Summary of RELAP5 Assessments Performed in Relation to Conversion Analysis of Research Reactors

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Nuclear Engineering Division

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TABLE OF CONTENT

1. INTRODUCTION	4
2. COMPARISON WITH EXPERIMENTAL DATA	4
2.1. Operation at Constant Power and Rod Withdrawal Transient at NIRR-1	4
2.2. Startup tests at MITR-II	7
2.3. Loss-of-Flow and Loss-of-Pressure Tests at BR2	7
2.4. SPERT Reactivity Insertion Experiments.....	10
2.5. Pressure Drop for MARIA Fuel Assemblies	11
2.6. Measured Fuel Temperatures for TRIGA Reactors.....	12
3. CODE-TO-CODE COMPARISON.....	15
3.1. Comparison of RELAP5 and STAR-CD	16
3.2. Comparison of RELAP5 and PLTEMP/ANL.....	18
3.3. Comparison of RELAP5 and PARET/ANL	22
4. ADDITIONAL CONSIDERATIONS	25
4.1. Analysis of Critical Heat Flux in TRIGA	25
4.2. Boiling model.....	28
5. SUMMARY AND CONCLUSIONS	29
6. REFERENCES	30

1. INTRODUCTION

RELAP5 is a standard tool used by the nuclear industry. The code has been used for the analysis of transients and postulated accidents in Light Water Reactor (LWR) systems, including both large-break and small-break loss-of-coolant accidents (LOCAs). Even though most of the code assessments were performed for regimes more relevant to the analysis of power reactors than research reactors, the fact that the code has received extensive verification and validation gives confidence in the quality of the code. Since research reactors were designed with a wide variety of operating conditions (flow, pressure and power density), the applicability of RELAP5 should be evaluated case by case based on the specifics of the reactor and the intended analysis.

The current document summarizes a series of RELAP5 assessments performed in relation to the conversion of research reactors from highly-enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel. Even though this document does not present a formal validation case for all uses of RELAP5 for all research reactors, it provides a strong technical basis to justify the use of the code for many applications. It should be also noted that the assessments presented in this document do not represent an exhaustive list of all RELAP5 verification, validation, and benchmarking activities relevant to research reactors. Therefore it is expected that future revisions of this document will contain additional assessments.

Section 2 presents comparisons with experimental measurements performed as part of reactor conversion projects. Section 3 presents code-to-code comparisons performed for steady-state and transient conditions. Section 4 presents additional considerations related to the use of RELAP5 for research reactors.

2. COMPARISON WITH EXPERIMENTAL DATA

This section summarizes a series of comparisons of RELAP5 results with measurements obtained from tests performed at various research reactors. These comparisons were performed within the context of the converting these research reactors to LEU fuel.

2.1. Operation at Constant Power and Rod Withdrawal Transient at NIRR-1

As part of the conversion analysis of the Miniature Neutron Source Reactor (MNSR) at the Center for Energy Research and Training at Ahmadu Bello University, RELAP5-3D Version 2.3 predictions were compared to measured data obtained during operation at constant power and during a rod withdrawal transient. This comparison and the figures reproduced here were obtained from Ref. [1].

The NIRR-1 MNSR is a 15kW pin-type reactor cooled using natural circulation through the core which is located inside a tank. The heat removal from the tank is obtained by conduction through the tank wall to the water in a surrounding pool. During normal operation, the core inlet temperature slowly increases until the negative void feedback reactivity cannot be compensated by the control mechanism and the reactor shuts down.

2.1.1. Operation at Constant Power

The authors state in Ref. [1] that the calculated inlet and outlet temperatures agree well with the measurements indicating that for most of the transient the RELAP3-3D Version 2.3 model accurately accounts for all of the significant heat transfer in the system, i.e.,:

- The heat capacity of the water in the tank in the early part of the transient
- The heat transferred through conduction across the tank wall to the pool
- The heat capacity of the pool itself

They noted however that heat transfer from the pool to the air was not included in the RELAP5-3D model which could explain why the calculated inlet and outlet temperatures rise toward the top of the experimental data band after 4.9 hours of operation.

Figure 1 (extracted from Ref. [1]) shows the measured and calculated inlet and outlet coolant temperatures and pool temperatures during the operation of NIRR-1 at a constant power of 15 kW.

Figure 1. Inlet and Outlet Coolant Temperatures for NIRR-1 Operation at Constant Power [1].

2.1.2. Rod Withdrawal Transient

After comparing RELAP5 predictions with experimental measurements of a 3.77 mk reactivity insertion, the authors concluded that the calculated results agree almost exactly with the measured data and that the reactivity feedback is modeled accurately by RELAP5-3D Version 2.3 for the NIRR-1 MNSR reactor. The authors noted that the positive reactivity from heated water above the core and from the radial beryllium reflector were included in the transient calculations. Figures 2 and 3 (extracted from Ref. [1]) compare RELAP5-3D Version 2.3 predictions with measurements for the core power as well as the inlet and outlet coolant temperature.

Figure 2. Reactor Power following a 3.77 mk Reactivity Insertion [1].

Figure 3. Inlet and Outlet Temperatures for a 3.77 mk Reactivity Insertion [1].

2.2. Startup tests at MITR-II

Ref. [2] presents a comparison of the predicted coolant outlet temperature from RELAP5-3D Version 2.3.5 with measured data for three thermocouples (TC-6, TC-7 and TC-9) during a MIT reactor (MITR-II) startup test. MITR-II, located in Boston, MA, USA, is a 6 MW research reactor moderated and cooled by light water and reflected by heavy water.

Figure 4 (extracted from Ref. [2]) compares the coolant exit temperature measurements with the RELAP5-3D predictions during the startup test.

Figure 4 Measured and Calculated Values for Coolant Outlet Temperature [2].

The authors state that their expectation was that the measured temperatures would fall between the predicted average and peak temperatures (within experimental error). They also observe that even though RELAP5-3D Version 2.3.5 seems to over-predict the temperature, the predicted trend and values are close to the measured values.

2.3. Loss-of-Flow and Loss-of-Pressure Tests at BR2

BR2 is a water-cooled reactor moderated by water and beryllium. The beryllium is a matrix of hexagonal prisms each having a central bore that contains either a fuel assembly (FA), a control rod, an experimental device, or a plug. The core is located inside an aluminum pressure vessel, and at nominal conditions, the channel inlet pressure is 10.3 bars while the inlet water temperature varies from 30 to 40°C. Normally, the coolant flows from the top of the core to the bottom with an average speed of 10.4 m/s in the fuel assemblies.

To support the conversion analysis of the BR2 research reactor (located at SCK-CEN in Mol, Belgium) and extend the validation basis of RELAP5-3D Version 2.4.2 for the safety analysis of the conversion of research reactors, the simulation loss-of-flow/loss-of-pressure (LOF/LOP) tests performed at BR2 in

1963 was performed. More details about this work are presented in Refs [3, 4, 5]. The figures presented in this section are extracted from Ref. [4].

These tests are characterized by loss of flow initiated at different reactor power levels with or without loss of system pressure, reactor scram, flow reversal and reactor cooling by natural circulation. Comparisons of RELAP5 predictions with experimental measurements for peak cladding temperatures during the transient at four axial locations in an instrumented fuel assembly (thermo-couples TC11, TC12, TC13 and TC14) were performed. RELAP5 simulations show that accurate representation of the pump coastdown characteristics, and of the power distribution, especially after reactor scram, between the fuel assemblies and the moderator/reflector regions are critical for the correct prediction of the peak cladding temperatures during these transients.

The first test, Test A, was performed at 400 W/cm^2 and is characterized by total loss of flow without loss of system pressure, followed by reactor scram, flow reversal and reactor cooling by natural convection. The measured and predicted cladding temperatures are shown in Figure 5.

Figure 5. Measured and RELAP5 cladding temperatures for Test A at 400 W/cm^2 [4].

In test A, the predicted maximum peak cladding temperature (TC14) is 10.5°C higher than the measured value, and the maximum discrepancy between predicted and measured temperatures at the time of the second peak is 11°C (TC12). The predicted time of the second peak is about 2 s longer than the measured time.

The second test, Test C at a peak heat flux of 600 W/cm^2 , is characterized by total loss of flow without loss of system pressure, followed by reactor scram, flow reversal and reactor cooling by natural convection. The measured and predicted cladding temperatures are shown in Figure 6.

Figure 6. Measured and RELAP5 cladding temperatures for Test C at 600 W/cm² [4].

In Test C, at steady state, the maximum discrepancy between predictions and measurements is 17.7°C at the location of thermocouple TC14. At the time of the second peak, the maximum discrepancy between predicted and measured temperatures is 14.4°C, also at the same location. RELAP5 predicts that the peak temperature is reached a little earlier than the measured time. The maximum discrepancy between predictions and measurements for the time of the peak is 2.8 s.

The last simulated test, Test F, was performed from a steady state peak heat flux of 400 W/cm² and is characterized by a total loss of flow with loss of system pressure, reactor scram, flow reversal and reactor cooling by natural convection. The measured and predicted cladding temperatures are shown in Figure 7.

Figure 7. Measured and RELAP5 cladding temperatures for Test F at 400 W/cm² [4].

From Fig. 7, the maximum discrepancy between predicted and measured temperatures at steady state is 18.8°C at the location of thermocouple TC14. The maximum discrepancy between predicted and measured temperatures at the time of the peak is 7.4°C, also at the location of thermocouple TC14. The maximum discrepancy between predictions and measurements for the time of the peak is 0.6 s.

2.4. SPERT Reactivity Insertion Experiments

In Ref. [6], the results from RELAP5/MOD3.2 code were compared with the SPERT-IV series of experimental reactivity insertion transients. The authors concluded that the RELAP5/MOD3.2 code overestimates the transient data but still gives comparable results to the experiments for modest transients. They found that the addition of a correlation reflecting better the heating conditions during a rapid reactivity insertion such as Rosenthal & Miller (R&M) correlation [7] gives a much needed improvement in the RELAP5/MOD3.2 results. Despite that improvement, they observed that RELAP5 results diverge from the experimental data for insertions greater than $\beta 1.2$ (approximate threshold for nucleate boiling for the SPERT-IV reactor) and could not recommend RELAP5/MOD3.2 for transient above that threshold. Figures 8a, 8b and 8c (extracted from Ref. [6]) compare the peak power, energy generated and peak clad temperature obtained from RELAP5/MOD3.2 and SPERT-IV.

Figure 8. Calculated results from RELAP5/MOD3.2, RELAP5/MOD3.2 with R&M and SPERT-IV measurements for (a) peak power, (b) energy generated, and (c) peak clad temperature. [6]

2.5. Pressure Drop for MARIA Fuel Assemblies

In Ref. [8] the authors compare the relationship between pressure drop and flow rate predicted by RELAP5-3D Version 2.3 with measurements performed by the Institute of Atomic Energy (IAE, now the National Center for Nuclear Research) in an out-of-reactor test stand adjacent to MARIA reactor (Otwock, Poland). There is additional information about the fuel assembly in Sec. 3.1 of this report. The pressure drops measurements were performed at flow rates up to 30 m³/h using water temperatures in the range of 20 to 80°C.

Figure 9 and 10 (extracted from Ref. [8]) show representative results for two fuel assembly types. Experimental data is denoted by small solid markers; large open-center markers are used for RELAP5 results (denoted by “R5” prefix in legend). The pressure drops are identified as “p1-p2” for inlet to bottom, “p2-p3” for bottom to outlet, and “p1-p3” for inlet to outlet. The experimental data and RELAP5-3D calculated values have been adjusted to remove the static gravity head; therefore, the expectation is that there is no pressure drop when there is no coolant flow.

Figure 9. Measured and Predicted Pressure Drop vs. Flow Rates for MARIA HEU Fuel Assembly [8].

Figure 10. Measured and Predicted Pressure Drop vs. Flow Rates for MARIA LEU Fuel Assembly [8].

The authors noted that effort must be made to include in the RELAP5 model as much of the geometry as possible, in particular flow area changes and locations where the coolant flow splits into (and rejoins from) multiple paths. Hydraulic losses at area changes were treated using the abrupt area change junction option in RELAP5. The RELAP5 code does not automatically account for hydraulic loss associates with a change in coolant flow direction, therefore, a K-type constant loss coefficient was inputted at each relevant location, and the loss was computed as this value times velocity squared.

Finally, the authors concluded that the calculated results are in excellent agreement with the measurements which gives confidence that RELAP5-3D Version 2.3 could be used for other aspects of the design analysis, including changes in the geometry.

2.6. Measured Fuel Temperatures for TRIGA Reactors

As part of the HEU to LEU conversion of the Oregon State University (OSU) and University of Wisconsin (UW) TRIGA reactors, a series of limited comparisons between measured and calculated fuel temperatures were performed. The OSU and UW TRIGA are pool-type reactors, of 1.1MW and 1.0MW respectively, using UZrH fuel and cooled through natural circulation. This section summarizes these results as reported in the publically available conversion Safety Analysis Report (SAR), Request for Additional Information (RAI) and post-conversion startup testing.

2.6.1. Comparison with RELAP5-3D for the OSU TRIGA Reactor

Comparisons of measured and predicted fuel temperatures were performed [9, 10] as part of the conversion analysis of the OSU TRIGA reactor

The conversion SAR (Ref. [9]) states that the fuel temperatures measured by the Instrumented Fuel Element (IFE) (B4 position) ranged from 356°C to 373°C for the HEU Beginning-of-Life (BOL) core at a power of 1.0 MW. Using an evaluated power of 15.81 kW for the IFE, a RELAP5-3D^a calculation was performed to obtain a radial temperature profile across a fuel element using a gap of 0.1 mil. This approach is consistent with Ref. [10] where a parametric study was performed to determine the most appropriate gap thickness to use in the analysis. The authors of the conversion SAR state that the gaseous content of the gap was chosen to be the default setting for RELAP5-3D which produces conservative results.

Figure 11 (extracted from Ref. [9]) shows that RELAP5-3D predicted fuel temperatures in the pin are higher than the two measured values, which are shown in the circle, by approximately 17°C to 34°C depending on which IFE temperatures is used.

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^a The explicit RELAP5 version was not found but based on Ref. [10] and the reported date of the User's manual quoted, it is believed that RELAP5-3D Version 2.4 was used.

Figure 11. Calculated and Measured Fuel Temperatures in a Pin at 15.81 kW [9].

2.6.2. Comparison with RELAP5/MOD3.3 for the UW TRIGA Reactor

Comparisons of calculated and measured fuel temperatures were performed [11, 12] as part of the conversion analysis of the UW TRIGA reactor.

Figure 12 (extracted from Ref. [11]) compares fuel temperature measurements from the E3NE IFE taken during startup testing to RELAP5/MOD3.3 predictions for the pin E4SE of the analyzed configuration. The authors state that this pin was chosen because its calculated power peaking factor was identical to the power peaking factor to the reported peaking factor E3NE IFE taken during startup testing. E4NE pin was therefore modeled with a core power of 1.0 MW (i.e., an E4NE pin power of 12.1 k), which corresponds to the same conditions reported in the UW TRIGA SAR for the HEU core loaded with all FLIP elements.

Figure 12. Calculated (pin E4 NE) and Measured (pin E5 NE) Fuel Temperatures at a Pin Power (12.1 kW) using a 0.1 mil gap [11].

Figure 13 (extracted from Ref. [12]) compares the fuel temperatures measured by the E3NE IFE during startup testing measurements with the calculated fuel temperatures for a pin at the same location in the analyzed configuration.

Figure 13. Calculated and Measured Fuel Temperatures in E3NE location using a 0.1 mil gap [12].

Figure 14 (extracted from Ref. [12]) compares the fuel temperatures measured by the D4SW IFE during startup testing measurements with the calculated fuel temperatures for a pin at the same location in the analyzed configuration.

Figure 14. Calculated and Measured Fuel Temperatures in D4 SW [12].

Ref. [13] contains fuel temperature measurements performed using the two IFEs (E5NE and D4SW) as part of the start-up measurements required by the Nuclear Regulatory Commission (NRC) after the conversion. In addition to the two IFEs at the grid positions mentioned above, the report provides a map of the temperatures across the core for all positions except D5 (transient rod). Table 1 (extracted from Ref. [13]) compares calculated and measured fuel temperatures for the converted LEU core.

Table 1. Predicted and Measured Fuel Temperature in the LEU Core After Conversion [13].

	LEU Predicted	LEU Measured	
		Bundle 62	Bundle 72
D4 SW Temp (°C)	391-480	342.3	405.3
E3 NE Temp (°C)	264-317	269.1	318.6

3. CODE-TO-CODE COMPARISON

Over the years, code-to-code comparisons between different versions of RELAP5 and other codes have been performed. Since these codes may not be formally validated, these comparisons do not constitute a validation of RELAP5 but they increase the confidence that some of the physical models and various other aspects of RELAP5 are appropriate for research reactors.

3.1. Comparison of RELAP5 and STAR-CD

In Ref. [14] the authors assess RELAP5/MOD3.2 thermal-hydraulics model of peak power density fuel assembly (FA) in the MARIA reactor (located at the Polish National Centre for Nuclear Research (NCNR), Otwock, Poland) by comparison with results from the computational fluid dynamics code STAR-CD at steady-state operating conditions. The high flux research reactor MARIA is a water and beryllium moderated reactor of a pool-type with a graphite reflector and pressurized channels containing concentric six-tube HEU fuel assemblies. The LEU fuel assembly has only five tubes. The operating conditions for the peak fuel assembly were: a power density of 6046 MW/m^3 , inlet temperature of 50°C , volumetric flow rate of $25 \text{ m}^3/\text{h}$, and fuel assembly power of 2.0 MW .

In the RELAP5/MOD3.2 model used for this analysis, the heat is transferred to the coolant on both sides of the fuel tubes (i.e., one-dimensional radial heat transfer) and the coolant flow is modeled as one-dimensional axial flow between the fuel tubes. Single-phase heat transfer is modeled through the use of correlations. The STAR-CD model is fully three-dimensional which adds details such as axial and azimuthal heat conduction, three-dimensional velocity profiles and an improved representation of the fuel assembly plena. In STAR-CD, the heat transfer is computed on a somewhat "first principles" basis.

Figure 15 (extracted from Ref. [14]) presents a comparison of the coolant, clad and fuel temperatures across the radial direction^b of an LEU fuel assembly at the height of peak power density. The calculated results were obtained using: (1) the RELAP5/MOD3.2 with two single phase heat transfer coefficient correlations, Petukhov (Pet) and Dittus-Boelter (DB), and (2) STAR-CD.

Figure 15. RELAP5/MOD3.2 and STAR-CD Calculated Temperatures for Each Material Layer in a LEU Fuel Assembly at 2MW [14].

^b Referred to as material layers in Figure 15 legend

Figure 16 (extracted from Ref. [14]) presents a comparison of the RELAP5/MOD3.2 and STAR-CD predicted velocities at the top of the fuel meat for all coolant channels of a LEU fuel assembly.

Figure 16. RELAP5/MOD3.2 and STAR-CD Calculated Velocities in a LEU Fuel Assembly at 2MW [14].

Figure 17 (extracted from Ref. [14]) present a comparison of the RELAP5/MOD3.2 and STAR-CD predicted exit temperatures for each coolant channel of an LEU fuel assembly.

Figure 17. RELAP5/MOD3.2 and STAR-CD Calculated Exit Coolant Temperatures in a LEU Fuel Assembly at 2MW [14].

The authors conclude that the agreement between RELAP5-3D Version 2.3 and STAR-CD provides a high confidence that modeling the peak power sector using RELAP5 with the Dittus-Boelter heat transfer correlation will provide an adequate approximation for calculating the peak clad temperatures in the HEU and LEU MARIA fuel assemblies.

3.2. Comparison of RELAP5 and PLTEMP/ANL

PLTEMP/ANL has been developed at Argonne National Laboratory (ANL) under the sponsorship of the U.S. Department of Energy/NNSA. The code was originally created in 1980 to analyze steady-state thermal-hydraulic conditions in the Kyoto University High Flux Reactor (KUHFR). It has generally been used to predict steady-state thermal-hydraulic performance and safety margins of research reactors using assemblies composed of plates or nested tubes cooled by either light or heavy water.

3.2.1. Comparison of Clad Surface Temperature Predictions

As an example, Ref. [15] compares the surface clad temperatures obtained using RELAP5/MOD3.2.1.2 and PLTEMP/ANL for an LEU fuel assembly with a single blocked channel in the IRT-1 reactor (Tajoura, Libya). IRT-1 is a 10 MW pool-type research reactor using fuel assemblies composed on concentric square tubes with rounded corners. Figures 18 and 19 (extracted from Chapter 2 of Ref. [15]) compares the surface clad temperatures on each side of tubes 1 and 7 obtained using RELAP5 and the analytical solution method implemented in PLTEMP/ANL.

Figure 18. RELAP5/MOD3.2 and PLTEMP/ANL Steady-State Clad Temperatures of Tube 1 of a LEU Fuel Assembly in IRT-1 [15].

Figure 19. RELAP5/MOD3.2 and PLTEMP/ANL Steady-State Clad Temperatures of Tube 1 of a LEU Fuel Assembly in IRT-1 [15].

The authors conclude that the results obtained by the two codes are in good agreement.

3.2.2. Comparison of Mass Flow Rates and Exit Temperature under Natural Convection

Ref. [15] compares five different approaches for calculating mass flow rates and exit temperature under natural convection regime for a benchmark problem with the following characteristics: a 1.05 m long vertical coolant channel with a 0.75 m long heated length, an upper and lower unheated lengths of 0.15 m, a power of 25 kW distributed uniformly over the heated length, an inlet temperature of 25 °C, a rectangular cross section of thickness 3 mm and width 0.3 m, an inlet and exit pressure loss coefficient of 0.5 and 1.0, and an absolute pressure at the channel inlet is 5 bar.

The authors compare the following approaches: hand calculation, a *Mathematica*^{TM} solution, the NATCON code [16], RELAP5-3D Version 2.3, and PLTEMP/ANL. The results of these various calculations are reproduced in Table 2 (extracted from Chapter 9 of Ref. [15]).

Based on the results in Table 2, the authors concluded that the RELAP5-3D solution was slightly inaccurate because the nodal coolant temperatures at node exit are used by the code as node-centered temperatures in calculating buoyancy and frictional pressure drop resulting in a predicted flow rate about 5% higher than the PLTEMP/ANL-calculated flow rate.

Table 2. Comparison of Different Codes to Verify the Natural Circulation Mass Flow Rate and Exit Temperature [15].

Input or Calculated Quantity	Hand Calculation Using NATCON Properties	Mathematica Calculation			NATCON Code Solution	RELAP5 Code Solution	PLTEMP/ANL V3.5 Solution	
		Using NATCON Coolant Properties	Using PLTEMP Coolant Properties	Using RELAP5 Coolant Properties			General Method, Friction Factor = $96/Re$	General Method, Friction Factor = $94.7174/Re$
Inlet Temp, °C	25	25	25	25	25	25	25	25
Power per Channel, W	25000	25000	25000	25000	25000	25000	25000	25000
Loss Coeff. at Inlet	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
Loss Coeff. at Exit	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
CALCULATED QUANTITIES								
Flow per Channel, kg/s	0.1086	0.1088	0.1088	0.1114	0.1087	0.1141	0.1088	0.1093
Exit Temp, °C	79.967	79.900	79.891	78.660	79.970	77.44	79.891	79.646
BUOYANCY ΔP_{buoy}, Pa								
Axial Region 1	0.0	0.0	0.0	0.0	0.0	0.005	0.0	0.0
Axial Region 2 (Heated Section)	75.716	75.761	75.793	78.965	75.783	82.620	75.786	75.342
Axial Region 3	36.040	36.001	36.009	36.272	36.034	35.172	36.015	35.785
Channel ΔP_{buoy}	111.756	111.762	111.802	115.237	111.817	117.797	111.801	111.127
FRICTIONAL PRESSURE DROP ΔP_{fric}, Pa								
Inlet Loss	3.652	3.664	3.667	3.840	3.659	4.028	3.667	3.700
Axial Region 1	22.268	22.307	22.300	22.542	22.348	23.084	22.300	22.101
Axial Region 2 (Heated Length)	68.970	69.239	69.276	71.619	69.275	71.417	69.275	68.758
Axial Region 3	9.015	9.037	9.041	9.360	9.033	9.637	9.042	8.894
Exit Loss	7.487	7.513	7.518	7.875	7.503	8.254	7.518	7.585
Momentum Flux Term	0.368	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Channel ΔP_{fric}	111.760	111.761	111.802	115.236	111.817	116.420	111.803	111.126

3.2.3. Comparison of Pressure Drop during Subcooled Nucleate Boiling

As part of the verification of PLTEMP/ANL, pressure drops calculated with PLTEMP/ANL (ignoring the effect of the nucleate boiling) are compared to pressure drops calculated by RELAP5-3D Version 2.3 for a simplified problem representative of the conditions in MURR (single channel, power of 200 kW and an exit pressure of 5 bars). The pressure drops from both codes are compared as a function of flow rate from 1.2 kg/s to the minimum stable flow rate obtained using the Whittle and Forgan (W&F) flow instability correlation [17].

Table 3 (extracted from Chapter 15 of Ref. [15]) shows that up to the W&F predicted flow instability (0.6264 kg/sec in this problem), the predicted pressure drops agree extremely well with RELAP5 predicting 1.6% more pressure drop at the highest flow, which has no boiling, and 4.2% more at the flow expected to be the minimum stable flow.

Table 3. Pressure Drops versus Flow from PLTEMP/ANL and RELAP5-3D for Benchmark Problem [15].

Flow, kg/s	PLTEMP/ANL		RELAP5			
	Pressure Drop*, kPa	Boiling	Pressure Drop*, kPa	Boiling	Gravity Pressure Drop, kPa	Exit Void Fraction, %
1.1800	185.8	no	188.8	no	5.80	0.0
1.0285	145.8	no	148.5	yes	5.79	0.0
0.9454	125.8	no	128.3	yes	5.78	0.0
0.7576	85.8	yes	87.9	yes	5.75	0.0
0.6476	65.8	yes	67.6	yes	5.73	0.0
0.6264	62.2	yes	64.8	yes	5.72	2.0
0.6000			62.4	yes	5.69	7.2
0.5600			62.0	yes	5.61	22.5
0.5400			63.6	yes	5.54	32.9
0.5300			65.0	yes	5.49	38.5
0.5250			65.9	yes	5.47	41.5
0.5200			No Stable Solution			
0.5194	45.7	yes				

*Inlet pressure minus outlet pressure, including effects of differences in elevation.

Two other observations can be made from Table 3. First, it can be concluded that since PLTEMP/ANL is a single-phase code, the presence of nucleate boiling has a minimal impact on the predicted pressure drops for the problem considered. Secondly, since W&F flow stability criterion was shown to be reasonably reliable [18], there is good reason to question the stable flow results predicted by RELAP5-3D Version 2.3 for flows below 0.6264 kg/s, the lowest stable flow according to W&F correlation. This conclusion is consistent with other studies [19, 20] showing similar difficulties for RELAP5/MOD2 and RELAP5/MOD3.2.

3.3. Comparison of RELAP5 and PARET/ANL

PARET was originally created in 1969 at what is now Idaho National Laboratory (INL) for predicting the course of nondestructive (events in which fuel or structural deformation does not occurs) reactivity accidents in small reactor cores in which there are no space-time effects on the neutron flux (i.e., point-kinetics where only the neutron flux magnitude changes with time). More specifically, the code was designed to analyze reactivity insertion transients in the light-water-cooled plate-type and rod-type cores tested in the SPERT-III reactor. The use of PARET is also appropriate for fuel assemblies with curved fuel plates when their radii of curvatures are large with respect to the fuel plate thickness.

The PARET/ANL code is derived from the original PARET and has been developed at Argonne National Laboratory (ANL) under the sponsorship of the U.S. Department of Energy/NNSA. It has been used by the GTRI Program (formally the RERTR Program) to determine the expected transient behavior of a large number of research reactors.

3.3.1. Comparison with RELAP5/MOD3 for IAEA Benchmark Transients

Ref. [21] compares RELAP5/MOD3.2 and PARET/ANL for different benchmark transients (loss-of-flow and reactivity insertion with scram) described in the IAEA Research Reactor Core Conversion Guidebook [22]. The benchmark transients each assume only a single mode of failure (period trip is ignored) before reactor scram is initiated. The amount of nucleate boiling present is very limited and the feedback from voiding is not a significant factor in the shutdown of the reactor.

More specifically, the characteristics of those transients are:

1. Fast loss-of-flow (LOF) transient: flow is reduced as $e^{-t/T}$ with $T = 1$ second, and reactor scram is initiated at 85% of nominal flow with a 200 ms delay before control insertion begins.
2. Slow LOF transient: the exponential decay as $T = 25$ seconds with the same scram conditions as in case 1.
3. Slow reactivity insertion transient: a $0.09/s$ ramp is inserted in a critical reactor at an initial power of 1 Watt. The safety system trip point is set at 12 MW with a time delay of 25 ms before control blade insertion is initiated.
4. Fast reactivity transient: β is inserted in 0.5 seconds. The scram conditions are the same as for case 3.

Figure 20 (extracted from Ref. [21]) shows that both codes predicts the same coolant and clad temperatures (within a few degrees) for the analyzed slow and fast LOF events in the benchmark. For all practical purposes the two codes are predicting the same results. Table 4 (extracted from Ref. [21]) summarizes the peak temperatures from the different participant to the benchmark for these LOF events.

Figure 20. Peak Clad Temperatures for Fast and Slow Loss-of-Flow Benchmark Transients [21].

Table 4. Peak Temperatures from All Participants: Loss-of-Flow Transients Benchmark [21].

LOF Transients	1.0s Pump Decay Peak Temperatures, °C			25.0s Pump Decay Peak Temperatures, °C		
	Fuel	Clad	Coolant	Fuel	Clad	Coolant
ANL, USA	90.3	87.5	60.3	86.8	83.7	58.8
Interatom, FRG	91.9	89.3	56.5	88.2	85.5	55.4
JAERI, Japan	NA	97.1	58.1	NA	96.1	57.5
JEN, Spain	95.4	93.9	59.3	91.9	90.3	58.1
PARET/ANL (Natural Convection)	91.7 (116.0)	88.9 (115.7)	60.3 (99.6)	88.0 (91.8)	84.9 (91.7)	58.8 (81.7)
RELAP5/MOD3 (Natural Convection)	91.7 (117.0)	88.8 (116.8)	59.0 (97.0)	87.8 (92.3)	84.6 (92.2)	57.7 (79.2)

Figure 21 (extracted from Ref. [21]) shows that the two codes are in good agreement for the slow and fast reactivity insertions as well. For the fast reactivity insertion, the largest difference occurs for the peak clad temperature, where RELAP5/MOD3 predicts a temperature 12 degrees higher than PARET/ANL. For the slow reactivity insertions, the differences are less than a degree. Table 5 (extracted from Ref. [21]) summarizes the peak temperatures from the different participant to the benchmark for these reactivity insertion events.

(a) Slow Reactivity Insertion (0.09\$ /sec)

(b) Fast Reactivity Insertion (1.5\$ in 0.5 sec.)

Figure 21. RELAP5/MOD3 and PARET/ANL Power and Peak Temperatures Predictions for Slow and Fast Reactivity Insertions [21].

Table 5. Peak Temperatures from All Participants: Reactivity Insertion Transients Benchmark [21].

Reactivity Insertion Transients	\$0.09/s Peak Temperatures, °C			\$1.50/0.5s Peak Temperatures, °C		
	Fuel	Clad	Coolant	Fuel	Clad	Coolant
ANL, USA	80.6	77.7	53.9	183.4	156.7	82.0
Interatom, FRG	80.8	78.1	51.1	185.3	168.2	63.2
JAERI, Japan	81.2	78.5	52.8	171.0	149.2	62.7
JEN, Spain	73.2	71.9	48.8	166.4	156.6	80.4
PARET/ANL	81.2	78.0	54.1	188.3	157.9	83.3
RELAP5/MOD3	80.8	77.8	53.2	196.2	169.8	82.0

The authors conclude that the overall agreement between the PARET and RELAP codes for this series of benchmark transients is excellent, and the results agree well with the earlier guidebook results. They also conclude that given the assortment of codes that were used in these studies, all of the results are in reasonably good agreement with each other for each of the transients considered.

4. ADDITIONAL CONSIDERATIONS

This section presents additional topics that are important to consider when performing certain analysis using RELAP5.

4.1. Analysis of Critical Heat Flux in TRIGA

Ref. [23] provides insight into the thermal-hydraulic analysis and the prediction of critical heat flux (CHF) for Training, Research, Isotopes, and General Atomics (TRIGA) reactors that rely on natural convection for primary flow. Since during normal operation these reactors may operate with subcooled nucleate boiling, the margin to CHF can be a limiting design criterion. At the time, that effort was coordinated with a parallel study by General Atomics (GA) [24].

A CHF analysis in a TRIGA requires the determination of the coolant mass flow rate under the natural circulation regime. Ref. [23] states that the chosen simulation approach where each sub-channel formed by the cusps of immediately adjacent rods is modeled as an isolated channel (i.e., no cross-flow between sub-channels) is reasonable and conservative.

The author mentions that the CHF correlations available in versions RELAP5-3D Version 2.3 and 2.4^c, i.e. the PG-CHF correlation and the 1986 Groeneveld Table, produces non conservative estimates of the rod power at which CHF occurs when compared to the Bernath [25] correlation traditionally used for predicting CHF in TRIGA reactors. Moreover, the author states that TRIGA pressure, heated length and rod diameter are out of range of the PG-CHF correlation.

The author proposes a methodology to use a more recent version (2006) of the Groeneveld Table but adds that even though this Table is often judged to be the best choice, it is not recommended to use it alone for calculating CHF in TRIGA reactors. This recommendation is based on the fact that the amount

^c It was verified that these correlations are available in RELAP5-MOD3.2 and RELAP5-MOD3.3.

of measured data is insufficient to determine a definitive conclusion as to the power level at which CHF occurs.

The author recommended approach^d for CHF analysis in TRIGA is to use RELAP5 for calculating the thermal hydraulics solution and use that information to evaluate the margin to CHF outside the code using 1) the Bernath correlation and 2) his proposed methodology for the use of the 2006 Groeneveld Table. The limiting rod power with respect to CHF should be the minimum of these two values.

Figure 22 (extracted from Ref. [23]) compares the predicted powers at which CHF occurs over a range of flow rates for different CHF correlations.

Even though this analysis focuses on evaluating the margin to CHF, the author also discusses an oscillatory flow phenomenon experimentally observed. As a mechanism for this oscillatory behavior, the author postulates that

- Subcooled boiling increases the buoyancy in the channel which increases the channel flow rate under natural circulation.
- The increased flow rate, with a constant channel power, causes a reduction in channel coolant temperatures, which causes the vapor to condense back to liquid.
- The buoyancy therefore decreases and causes the channel flow rate to decrease.
- This, in turn, causes the channel coolant temperatures to increase and the voids to reappear, thus completing the cycle.

This phenomenon should not be confused with the onset of flow instability predicted by the Whittle and Forgan correlation which occurs in fuel assemblies with parallel channels and under forced circulation.

These voids may then introduce reactivity effects (due to the negative void reactivity coefficient) which produce power oscillations that complicate the “chugging” phenomenon observed in a reactor. The author mentions that this phenomenon will occur prior to CHF.

The author also states that flow oscillations observed in a RELAP5-3D CHF analysis may be indicative of this phenomenon even though they may not accurately represent the flow behavior.

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^d This conclusion is consistent with the approach recommended by the author of Ref. [20].

Figure 22. Predictions for Hexagonal Pitch TRIGA Reactor Using Different Correlations and Codes [23].

The applicability of a various CHF correlations was further studied experimentally and analytically by the authors of Ref. [26]. They performed an experiment studying low-pressure, natural convection CHF with simulated (electrically heated) full fuel pins under TRIGA reactor conditions. The authors evaluated a series of correlations against their experimental data. They showed that the Bernath correlation used in Refs [23, 24] predicted their measured CHF values within $\pm 33\%$. More details about these experimental and analytical studies can be found in Ref. [27].

4.2. Boiling model

RELAP5/MOD3.2 is the parent code of both RELAP5-3D (developed and maintained by Idaho National Laboratory) and RELAP5/MOD3.3 (distributed by the Information Systems Laboratory, Inc. for the NRC). The RELAP5/MOD3.3 manual [28] states that a new subcooled boiling model was implemented in the code to consider low pressure situations.

This new model was assessed using data from the Shoukri subcooled flow boiling and condensation experiments [29]. Ref. [28] states that the Shoukri experiment 3c was performed under the following conditions: i) pressure of 0.153 MPa, heat flux of 0.7237 MW/m^2 , mass flux = 450.0 kg/s-m^2 and inlet subcooling of 26.25 K. Figure 23 (extracted from Ref. [23]) compares the measured and calculated axial void fraction for the Shoukri experiment 3c. It can be observed that low-pressure boiling model in RELAP5/MOD3.3 produces a much better agreement than the one implemented in RELAP5/MOD3.2.

Figure 23. Measured and Calculated Axial Void Fractions for the Shoukri Experiment 3c [23]

5. SUMMARY AND CONCLUSIONS

A summary of key RELAP5 assessments performed in relation to the conversion of research reactors to low-enriched uranium (LEU) fuel at ANL (or in collaboration with ANL) was developed. Even though it does not present a formal validation case for RELAP5 for all research reactors, this summary provides a strong technical basis to justify the use of the code for many applications.

Section 2 presented comparisons against experiments for a wide variety of research reactors (different flow regime, geometry, inlet pressure, power and temperatures) and applications (steady-state, reactivity insertion, loss-of-flow, startup and slow system transient). All comparisons showed good agreement between measured and calculated values under single-phase conditions for all versions of RELAP5 tested. Good agreement between the measured and calculated fuel temperatures for TRIGA operating with subcooled nucleate boiling was also shown. For fast reactivity insertion tested in the SPERT-I experiments, a marked departure from the measured data was observed for RELAP5/MOD3.2 when nucleate boiling becomes significant.

Section 3 presented comparisons with other codes (STAR-CD, PLTEMP, PARET/ANL and others) for different reactor systems and benchmark problems. The results presented included comparisons for cases at steady-state, during loss-of-flow events (fast and slow) as well as reactivity insertion events (fast and slow). Good agreements between all RELAP5 versions considered and the other codes was obtained for flow rates, temperatures (coolant, clad and fuel), power and pressure drops under single-phase and limited nucleate boiling conditions. References to work showing RELAP5 difficulties with predicting Onset of Flow Instability were provided.

Section 4 presented two additional considerations that are useful for the analysis of research reactors using RELAP5. It was shown that the critical heat flux (CHF) correlations included in RELAP5 result in a non-conservative estimate of power at which CHF occurs in TRIGA reactors with respect to the traditional Bernath CHF correlation. Until further comparisons with respect to the experimental data presented in Refs [26, 27] are performed, the level of conservatism with respect to measurements remains undetermined. It was also shown that for low-pressure system, the boiling model implemented in RELAP5/MOD3.3 is more accurate than the one in RELAP5/MOD3.2 and RELAP5-3D.

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