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Loss-of-Flow Simulations for the Conversion of BR2 to Low Enriched Uranium Fuel

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ABSTRACT

The RELAP5/Mod 3.3 code is being utilized for the safety analyses of a representative core configuration of the BR2 research reactor for conversion from Highly-enriched to Low Enriched Uranium fuel (HEU and LEU, respectively). Flow distribution and pressure drop data from a mock-up facility and the BR2 reactor, both with several core configurations, were used to determine generic minor loss coefficients for the reactors flow channels. The model was then applied to the 1963 BR2 loss-of-flow experiments and shown to obtain good agreement with the measured peak cladding temperatures. Following this, the model was applied to the representative core and used to predict cladding temperatures for HEU and LEU fuel at nominal conditions. Future work includes repeating the 1963 BR2 loss-of-flow experiment conditions with the representative core configuration for both fuel types.

1. Introduction

Belgium Reactor 2 (BR2) is a research reactor used for radioisotope production and materials testing. It's a tank-in-pool type reactor cooled by light water and moderated by beryllium and light water. The reactor has a single primary loop containing a set of primary pumps and heat exchangers to control the coolant inlet temperature (~30^{o}C) and flow rate (variable). A pressurizer maintains the inlet pressure at 1.38 MPa. In case of an accident, there are two isolation valves that can isolate the reactor core. A bypass valve can open to assist natural circulation flow through the core and a pool connection valve. The reactor core is located

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inside an aluminum pressure vessel that contains 79 channels in a hyperboloid configuration. Outside the core region the channels are stainless steel tubes; however, in the core region the channels are comprised of hexagonal beryllium with a central circular bore. The core configuration is highly variable as each channel can contain a fuel assembly, a control or regulating rod, an experimental device, or a beryllium or aluminum plug. Normally the coolant flow is downward and enters the channels through perforations in the channel walls located above the core region. The coolant leaves at the channel end at the support grid. Coolant can also bypass the core through gaps and cooling holes in the beryllium matrix.

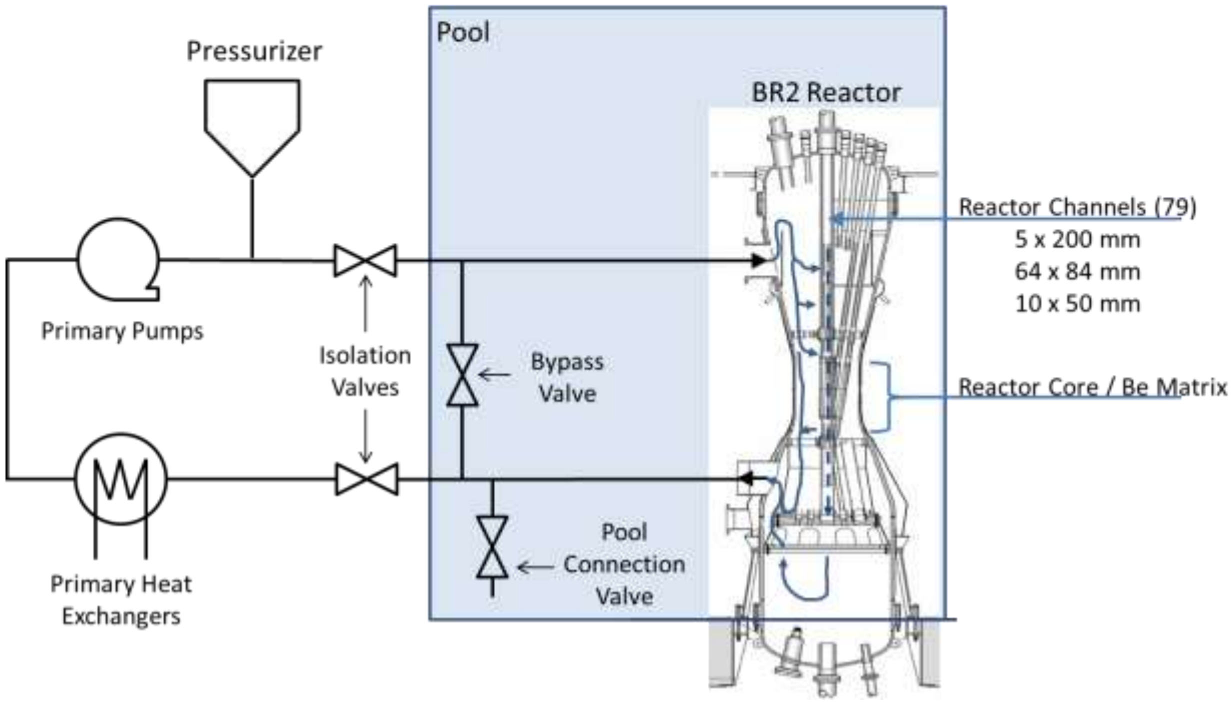

Figure 1. Conceptual drawing of the BR2 research reactor.

In support of converting the BR2 research reactor from Highly-enriched to Low Enriched Uranium fuel (HEU and LEU, respectively), the code RELAP5/Mod 3.3 [1] will be used to simulate several Loss-of-Flow Accident (LOFA) scenarios. Because the configuration of the core is variable, a representative core configuration has been defined for the fuel conversion analyses (Figure 2). However, in order to assure credibility in the predicted results, simulations have been performed for the 1963 core configuration for comparison with a number of LOFA experiments that were performed in the BR2 reactor. Additionally, experimental data was also available for steady-state flow and pressure distributions in both the BR2 facility (1962) and a hydraulic mock-up facility for various core configurations. This data was used to generate the minor loss coefficients applied to the RELAP5 model to obtain the proper channel flow rates and reactor pressure distributions for any given core configuration.

This paper presents the RELAP5 model, the method used to obtain the minor loss coefficients from the available hydraulic data, the LOFA simulation results compared to the 1963 experimental results, and the results for the representative core at nominal conditions for HEU and LEU fuel.

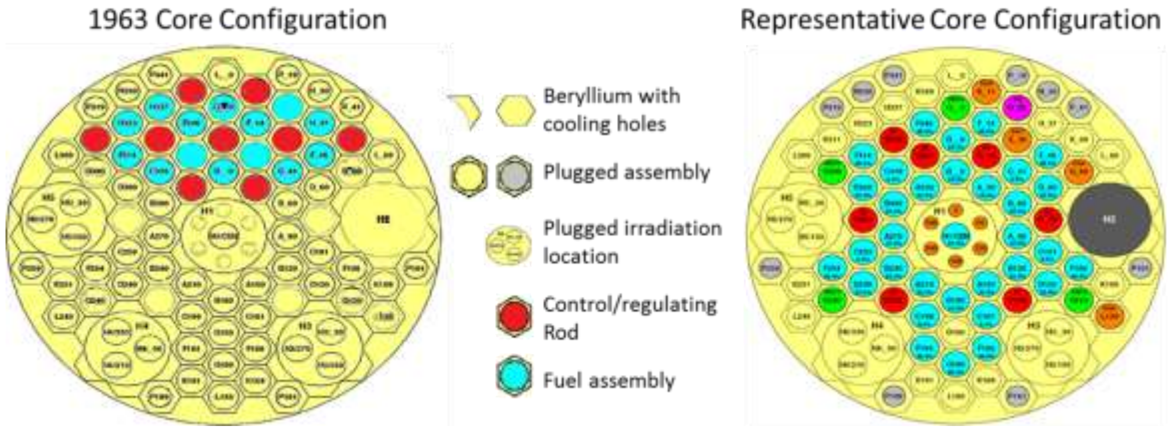

Figure 2. The 1963 and representative core configurations.

2. RELAP5 Model

In order to appropriately model the BR2 channels and maintain a simplified RELAP5 model, the 79 channels (80 including the bypass flow) have been consolidated into four representative flow channels (Figure 3); including the bypass flow (volumes 10-20), a single flow channel containing the highest heat flux fuel assembly (volumes 30-40), the remaining fuelled flow channels (volumes 50-60) and all remaining non-fuelled flow channels (volume 70-80). The volume containing the highest heat flux fuel assembly (volume 36) was further discretized (volumes 360, 365, 366, and 367) to better represent individual sub-channels associated with the fuel plates (Figure 4). Previous work has shown that 3 explicit sub-channels are sufficient to predict the peak cladding temperature in a LOFA simulation [2]. The sub-channels represent only a 10 degree arc of the sector to properly model the azimuthal power peak-to-average ratio. Computational fluid dynamic (CFD) simulations [3] have demonstrated the validity of this approximation for both normal operation and natural circulation since both azimuthal coolant mixing and azimuthal heat conduction in the fuel plate are relatively small.

The four flow channels have been discretized into six axial sections, three above the core and two below. Three of these sections have crossflow junctions with minor loss coefficients to represent the channel perforations. The core region was further discretized into 22 axial volumes of which the middle 20 are associated with the region of the fuel meat.

Heat structures have been created for most of the primary piping within the pool, the reactor vessel and its internal components and the fuel channels and their internal contents. Three radial nodes were used for discretizing all structures, except the fuel, which contained nine; three in the fuel and three in each of the inner and outer cladding. The pool was approximated by connecting the associated heat structures to a time dependent volume (infinite energy sink) maintained at 25^{o}C. An exception to this was the portion of the reactor vessel in contact with the cooling shroud (volume 166) which was operated with a 33^{o}C inlet temperature and mass flow rate of 111 kg/s.

The pressurizer was modeled with a time dependent volume to obtain the reactor inlet pressure at steady-state conditions. An isolation valve was added to the pressurizer so that it could be disconnected for loss of pressure simulations.

The secondary side of the heat exchanger was modeled as a temperature boundary condition to obtain the reactor inlet temperature at steady-state conditions.

The primary pump homologous curves were based on the BR2 pump data. The 1963 pump coast down data was replicated by calibrating the torque friction coefficients in the RELAP5 model. Good agreement was obtained above 10% of nominal flow. Below this, differences were attributed to measurement uncertainty. Results from a sensitivity study show the uncertainty in flow impacts the time that flow reversal in the core occurs but has little effect on the magnitude of the peak cladding temperature.

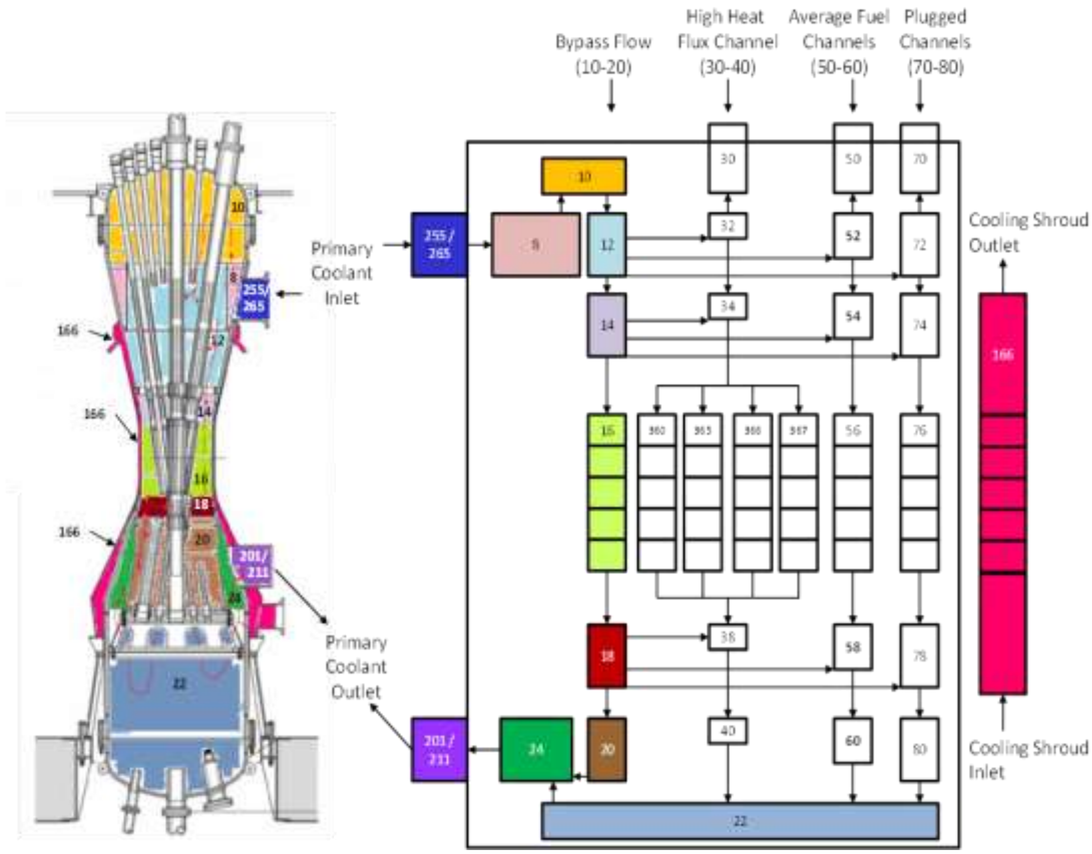

Figure 3. RELAP5 model of the reactor vessel coolant volumes

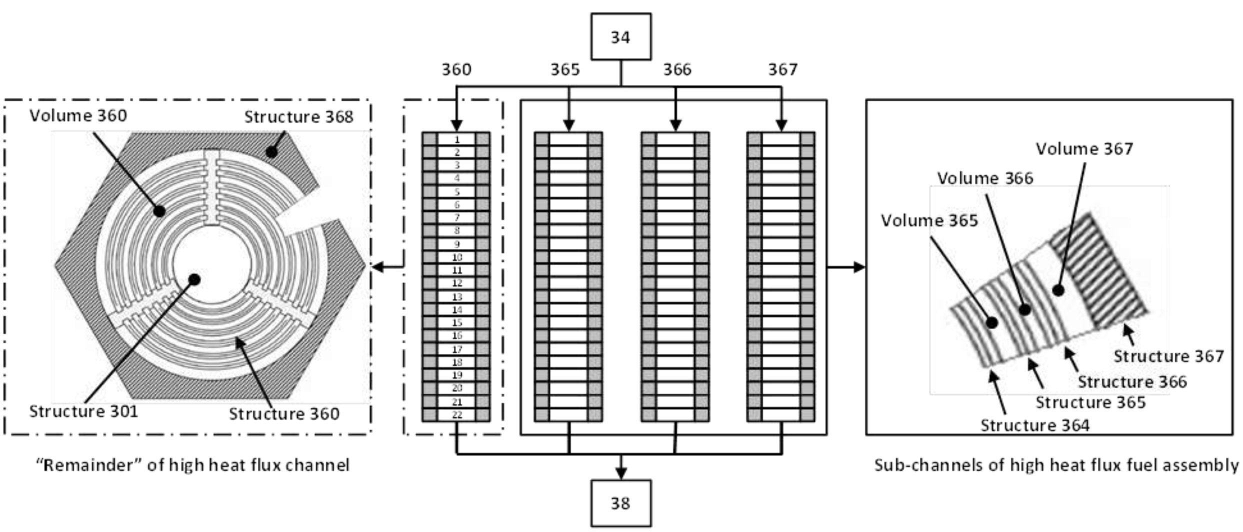

Figure 4. RELAP5 model of the high heat flux fuel element.

3. Model Calibration

Experimental data has shown that the flow rate per channel type is essentially independent of core configuration and the values presented in Table 1 are valid for a core pressure drop of 2.1 kg/cm^2.

Table 1. Flow rate per channel type for a pressure drop of 2.1 kg/cm^2 across the core.

Channel (diameter)	Contents	Flow Rate
Standard Channel (84 mm)	with beryllium plug	4.3 kg/s
	with fuel element	35.7 kg/s
	with control rod	7.4 kg/s
H Channel (200 mm)	with beryllium plug	22.5 kg/s
Reflector Channel (50 mm)	with beryllium plug	2.3 kg/s
Bypass Channel		361.6 kg/s

In order to obtain the proper flow distribution and pressure drop for the core in the RELAP5 model, minor loss coefficients were added to coolant volumes 16, 360, 365, 366, 367, 56 and 76. These minor loss coefficients were determined by utilizing a subset of data from the BR2 1962 tests for several core configurations. Both a RELAP5 model and a separate core hydraulic model based on steady state equations were created to determine the minor loss coefficient for each core configuration that was evaluated. The RELAP5 model solution without minor loss coefficient was used as the input to the core hydraulic model. Utilizing these results and the target flow rate and pressure drop values, the core hydraulic model was used to solve for the required minor loss coefficients. By repeating this process for several of the 1962 core configurations an average minor loss coefficient was determined for each channel. Figure 5 shows that the RELAP5 models for selected core configurations produce good agreement with the 1962 data for core pressure drop and flow rate.

Comparison of data to the calculated pressure distribution indicated that the pressure loss with RELAP5 volumes 14, 34, 54, 74 was slightly under predicted. To correct this, the calibration process was repeated at this elevation to obtain good agreement with the measured data. A final minor loss coefficient was applied to the outlet piping to obtain better agreement with the overall pressure drop measured between the inlet and outlet piping.

As is, the RELAP5 model could not be extended to the 1963 and representative core configurations since the minor loss coefficient is not a constant for channel 76 (plugged channel); its contents are a consolidation of a variety of components with different geometry. As shown in Figure 6, the hydraulic diameters for core configurations of interest are much larger than the 1962 core configurations. In order to bridge the gap between the representative and 1962 cores the calibration process was repeated for the core configurations studied in the BR2 hydraulic mock-up facility. From this, a correlation based on the Darcy-Weisbach equation[1] was created to describe the minor loss coefficient for channel 76.

Returning to Figure 5, it can be seen that the RELAP5 model for the 1963 and representative core configuration produce the expected core pressure drop (2.1 kg/cm^2) at nominal conditions.

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[1] $\Delta p = \left(k + f_D \frac{L}{D_h}\right)\frac{\rho V^2}{2}$; where the minor loss coefficient (k) was determined to be $k = \frac{487.4}{\left(\frac{\dot{m}}{996 \cdot A}\right)^2} - \frac{20.9}{D_h}$

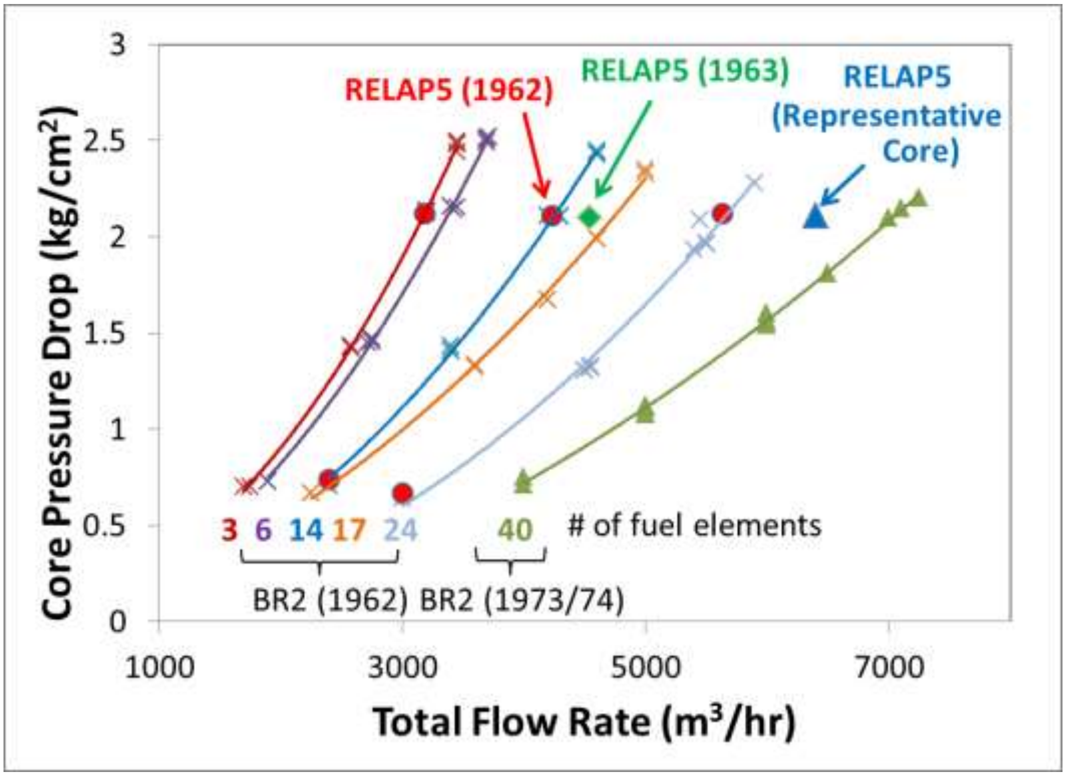

Figure 5. Comparison of measured data with RELAP5 calculated pressure drop vs. flow rate for different core configurations.

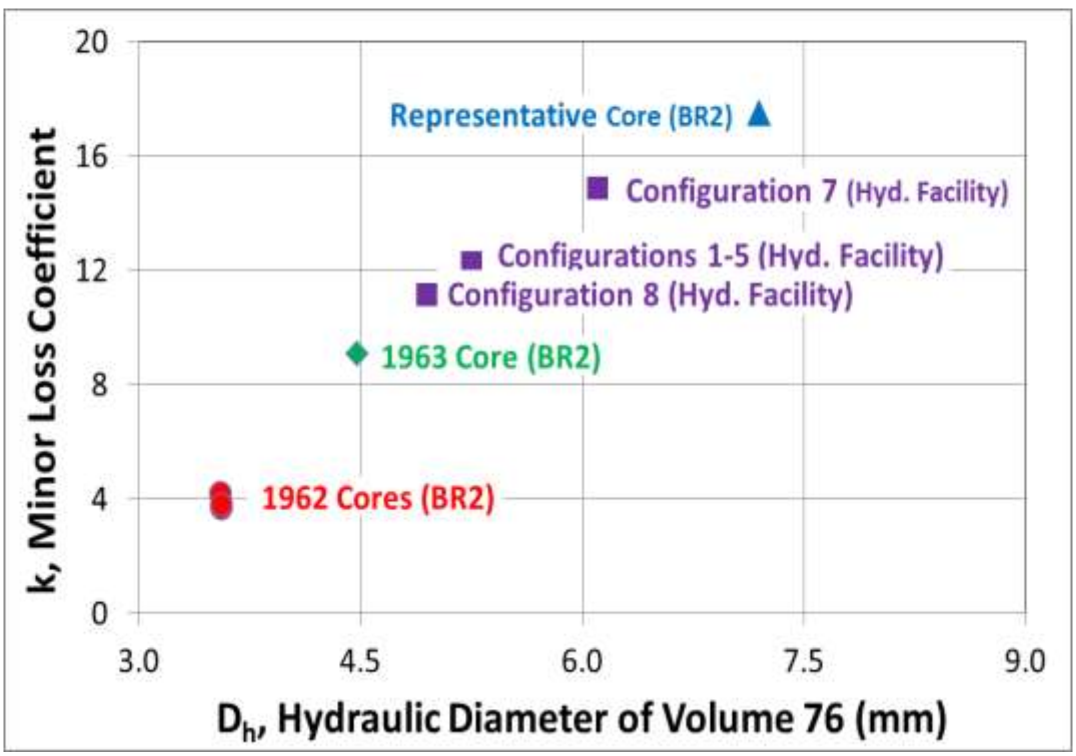

Figure 6. Minor loss coefficients determined for channel 76 (plugged channel).

4. 1963 LOFA Simulations

To obtain credibility for the RELAP5 model of BR2 for fuel conversion analyses, simulations were performed for a subset of BR2 LOFA experiments with the following characteristics:

Test A/400/1, loss of flow at maximum heat flux of 400 W/cm^2

Test C/600/3; loss of flow at maximum heat flux of 600 W/cm^2

Test F/400/1; loss of flow with loss of pressure at maximum heat flux of 400 W/cm^2

The steady state and transient power distributions for the RELAP5 model were obtained from MCNP and ORIGEN calculations described in [4]. Of note is that peak heat flux determined from MCNP analyses utilizing measured reactor power was approximately 310 W/cm^2 instead of 400 W/cm^2 for test A. Similar differences in heat flux were calculated for tests C and F. However, as will be shown later, the measured and RELAP5 calculated cladding temperatures are in good agreement for steady state conditions and indicate the validity of the MCNP calculated heat fluxes. For the transient analyses, neutronics calculations took into account the redistribution of energy following a scram. The transient power distribution calculated for test A is given in Table 2; the values are similar for tests C and F.

Table 2. Power distribution for BR2 LOFA test A.

Zone	Steady-State	Transient		
		0.1 s	25 s	50 s
Fuel	0.959	0.824	0.744	0.718
Beryllium	0.026	0.112	0.163	0.180
Other	0.015	0.064	0.093	0.103

The RELAP5 model for Test A was characterized by a loss of power to the main pumps at 5.35 s followed by a reactor scram at 7.7 s. The bypass valve started to open at 22.0 s and was completely open at 35.6 s. All other parameters were unchanged.

Of particular interest in this work is the comparison of the RELAP5 simulation results to the measured fuel cladding temperature of the instrumented fuel assembly. Thermocouples were located at 300 mm (TC11), 150 mm (TC12), 0 mm (TC13) and -150 mm (TC14) from the fuel centerline at the outer face of plate 6. The results for test A are shown in Figure 7. Reasonable agreement was obtained for both the steady-state (time $<$ ~5 s) and the transient cladding temperature at all axial locations. The agreement at steady-state indicates that the heat flux profile in the experiment and that used in RELAP5 simulations are quite similar. Following the scram, the power decreases rapidly and the coolant flow rate remains sufficiently high to reduce the fuel element temperature to near the coolant temperature. As the pump speed approaches zero and the coolant stagnates, it begins to acquire sufficient decay heat to increase in temperature; this causes the cladding temperature to increase with time. The cladding temperature reaches a peak value very near the time a balance is reached between the friction and buoyant forces; in other words, the time at which reversed flow is established. Once the coolant velocity is established heat removal is again effective and the peak cladding temperature decreases. It is interesting to note that the RELAP5 simulations predict a secondary peak cladding temperature approximately 7 s following the first and that this is not present in the data. In the RELAP5 simulations, the two peaks are the result of the difference in time that flow reversal is established in volumes 366 and 367. Similar results have been obtained in previous simulations [5] and further work is required to understand the reason for this difference. Simulations have also been performed for tests C and F but have been omitted here since reasonable agreement was obtained for each.

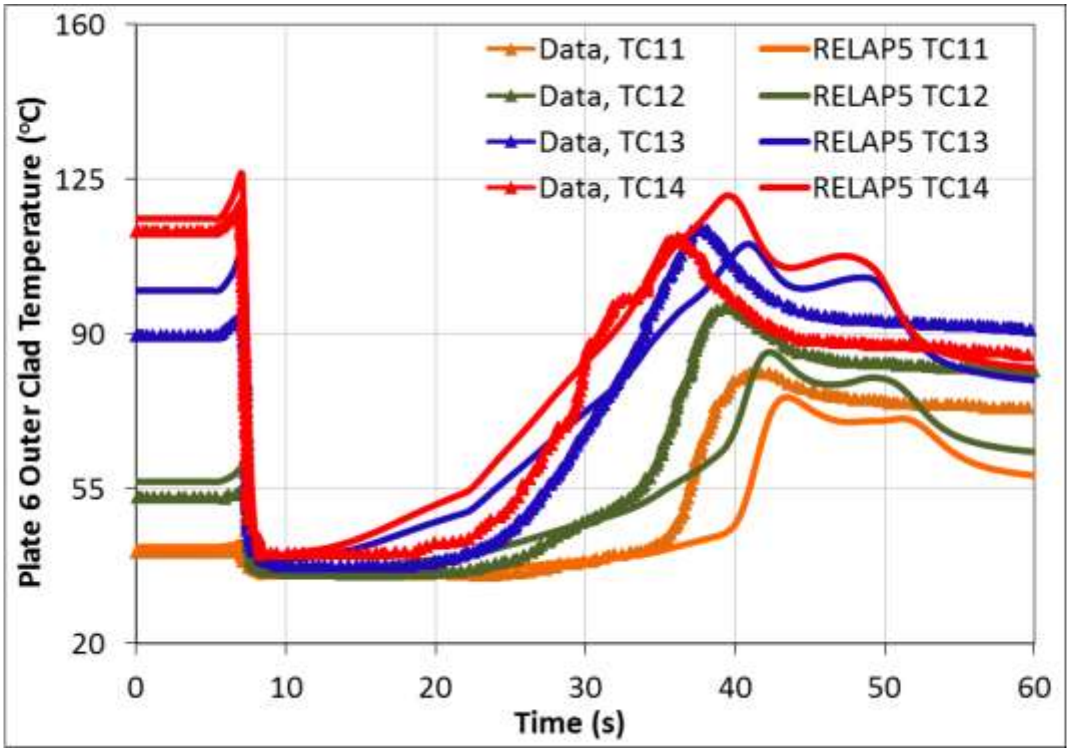

Figure 7. Comparison of RELAP5 clad temperature with BR2 data for test A.

5. Representative Core

Following the successful comparisons with the 1963 BR2 experimental data, the RELAP5 model was updated to the representative core configuration shown in Figure 2. Simulations were performed for steady state conditions at a peak heat flux of 400 W/cm^2, 470 W/cm^2 and 600 W/cm^2, located at the hot stripe; utilizing the power distributions calculated by MCNP for the representative core configuration. A heat flux value of 470 W/cm^2 was included since the maximum allowable heat flux has been increased from 400 W/cm^2 [6]. Figure 8 shows that the cladding temperature obtained for the HEU and LEU power distributions are quite similar for a given heat flux (±2 ^{o}C for the peak) and that location at which the peak occurs for LEU is ~4 cm higher than calculated for HEU. It should be noted that boiling is found to occur at a single axial node for the HEU fuel at a peak heat flux of 600 W/cm^2. This indicates that the representative core configuration is near the limit for onset of nucleate boiling at this heat flux. Based on these results it is expected that the peak cladding temperature during the transients will be quite similar for both HEU and LEU fuel.

6. Future Work

Similar to the work performed for simulating the 1963 BR2 LOFA experiments, MCNP and ORIGEN calculations are required to obtain the transient power distributions for the representative core configuration. LOFA simulations with RELAP5 will be repeated for the test conditions defined for tests A, C and F. Tests A and F will also be repeated with a peak heat flux of 470 W/cm^2 as this is the current limiting heat flux.

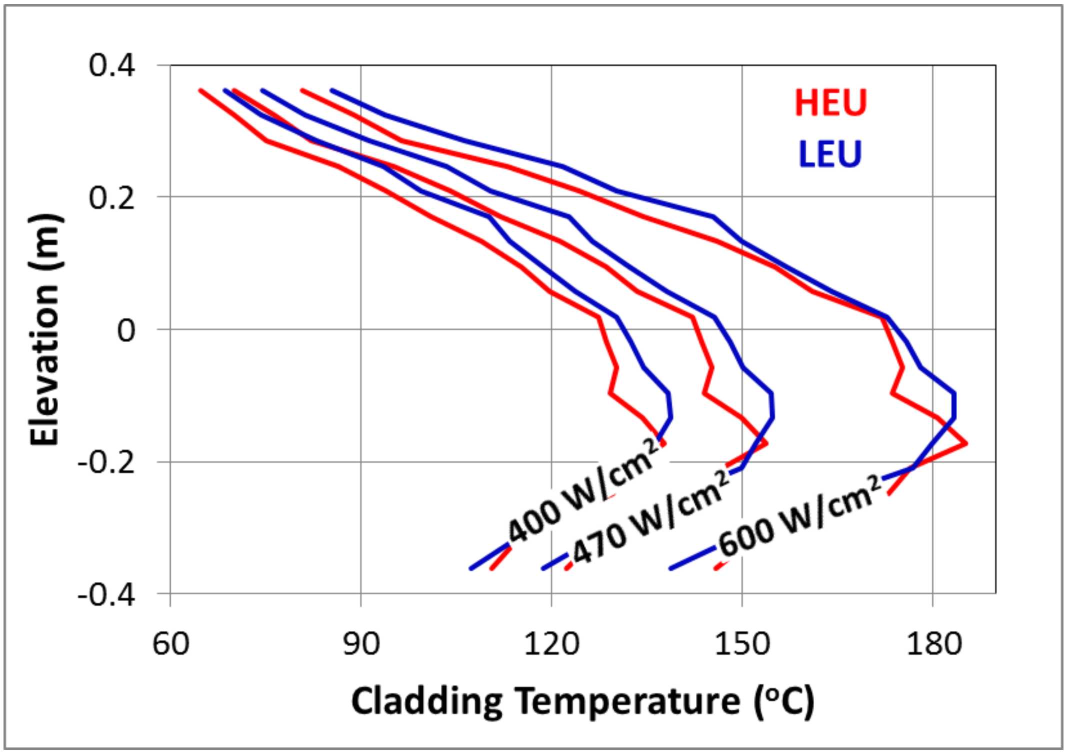

Figure 8. Steady-state cladding temperatures for HEU and LEU fuel in the representative core configuration for target heat flux values.

7. Summary

The purpose of this work is to develop a RELAP5 model of the BR2 research reactor and evaluate the impact of LEU fuel conversion by performing LOFA simulations and demonstrating that there is adequate margin to boiling, or if boiling would occur for a short time, identify when the cladding temperature reaches a peak value and show that it would not affect the integrity of the fuel.

The model geometry has been updated to reflect current reactor design drawings and component changes that have occurred over the life time of the reactor. This means that most components in the RELAP5 model will not require changes throughout the conversion process. For those components that do change due to core configuration, the model setup and calibration is now sufficiently established such that any changes can be easily included without impacting model credibility. The model also utilized a hot stripe approach to better capture the effect of azimuthal power peaking.

A significant effort in this work was the determination of the minor loss coefficients for the RELAP5 model. Rather than directly calibrating the RELAP5 model to the 1963 experimental data, comparisons were made to a subset of the detailed flow and pressure distribution data available from the 1962 BR2 and hydraulic mock-up facility flow tests. The fact that the average minor loss coefficients determined from several core configurations produces flow rate and pressure distribution values that agree well with the experimental data is significant as it established credibility for the predicted flow and pressure distribution for the representative core configuration.

The reasonable agreement obtained for the evolution and magnitude of the cladding temperatures between RELAP5 simulations and the experimental data for the 1963 loss-of-flow and loss-of-pressure accident simulations provide verification for the modeling approach.

RELAP5 models have been adapted to simulate the BR2 representative core configuration with HEU and LEU fuel. Steady state simulations have been performed and indicate similar behavior for each fuel type (±2 ^{o}C for the peak temperature) with the peak occurring ~4 cm higher for the LEU fuel. The fact that boiling is found to occur at a single axial node for the HEU fuel at a peak heat flux of 600 W/cm^2 is an indicator that the core configuration is near the limit for onset of nucleate boiling at this heat flux. The remaining items required to complete this conversion analysis includes determining the transient power distribution for each fuel and repeating the LOFA simulations performed in 1963. Based on the results presented here it is expected that the peak cladding temperature during the transients will be quite similar for both HEU and LEU fuel. The peak cladding temperature reached during the LOFA transients has yet to be calculated.

8. References

[1] Information Systems Laboratories Inc., 2003. RELAP5/Mod3.3 patch 03 Code Manual.

[2] C.P. Tzanos and B. Dionne, "Analysis of the BR2 Loss-of-Flow Test A," Nucl. Technol., 179. 3 (2012).

[3] B. Dionne and J.G. Stevens, "Applicability of a hot channel factor-based hot stripe approach to model the azimuthal power peaking in a BR2 fuel assembly", 33rd International Meeting on Reduced Enrichment for Research and Test Reactors, October 23-27, 2011.

[4] B. Dionne and C.P. Tzanos, "Estimation of Steady-State and Transient Power Distributions for the RELAP Analyses of the 1963 Loss-of-Flow and Loss-of-Pressure Tests at BR2" ANL/RERTR/TM-11-19, (2011).

[5] C.P. Tzanos and B. Dionne, "Analysis of the BR2 Loss-of-Flow Test A," Nucl. Technol., 179. 3 (2012).

[6] "Volume 2: BR2 operation and operational safety," SCK•CEN dossier Vol VI; 72S7680 : Reactor BR2, vol 2-4-2, SAR001-3.1-ed 3, SCK•CEN, (2007).