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ANALYSIS OF THE JAMAICAN SLOWPOKE-2 RESEARCH REACTOR FOR THE CONVERSION FROM HEU TO LEU FUEL

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ABSTRACT

The Jamaican SLOWPOKE-2 (JM-1) is a 20 kW research reactor manufactured by Atomic Energy of Canada Limited that has been operating for 30 years at the University of the West Indies, Mona Campus in Kingston, Jamaica. The University, with IAEA assistance under the GTRI/RERTR program, is currently in the process of converting from HEU to LEU. Full-reactor neutronic and thermal hydraulic analyses were performed, using MCNP5 and PLTEMP/ANL v4.1 respectively, on both the existing HEU and proposed LEU core configurations. Although conversion will result in the full nominal reactor power increasing from 20 kW to approximately 22 kW, in order to maintain the 10^{12} $n \cdot cm^{-2} s^{-1}$ flux in the inner irradiation channels, and maximum fuel temperature to increase from ~82°C to ~113°C, the analysis illustrates that increased safety margins will be obtained. No significant reactor behavior changes are expected and the characteristic SLOWPOKE-2 reactor inherent safety features will be preserved.

1 Introduction

For 30 years, the JM-1 reactor, with a full power rating of 19.08 kW, has been primarily used for neutron activation analysis of trace elements in health, environment and agriculture studies. In 2009, a formal request was made via the IAEA to the Global Threat Reduction Initiative (GTRI) and the Reduced Enrichment for Research and Test Reactors (RERTR) programs to convert JM-1 to LEU fuel and to ship the spent fuel to the U.S. where the uranium was enriched [1]. The 296 fuel pins of the JM-1 HEU core are made of a coextruded U-Al alloy (28wt% U) enriched to ~93% (^{235}U mass of ~827 g) and Al cladding, which is also the fuel cage material. The core is encased in a 102 mm-thick, 228 mm-high annular beryllium reflector and sits on a bottom 102 mm-thick beryllium (Be) disc 322 mm in diameter. The top reflector consists of semicircular beryllium plates called shims, each only a few millimeters thick, which are added as necessary to compensate for burn-up. Five inner irradiation sites are located in the annular reflector and one outer site sits just outside the annulus. All the components of the HEU core assembly will be reused, without modification, with the new LEU core, with the exception of the fuel pins and fuel cage. The proposed LEU core for the JM-1 reactor will have 198 fuel pins manufactured from Zircaloy-4, like the fuel cage, and filled with sintered UO_2 pellets (enriched to 19.86%), resulting in a total U mass of ~5600 g and ^{235}U mass of ~1100 g. The LEU configuration will also include one self-powered cadmium flux detector in irradiation site #2 and a non-operational one that got stuck inside the annular Be reflector shortly after commissioning.

This paper presents the neutronic and thermal hydraulic safety analyses results of the JM-1 HEU to LEU conversion, which should be completed by September 2015, now that the fuel fabrication process is well advanced. The main objective of this study was to create and validate neutronic and thermal hydraulic models of the JM-1 HEU reactor, subsequently adapting them to the new LEU configuration, and to perform all necessary analysis to ensure that the core conversion will not adversely affect the overall behavior, performance or inherent safety features of the reactor.

2 Neutronic analysis

2.1 HEU core reactivity

A detailed model of the JM-1 HEU reactor was created, with MCNP5, using data from engineering drawings of the reactor components. This model was based on the HEU core configuration at commissioning, though it was not entirely possible to exactly capture all features of the reactor in the model. This is because no reliable data was available on the precise uranium vector (isotopic composition) and the impurities present in the original HEU fuel. The fuel composition used in the HEU model was from the IAEA's Research Reactor Core Conversion Guidebook (RRCGB) [2], which presented the typical composition of U-Al fuels enriched to 93.19% ^{235}U, almost identical to the 93.18% enrichment used in the JM-1 reactor. The HEU reactor, as modeled in MCNP5, is illustrated in Figures 1 and 2.

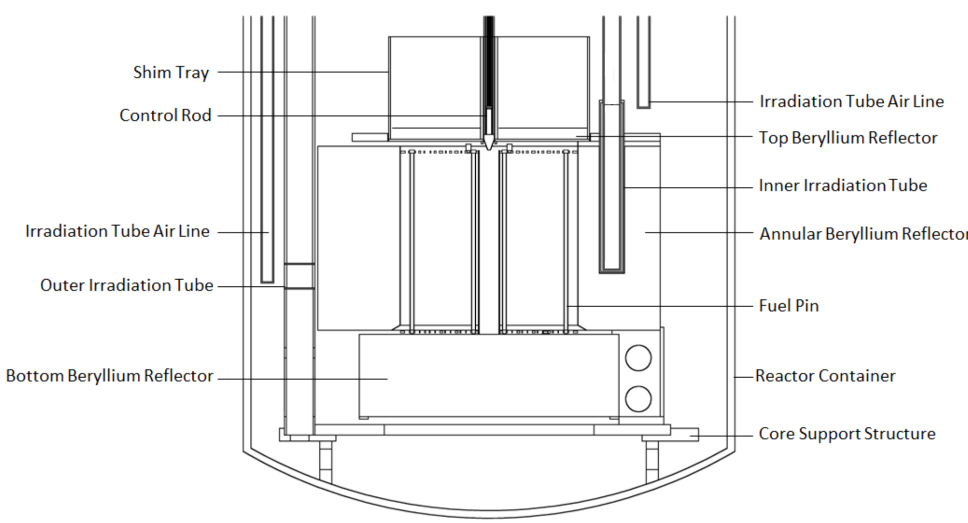

Figure 1: Vertical cross-section of the JM-1 reactor model

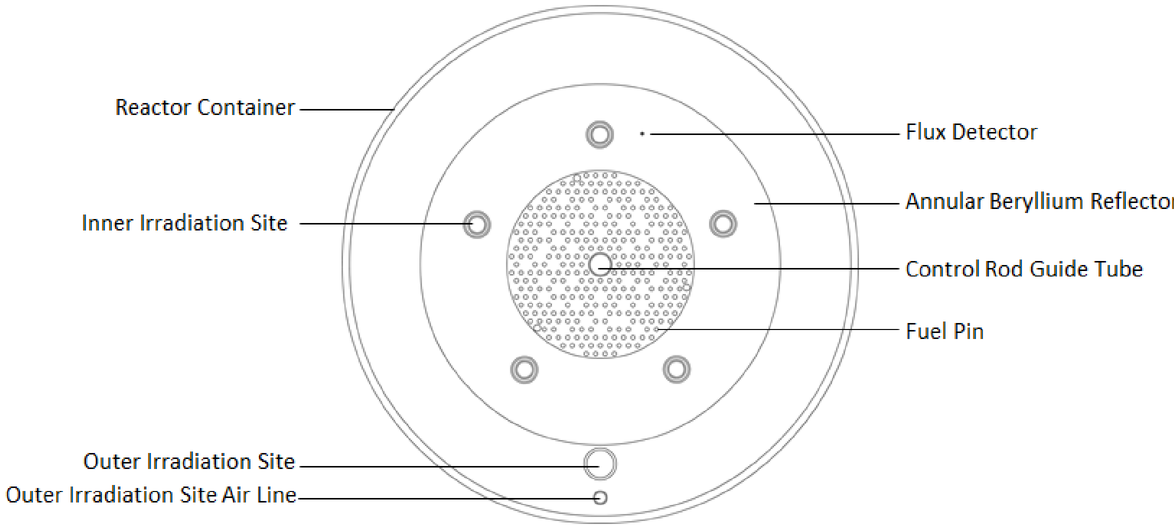

Figure 2: Horizontal cross-section of the JM-1 HEU reactor core model

The MCNP5 estimated HEU core excess reactivity was 4.97 ± 0.04 mk (1σ used throughout the paper), 1.57 mk higher than the 3.4 mk of excess reactivity installed at commissioning. This discrepancy is likely to be partially due to the uncertainties in the fuel composition and in some Be reflector impurity concentrations reported as being below detection thresholds.

2.2 LEU core reactivity

As mentioned, the only reactor changes during the conversion will be the fuel, fuel cladding and fuel cage material. The model also includes the additional flux detector, as illustrated in Figure 3.

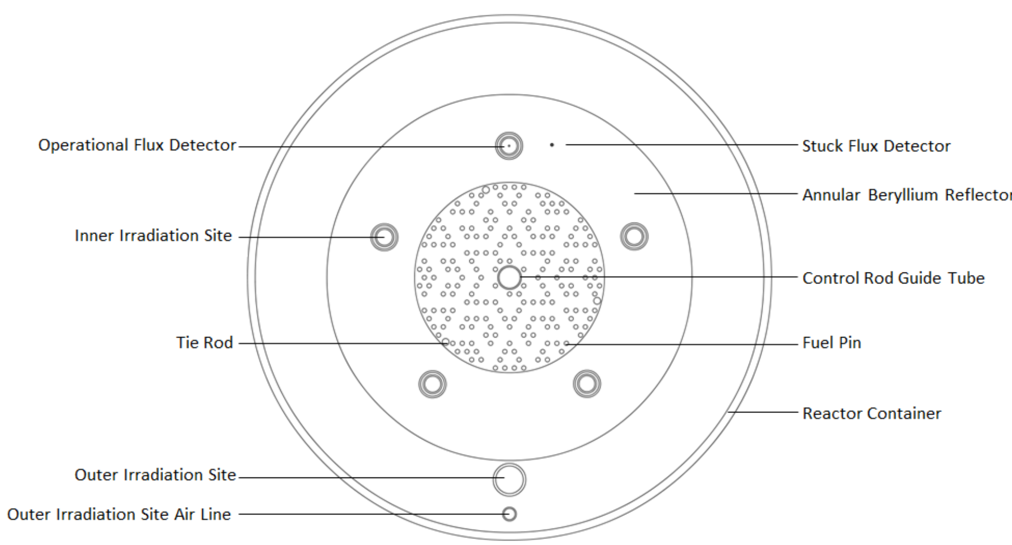

Figure 3: Horizontal cross-section of the proposed JM-1 LEU reactor core model

The estimated excess reactivity of the LEU core was 4.59 ± 0.04 mk with the beryllium impurities concentrations set to 66% of the detection limits. The resulting excess reactivity slightly overestimates the commissioning value of 3.69 mk at the HEU to LEU converted

SLOWPOKE-2 reactor at École Polytechnique de Montreal (EPM) [3]. However, the EPM reactor has four additional outer irradiation tubes, with an estimated reactivity worth of approximately −1 mk, resulting in excellent agreement. Still, some other small differences may exist between the reactors, such as reactor material impurities, that were not accounted for due to a lack of reliable data. Their reactivity influence is expected to be less than 1 mk, therefore the agreement is remarkable in any case. Prior to this work, the best agreement between calculations and SLOWPOKE-2 LEU measurements had been an overestimation of 6 mk [4].

2.3 Coefficients of reactivity

Accurate excess reactivity estimates contribute to the overall model confidence, though its most important role lies in its capability to accurately replicate the reactor's dynamic behavior determined by reactivity changes. These changes, in absolute terms, are less influenced by slight composition differences and other possible small model inaccuracies than total reactivity is.

2.3.1 Moderator void coefficient of reactivity (MVCR)

Void generation within the core as a result of the coolant-moderator boiling would displace moderator, decreasing effective moderator density and increasing neutron leakage, thus reducing the system reactivity. The HEU core average MVCR estimated by MCNP5 was −3.132 mk/% of void fraction. This is equivalent to −0.041 mk/cm^3 (core water volume of ~7.7 L) which compares very well with the previously calculated value of −0.046 mk/cm^3 and the measured −0.042 mk/cm^3 at the center of a SLOWPOKE-1 reactor core using a 7 cm^3 void capsule [5].

The average calculated MVCR for the LEU core is −2.62 mk/% of void fraction which is equivalent to −0.034 mk/cm^3 (~8.2 L). This compares very well with the previously calculated value of −0.033 mk/cm^3 and measured value of −0.036 mk/cm^3 in the EPM LEU reactor [6].

2.3.2 Fuel temperature coefficient of reactivity

The negative fuel temperature coefficient is essentially associated to the increase in parasitic neutron capture by ^{238}U due to Doppler broadening of its neutron absorption resonances. As expected in a HEU-fueled reactor, with a small ^{238}U isotopic fraction, the corresponding neutron absorption increase with rising temperature is also small, resulting in a negative but essentially negligible fuel temperature reactivity coefficient: −0.0009 mk/°C according to MCNP5 results. Correspondingly, given that the LEU fuel is predominantly ^{238}U, the Doppler broadening effect will be more evident. The calculated fuel temperature coefficient for the JM-1 LEU core is −0.008 mk/°C compared to a calculated value of −0.009 mk/°C for the EPM reactor [6].

2.3.3 Combined temperature reactivity response

Experimentally, in the HEU core, it is found that the reactivity increases with temperature and peaks at 19.4°C, then decreases with further temperature increases. Any sustained reactor operation, even at low power (<2 kW), will readily increase moderator temperature above 19.4°C causing the reactor temperature coefficient to become negative at operating conditions. The MCNP5 estimated reactivity behavior as a function of temperature is shown in Figure 4.

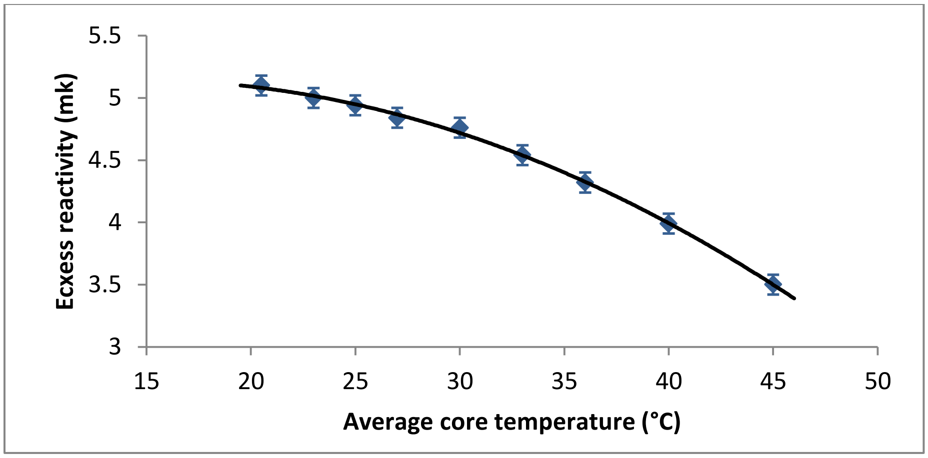

Figure 4: Calculated excess reactivity as a function of core average temperature in the HEU core

The simulation's lowest temperature is limited by the cross section libraries' lowest available temperature, which is 20.45°C. Cross section libraries can be generated at higher temperatures by means of Doppler broadening, but that's not possible in the reverse direction. Nonetheless, the results show a continuous decrease in reactivity with increasing core temperature beyond 20.45°C and the trend is compatible with the experimentally observed maximum at 19.4°C.

The combined temperature feedback of the LEU was also estimated using MCNP5. All the experimental data as well as the MCNP5 simulation results were shifted to the same maximum excess reactivity to allow for an easier and more direct comparison without the interference of small inaccuracies in the total reactivity calculations or different initial conditions at the start of the each experiment. Given that what is relevant, in terms of reactivity coefficients, are the relative variations of the reactivity as a function of temperature, this shifting does not affect the accuracy of the analysis. The resulting comparison in Figure 5 shows that the calculated LEU reactor behavior fits experimental data very well. A previous reactivity curve calculation attempt

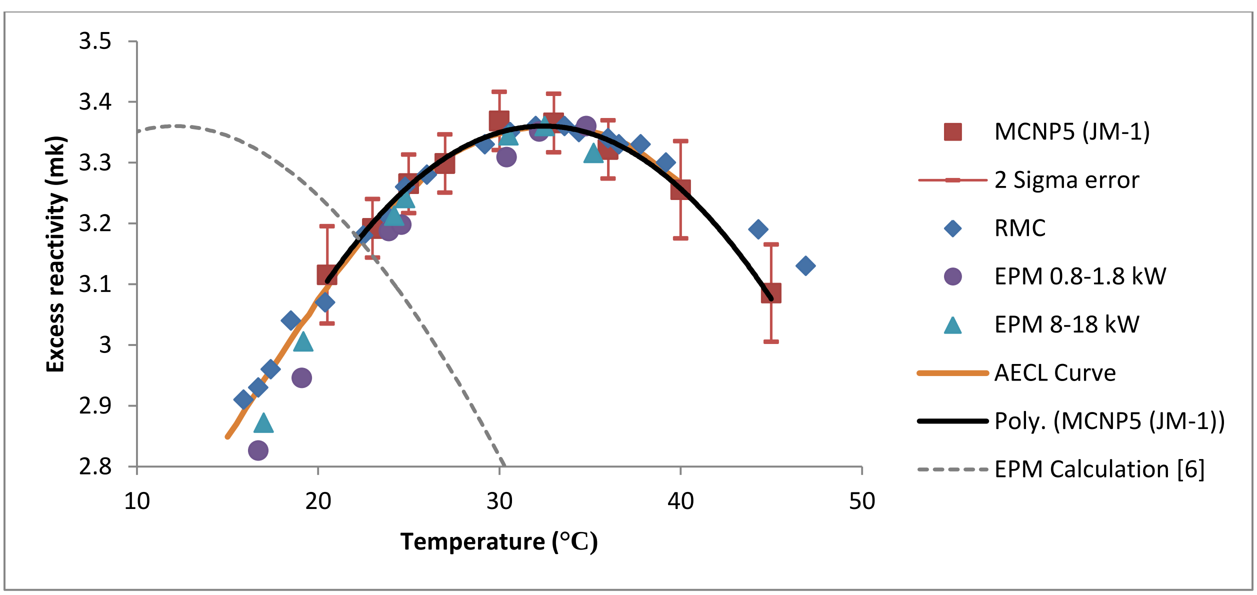

Figure 5: Comparison of calculated reactivity versus reactor temperature and experimental data

at EPM is also provided for comparison. RMC refers to the LEU SLOWPOKE-2 reactor at the Royal Military College of Canada.

2.3.4 Flux distribution in inner irradiation channels

To maintain a flux of 10^{12} n·cm^{-2} s^{-1} in the inner irradiation channels with the HEU to LEU conversion, a ~14% power increase is required, resulting in a full nominal power of 21.72 kW. A similar increase was also necessary at EPM, resulting in a ~15% increase (17.2 kW to 19.84 kW) [7,8]. The HEU and LEU inner channels flux distributions are compared in Figure 6.

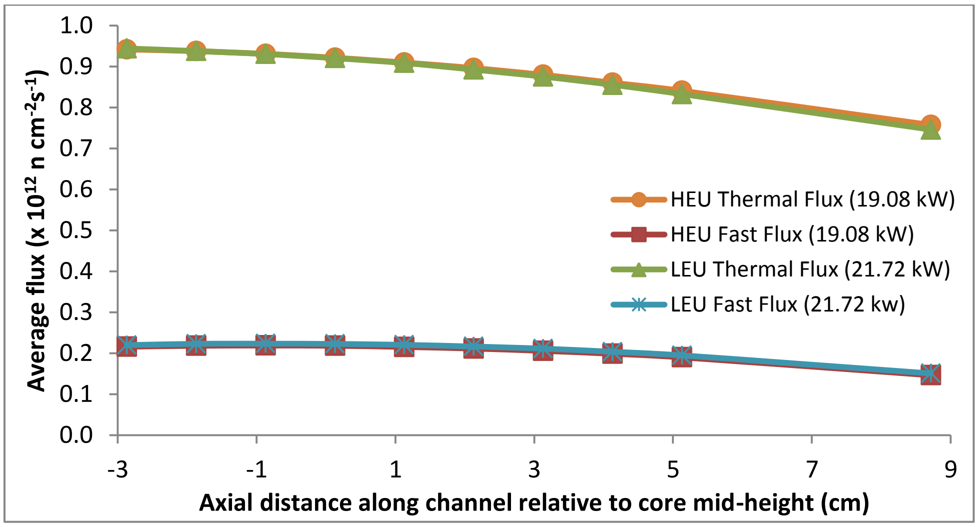

Figure 6: Calculated flux distribution in the inner irradiation channels at nominal power

3 Thermal Hydraulic Analysis

The purpose of the thermal hydraulic analysis was to determine the reactor safety margins and maximum allowed operating power. Given that this is a natural circulation reactor and significant flow and heat transfer uncertainties exist for natural circulation two-phase flow, the maximum allowed power is conservatively assigned to be the power at which the Onset of Nucleate Boiling (ONB) occurs, even though heat transfer will still increase substantially beyond that point.

The thermal hydraulic analysis was performed using PLTEMP/ANL v4.1 code. Among other model parameters, it requires the coolant minor losses coefficient and the estimated friction coefficient (*f.Re*), which is the product of Darcy friction factor (*f*) and Reynolds number (*Re*) [9], to calculate the flow. Typically, for similar reactor types with a uniform core lattice, a single channel or fuel pin model with the appropriate representative average properties of the core is used. The model is subsequently calibrated with experimental data of coolant temperature evolution as a function of reactor power to obtain the minor pressure loss coefficient suitable to accurately simulate the coolant flow at different operation regimes. Similarly, another one-pin model of the hottest fuel pin is later used to determine the most severe fuel operating conditions and calculate the corresponding safety margins. In a uniform core, flow area, heated perimeter and friction coefficient values are essentially the same for all pins or flow channels. The innermost and outermost rings would be an exception but the overall error introduced is small. Therefore, this simple modeling concept is deemed to be accurate enough.

For the SLOWPOKE-2, the standard approach was deemed to introduce a significant degree of uncertainty in its predictions, as a result of low core uniformity, due to the relatively large number of vacant fuel pin positions in the core lattice. Given the substantial non-linearity of the thermal hydraulic phenomena involved, the accuracy and reliability of the results might degrade if all relevant core parameters where simply averaged throughout the core. In particular, the flow area and flow conditions associated to the hottest fuel pin might differ significantly from the core average, especially for the LEU core, where more than 40% of the available fuel pin positions in the lattice are vacant. Consequently, a model of the hottest fuel pin using the calibration parameters of the whole core might provide inaccurate results due to unrealistic flow conditions. Accordingly, it was decided to create a dual-channel model that consisted of the hottest fuel pin and its corresponding coolant channel and one pin representing the remainder of the fuel pins and its corresponding coolant channel, modeled as two separate assemblies with a common inlet and outlet. Under such conditions, the combined minor pressure losses should remain the same as the ones obtained by calibration of the single average pin model but the flow around the hottest pin would be allowed to differ from that of the core average as appropriate.

An additional problem we faced during modeling as a result of core non-uniformity was the determination of the friction factor. This parameter was determined theoretically, based on available core geometry and flow regime, laminar in this case. What would be typically used is the friction coefficient for an infinite triangular rod array with a given pin pitch to diameter ratio (P/D). Again, a single ratio cannot be defined for HEU or LEU cores due to pin vacancies, and a P/D ratio simply averaged across the core could introduce a significant bias due to complex semi-empirical relationships between geometry and tabulated friction coefficient values. The alternative approach selected was to consider the core as a finite bundle of rods in a triangular array, for which a calculation method involving a weighted average for each flow channel was found in the available literature [9]. The way total flow area of the bundle is divided into sub-channels according to this method is illustrated in Figure 7.

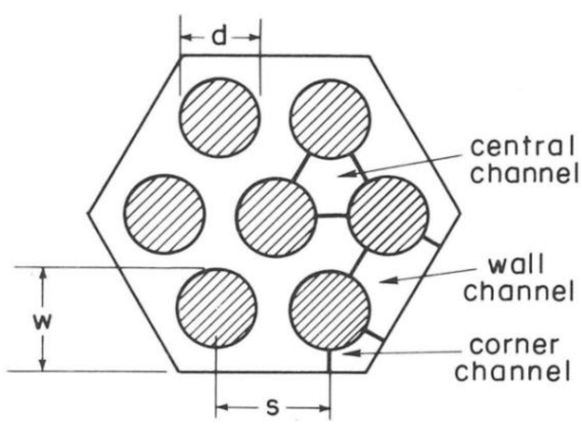

Figure 7: Central, corner, and wall channels of a rod bundle in a hexagonal tube [9]

This procedure required the whole core to be subdivided in flow channels assigned to each pin or tie rod (Figure 9) and involved some minor geometry approximations for individual sub-channels, but was deemed to be significantly more accurate and reliable than existing alternatives. Equation 1 shows how the overall friction coefficient for any given region of the core is obtained by means of an appropriately weighted average.

$$\frac{1}{f\ \mathrm{Re}} = \sum_{i=1}^{m} \frac{1}{(f\ \mathrm{Re})_i} \left(\frac{D_{h,i}}{D_h}\right)^2 \left(\frac{A_{c,i}}{A_c}\right) \qquad \text{(Eq. 1)}$$

Where f is the Darcy friction factor; Re, the Reynolds number; D_h, the hydraulic diameter; A_c, the flow area and m is the total number of flow channels [10].

The single-pin model of the whole HEU core was used as mentioned for calibration, associating reactor power with experimental data of coolant temperature rise through the core obtained during JM-1 commissioning [11]. Given the complex geometry of the coolant path at the core inlet and outlet and the even more complex behavior of coolant flow along such a path, the pressure drop based on calibration using actual experimental data was expected to be much more accurate and reliable than any theoretical estimate that could have been obtained otherwise. The ability of the resulting model to accurately match the collection of experimental coolant temperature increments at different power levels, shown in Figure 8, served as validation and provided confidence in the results beyond those conditions.

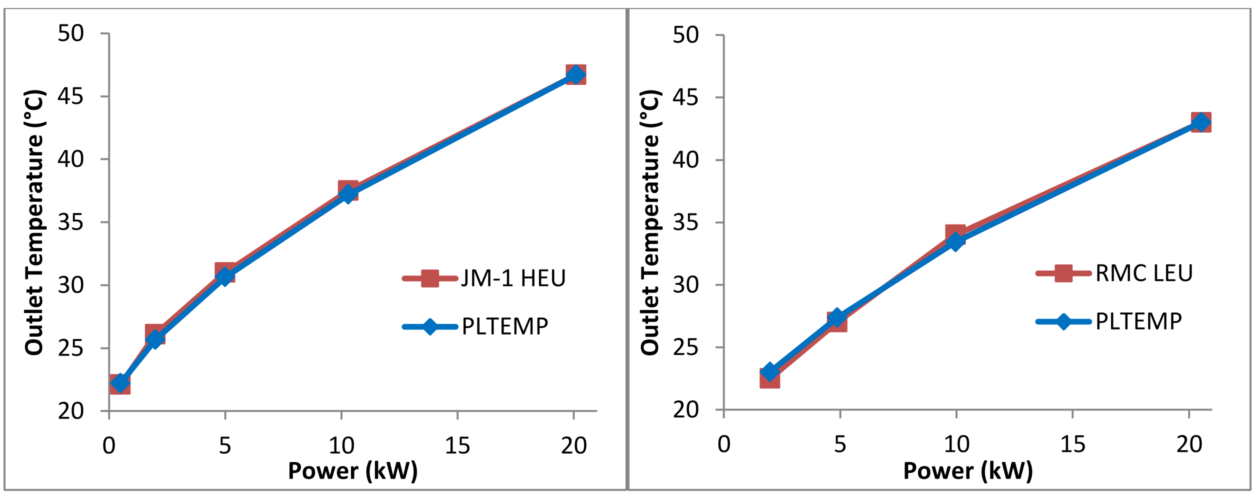

Figure 8: Calculated and experimental coolant outlet temperatures for JM-1 HEU and RMC LEU reactors

The dual channel model, one flow channel for the hottest pin and the other for the rest of the core (with appropriately different dimensions), was used to estimate the maximum allowed operating power of the reactor. Hot channel factors (HCFs), accounting for the uncertainty margins of the relevant simulation parameters, were incorporated into the model in this case to ensure that no foreseeable errors in the process results in larger safety margin estimates than those actually corresponding to the real reactor. In other words, all relevant safety parameters are multiplied by their corresponding hot channel factor, so that the results obtained when all HCFs are considered correspond to the conditions where all those variables simultaneously have their most adverse values within their foreseeable error range.

Given the non-uniform HEU fuel pin arrangement and variable coolant flow corresponding to each fuel pin, the hottest pin should be identified based on the highest power to flow area ratio (P/A_F). The highest power pins in the HEU core are located in the innermost “ring”, or hexagon strictly speaking, next to the central spindle, as illustrated in Figure 9. Some of pins of the second ring would actually be hotter due to a higher P/A_F. However, we deemed that the most conservative assumption was to consider a triangular channel delimited by 3 pins of those rings (Figure 9). We assumed no coolant mixing or heat transfer between adjacent flow channels, which is a necessary conservative assumption if more complex CFD methods are to be avoided. The resulting reference cell has half the area of a regular lattice cell and includes half a fuel pin as well. Correspondingly, the chosen pin power value was half the average of the 3 fuel pins involved, equating to 94.72% of the highest power pin. The constant power distribution assumption, to ensure this choice was also valid at higher power levels, was verified using MCNP for a wide range of reactor power levels, resulting in a maximum axial offset difference of less than 1% and an even lower radial offset, which is the one relevant in the current analysis.

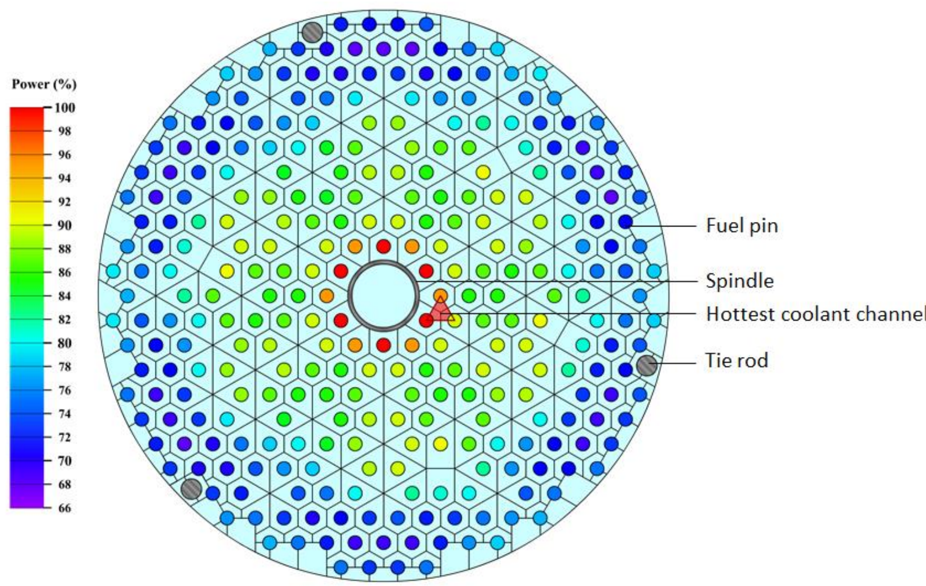

Figure 9: JM-1 HEU core cross section with power distribution and coolant channels segmentation

A similar approach was taken for the analysis of the LEU core. The model calibration results based on experimental data from the RMC reactor are shown in Figure 8.

In the LEU core, both the highest power and hottest pins are located in the innermost “ring”, as shown in Figure 10. The resulting reference cell is almost triangular as well in this case, though larger and including a full fuel pin. The hottest pin power in the LEU core is 16.7% higher than the core average, power distribution being somewhat flatter than in the HEU core.

The calculated maximum allowed reactor operating powers were found to be 46.3 kW and 39.7 kW for the JM-1 HEU and LEU cores respectively without the hot channel factors applied. With all six hot channel factors, the maximum allowed steady state operating powers were found to be 36.45 kW and 31.2 kW respectively for the maximum credible inlet temperature of 40.5°C.

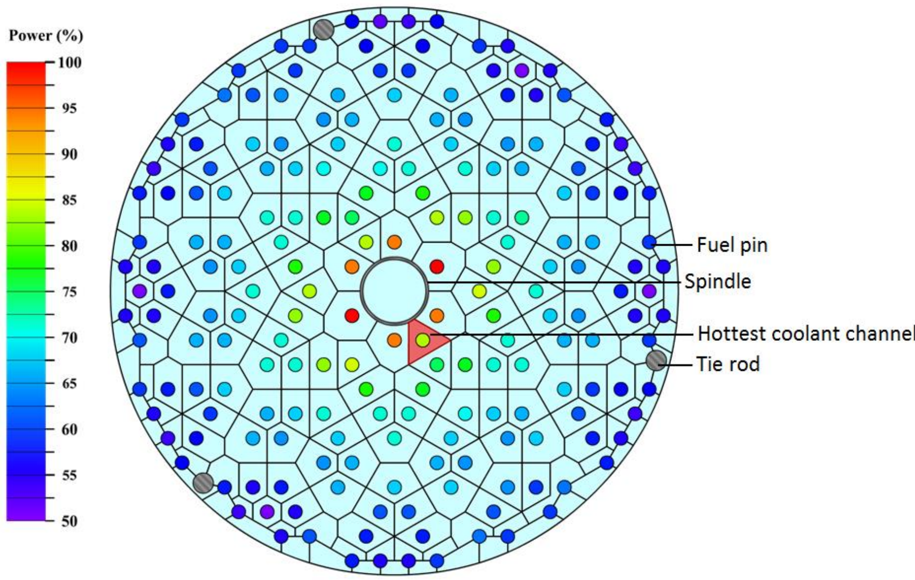

Figure 10: JM-1 LEU core cross section with power distribution and coolant channels segmentation

Under normal operating conditions at maximum power, calculated parameters of the JM-1 reactor are given in Table 1.

Table 1: Calculated operating parameters of the JM-1 reactor (inlet temperature of 40.5°C)

Parameter	HEU	LEU
Nominal Power (kW)	19.08	21.72
Average Fuel Pin Power (W)	64.5	109.67
Peak Fuel Pin Power (W)	80.3	167.73
Coolant Flow Rate (kg s^{-1})	0.239	0.267
Peak Clad Temp.(°C)	81.7	108.6
Average Fuel Temp. (°C)	68.9	89.8
Peak Fuel Temp. (°C)	81.9	112.6

Because the number of pins will be reduced from 296 to 198, the power-per-pin will be increased resulting in higher pin temperatures. However, the margin between peak operating temperatures and melting temperatures will increase significantly. The safety margin between the peak fuel temperature and the fuel melting temperature will increase from ~649°C for the HEU to ~2752°C for the LEU fuel, and the safety margin between peak cladding and cladding melting temperatures will increase from ~568°C for the HEU to ~1741°C for the LEU.

4 Conclusions

The neutronic and thermal hydraulic safety analyses of the Jamaican SLOWPOKE-2 research reactor were performed using MCNP5 and PLTEMP/ANL v4.1 respectively, for the conversion of the reactor from HEU to LEU. The inherent safety of the JM-1 reactor will be maintained with the conversion and despite an increase in reactor power and peak fuel temperatures; there will be increased safety margins in operating the reactor with LEU fuel.

5 References

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