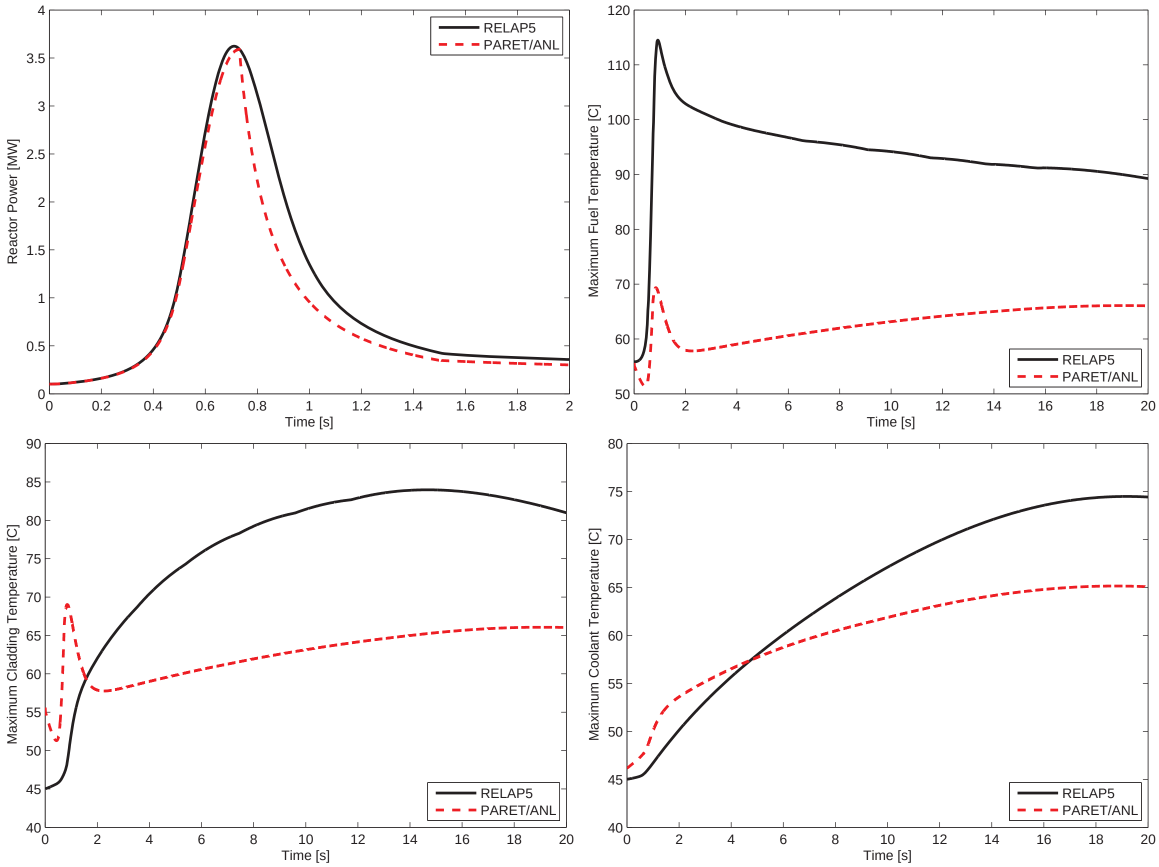

Figure 3: Comparison of RELAP & PARET/ANL for fast reactivity insertion ($1.2/0.5s) – (Top Left) Reactor Power vs. Time, (Top Right) $T_{fuel,max}$ vs. Time, (Bottom Left) $T_{clad,max}$ vs Time, (Bottom Right) $T_{coolant,max}$ vs Time

8.2 Steady State Analysis

The RELAP steady state solution for the nominal case had a 3°C larger $\Delta T_{clad,coolant}$. RELAP5 deviates with PLTEMP/ANL at predicting flows at lower pressures when nucleate boiling is most likely to be present.[26] The large deviations in predicted $CHFR_{m}in$ as shown in Table IV can be attributed to the data ranges for the correlations. The low flow nominal condition is well below the mass flux range in which the models are valid.

The Groenveld 2006 Lookup Table applies over a broad range of pressure, mass flux, quality, tube diameter, geometry, and heat flux shape. Both RELAP5 and PLTEMP/ANL utilize the latest lookup and is recommended for consistency and relative accuracy.

8.3 Transient Analysis

The large differences in the transient analysis results can be attributed to the difference in two-phase modeling between RELAP and PARET/ANL and the modeling nodalization. The average channel node in the RELAP model does not void at the same time and the hot channel and affects the point kinetics

Table VI: Predicted Values for Fast Reactivity Insertion

Parameter	RELAP	PARET/ANL
$Power_{max}$ [MW]	3.62	3.59
$T_{coolant,max}[^{\circ}C]$	74.49	65.15
$T_{clad,max}[^{\circ}C]$	83.98	69.04
$T_{fuel,max}[^{\circ}C]$	114.51	69.35

solution.

Previous versions of RELAP5 have been shown to overestimate transient data for MTR type reactors but still gives comparable results to the experiments for modest transients.[28] However, PARET/ANL has previously tracked well against RELAP for benchmark analysis [27], and suggests model improvements can be made. Transient analysis within bounds of licensing basis is expected to verify and compare model applicability to the University of Florida Training Reactor.

9. Conclusions

The University of Florida Training Reactor (UFTR) has been analyzed under steady state conditions and with transients. The results presented included comparisons for cases at steadystate and two reactivity insertion events. RELAP, which was developed for LWRs, has proven to predict reasonable, conservative values for regulatory required analyses. Further model refinement coupled with data from the operational UFTR will provide the basis for final model validation.

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