COMPARISON OF MCNP AND WIMS-AECL / RFSP CALCULATIONS WITH HIGH TEMPERATURE SUBSTITUTION EXPERIMENTS IN ZED-2 USING CANFLEX-LVRF

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Abstract

This paper summarizes comparisons of calculation results from MCNP5 and WIMS-AECL / RFSP with experimental results obtained from the Zero Energy Deuterium (ZED-2) critical facility, examining CANFLEX Low Void Reactivity Fuel (CANFLEX-LVRF) in heated channels, substituted into a reference lattice and cooled under ACR-like coolant conditions, with H_2O, air, or CO_2 as an air substitute. CANFLEX-LVRF shares features in common with the ACR-1000 fuel, notably an increase in enrichment (over natural uranium) in the outer elements of the fuel bundle, and presence of a neutron absorber in the central element. The reference and substituted fuel channels were arranged in a 24.5-cm hexagonal lattice in order to provide neutron similarity to the 24-cm square lattice pitch of the ACR-1000. These results therefore provide useful data for validation of the reactor physics toolset for use in ACR-1000 applications. For the mixed lattices, results for both MCNP5 and WIMS-AECL / RFSP show small biases in k_{eff}, ranging from –7 mk to –5 mk, small biases in coolant void reactivity, ranging from –1 mk to +0.5 mk, and good agreement for copper activation rate distributions (based on calculated neutron flux). Bare core MCNP and WIMS-AECL stand-alone results, based on substitution analysis, also show small biases in k_{eff}, ranging from -6 mk to -0.4 mk, and small biases in coolant void reactivity, ranging from –0.3 mk to +3.7 mk. This validation exercise thus gives good agreement between measurement and calculation and provides confidence in the accuracy of the physics toolset.

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1. Introduction

Atomic Energy of Canada Limited (AECL) is developing the ACR-1000 for both the Canadian domestic and international markets [1]. The ACR-1000 is based on the CANDU reactor design with fuel bundles in horizontal pressure tubes, on-line refuelling and heavy water moderation. The ACR-1000 design achieves compact reactor core design and reduced capital costs through the use of light-water as the primary coolant and a reduced fuel channel lattice pitch (24 cm). To fully address the features of the ACR-1000 core design, the accepted CANDU physics analysis codes have been reviewed and enhanced [2]. A comprehensive program of testing and analysis is in place to provide incremental validation of the capabilities and accuracy of these advanced physics codes [3]. This paper summarizes the results of one validation study comparing calculation results from MCNP5 and WIMS-AECL / RFSP with experimental results obtained from the ZED-2 critical facility, using CANFLEX Low Void Reactivity Fuel (CANFLEX-LVRF).

LVRF is an advanced fuel technology that was developed by AECL for potential applications where a reduced or negative coolant void reactivity is desirable [4, 5]. CANFLEX-LVRF is a particular version of LVRF based on the CANFLEX fuel bundle geometry. This fuel is appropriate for the validation tests presented here because it has features in common with the ACR-1000 fuel, notably an increase in enrichment (over natural uranium) in the outer elements of the fuel bundle, and presence of a neutron absorber in the central element. The use of light water coolant and 24.5-cm hexagonal lattice pitch also provide neutron similarity to ACR-1000 conditions. The purpose of the study presented here was to compare calculations using the physics codes MCNP5 and WIMS-AECL / RFSP with results of critical experiments using CANFLEX-LVRF. Mixed-lattice (i.e., substitution) experiments were performed using AECL's Zero Energy Deuterium (ZED-2) tank-type heavy-water moderated critical facility [6]. These experiments provide measurement data used to help quantify the accuracy of reactor physics codes for use in ACR-1000 reactor analysis [2]. A number of reactor physics phenomena were investigated: coolant-density-change induced reactivity, coolant-temperature-change induced reactivity, fuel-temperature-change induced reactivity, and flux distribution in space and time. The experiments included measurements of critical moderator heights and the activation rates of copper foils distributed through the core.

The effects of coolant density, coolant temperature and fuel temperature on core reactivity were inferred from measured changes to moderator critical height and normalized copper foil activation distributions. All experiments in ZED-2 were performed under critical conditions. Models of the ZED-2 core, corresponding to various operational conditions, including critical moderator height, were simulated using MCNP5 [7] and WIMS-AECL / RFSP [8, 9] in order to obtain calculated values for reactivity (k_{eff}) and flux distributions within the core. Comparison of these calculated values with those obtained experimentally are used for validation of both MCNP5 and the WIMS-AECL / RFSP code set [2].

In addition to the analysis of the full-core, mixed-lattice results, it was also of interest to isolate the reactor physics properties of the CANFLEX-LVRF test fuel. This was achieved via MCNP-based substitution analysis of the experimental results, which were used to obtain the bare critical dimensions and bucklings for a pure lattice of CANFLEX-LVRF [8]. These results were then used for stand-alone WIMS-AECL calculations of k_{eff} to assess the accuracy of the deterministic lattice physics code.

2. Experiments

2.1 Test fuel

Experiments were performed in the ZED-2 critical facility, using three fuel types, 28-element natural uranium fuel (28-NU), 19-element uranium metal (19-UM) and CANFLEX-LVRF. In the CANFLEX-LVRF, the outer 42 pins consist of 1.0 wt% $^{235}U/U$, and the centre pin contains a mix of dysprosium and natural uranium. The reference lattice consisted of fifty-five D_2O-cooled 28-NU channels surrounded by thirty channels of D_2O-cooled 19-UM. For substitution experiments, the fuel bundles in the seven central channels were replaced with CANFLEX-LVRF, which was air-cooled, CO_2-cooled or H_2O-cooled. A top down view of the experimental lattice is shown schematically in Figure 1. The 28-NU was used as the primary reference fuel (driver fuel), while the 19-UM was used as a secondary reference fuel (booster fuel).

2.2 Channel characteristics

The reference and substituted fuel channels were arranged in a lattice with a 24.5-cm hexagonal pitch. This lattice pitch was selected because it provides neutron similarity with the 24-cm square lattice pitch of the ACR-1000. Substitution measurements with CANFLEX-LVRF were performed using ZED-2 hot channels in the centre-most channels. These channels are similar in geometry and material composition to CANDU-type channels except that they have been designed so that the contents inside the channels can be heated up to temperatures and pressures representative of the coolant conditions in a power reactor. High temperature measurements required the use of ancillary equipment in each channel, including a heater, heater leads, shroud, support rods, and thermocouples. For room temperature measurements the hardware was removed from the channels, since it was not required. CANFLEX-LVRF channels were cooled with H_2O, or, to represent a voided condition, were cooled with air at room temperature, or with CO_2 at elevated temperatures with heating equipment in place. The 28-NU were placed in CANDU-type aluminium channels while the 19-UM were placed in aluminium coolant tubes. Both channel types were open at the bottom, allowing the D_2O moderator in the calandria to serve both as moderator and as coolant for these channels.

2.3 Operational conditions

For each experiment, the moderator critical height, moderator temperature and moderator purity were measured. For the heated channel experiments, the hot site temperature and hot site pressure were also measured. Over the course of the experiments, the moderator purity was approximately 99.7 wt%D_2O, the moderator was at room temperature and hot channels were heated from room temperature to temperatures ranging from 50 °C to 300 °C. In the ZED-2 facility, the D_2O pool extends beyond and sits below the fuel channel lattice region, acting as a radial and bottom axial reflector. Surrounding and below the aluminium calandria tank is graphite, which acts as an additional radial and bottom axial reflector. Each CANFLEX-LVRF fuel channel contained five bundles of fuel, and the bottom of the fuel was approximately 15 cm above the ZED-2 calandria floor. The CANFLEX-LVRF fuel bundles were approximately 50 cm in length, and the moderator critical height varied between 218 and 238 cm. Consequently, between approximately 25 and 50 cm of fuel was exposed above the moderator. Although not moderated, this fuel can make a non-trivial contribution to the total core reactivity.

2.4 Copper Activation Profile / Flux Map Experiments

The neutron flux profile within the core can be inferred from the activation profile of Cu foils distributed within the core, since Cu activation arises primarily from capture of thermal neutrons. Cu foil activation experiments were performed on three critical lattice configurations at room temperature: a reference lattice, a substitution lattice with seven air-cooled CANFLEX-LVRF central channels, and a similar substitution lattice with seven H_2O-cooled CANFLEX-LVRF channels. For radial flux measurements, Cu foils were distributed radially, through the centre of the core, in the East-West direction, spaced at intervals of 24.5 cm (one lattice pitch). Each foil was located midway between two fuel channels, as shown schematically in Figure 1. Radial flux profiles were measured at three elevations, 95.0, 105.0 and 115.0 cm, relative to the calandria tank floor. Axial flux measurements were performed at two radial locations, 12.25 cm (one half lattice pitch) East and West of the centre channel. Foils in the axial direction were spaced at intervals of 10 cm distributed at elevations ranging from 15.0 to 205.0 cm. For all flux map experiments, both the coolant and moderator were at room temperature. Flux map experiments were only performed under room temperature conditions.

2.5 Heated channel experiments

For the heated channel experiments, CANFLEX-LVRF channels were cooled using either CO_2 or H_2O. Both the CO_2-cooled and H_2O-cooled experiments were performed at fuel/coolant temperatures ranging from room temperature up to 300 ^oC, while the moderator (and D_2O coolant in the 28-NU and 19-UM channels) remained at room temperature. For the H_2O-cooled experiments, a helium cover gas under pressure (0.46 MPa at room temperature) was used to suppress coolant boiling in the channels. For the heated channel experiments, Cu foils were not used, but the moderator critical heights were recorded at each fuel/coolant temperature.

2.6 Measurement Uncertainties

The absolute uncertainty in critical height was ±0.2 mm. The core and coolant temperatures both had absolute uncertainties of ±0.2 ^oC. The uncertainty in D_2O purity was ±0.005 wt% D_2O. These uncertainties were propagated into uncertainties in the values of k_{eff} calculated using MCNP and WIMS-AECL / RFSP for each experiment.

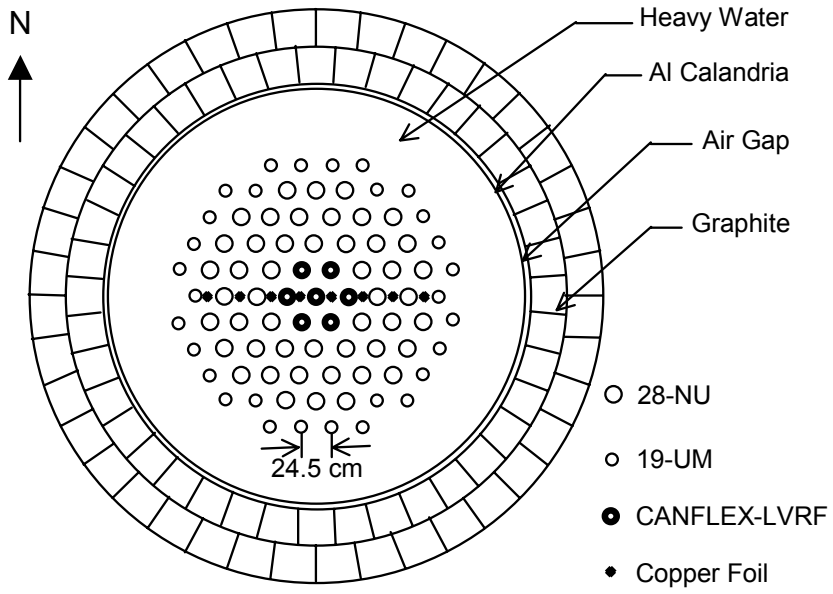

Figure 1 Top view of the ZED-2 lattice illustrating the core layout, positions of the test, reference and driver fuels and radial distribution of copper foils.

3. Computer Codes and Computational Models

3.1 MCNP5

MCNP, a general-purpose Monte Carlo N-Particle code, is a three-dimensional, stochastic, probabilistic, neutron transport code [7]. The code treats an arbitrary three-dimensional configuration of materials in geometric cells defined by surfaces. For the calculations presented here, MCNP5 was used in conjunction with a nuclear data library based on ENDF/B-VI, Release 8.

The MCNP5 models were precise representations of the actual experimental set-up, including the fuel bundles, aluminium fuel channels, bottom and radial D_2O reflectors, aluminium calandria tank, bottom and radial graphite reflector, outer neutron and gamma-ray shielding, steel hanger rods, various air gaps, etc. For the experiments that included Cu activation measurements, the aluminium-backed Cu foils were modelled using cylinders, which represented the rectangular prism strings of the actual experiments; cylinders were chosen in order to represent the arbitrary orientations of the stringers with location and time.

The tallied runs (experiments which included copper foil activation measurements) were performed with 900 million neutron histories in order to obtain low statistical uncertainties (~ ±1%) in the simulated foil activation rates. The remaining runs were performed with 30 million histories, which was sufficient to ensure that the statistical uncertainty in the calculation of k_{eff} was less than ±0.1 mk.

3.2 WIMS-AECL 3.1

WIMS-AECL 3.1 is a two-dimensional multi-group deterministic lattice physics code that solves the integral neutron transport equation using collision probabilities [8]. In addition, the code can take into account leakage due to finite reactor dimensions, based on pre-defined input bucklings, and can model strong heterogeneous effects via its multi-cell capability. For this study, WIMS-AECL was used in conjunction with an 89-group nuclear data library based on ENDF/B-VI.5 and ENDF/B-VI.7. For the results presented here, WIMS-AECL was used in two capacities: first, to produce homogenized two-group cross sections for use in RFSP core calculations, and second, in stand-alone mode, to evaluate k_{eff} based on bucklings obtained from MCNP-based substitution analysis.

Fuel channel models in WIMS-AECL consisted of two-dimensional slices through the channels, perpendicular to the channel axes. A number of approximations were performed in order to obtain two-dimensional representations of fuel bundles. These approximations include smearing of the fuel density, to account for volume lost due to the dish depth of the fuel pellets, and smearing of the cladding density in order to account for the reactivity contribution from material at the ends of the fuel bundles.

Two-group homogenized cross sections for use in RFSP were determined based on calculations using both single-cell and multi-cell WIMS-AECL models. Single-cell models were used for fuel channels that were surrounded by like fuel and coolant (e.g., the central CANFLEX-LVRF channel), while channels that neighboured either unlike channels or reflector were modelled using multi-cell models that accounted for the heterogeneous environment around the channel of interest.

For WIMS-AECL in stand-alone mode, leakage calculations were performed to evaluate k_{eff}. The bucklings used in the leakage models were derived from MCNP-based substitution analysis of the high-temperature mixed-lattice substitution experiments in ZED-2 [9]. Since these bucklings correspond to a critical bare lattice, the k_{eff} calculation can be used to provide a direct evaluation of the accuracy of WIMS-AECL.

3.3 RFSP 3-04

RFSP is a neutron-diffusion approximation-based Reactor Fuelling Simulation Program, which solves the three-dimensional, two-group neutron diffusion equations using the finite difference method in a Cartesian coordinate system [10]. The geometry is defined by sets of Cartesian meshes, where each mesh is assigned a set of two-group cross sections and corresponds to one point in the two-group flux solution of the diffusion equation. In this study, RFSP was used to evaluate reactivity levels and their changes, and two-group flux and power distributions. The homogenized two-group cross sections used in RFSP were evaluated (as described above) from WIMS-AECL calculations of single-cell and multi-cell lattice physics models.

Because RFSP is a deterministic, two-group diffusion code with a Cartesian coordinate system, a number of approximations and simplifying assumptions were made in the modelling of the hexagonal lattice ZED-2 experiments. For this study, each fuel channel was represented in the x-y

plane by a 2×2 arrangement of sub-cells, with sub-cell lattice spacings in the x and y directions defined asymmetrically, using x-spacings equal to 1/2 the lattice pitch and y-spacings equal to √3/4 the lattice pitch (see Figure 2). This arrangement preserved the centre-to-centre distance between fuel channels in the lattice as well as the channel positions within the core.

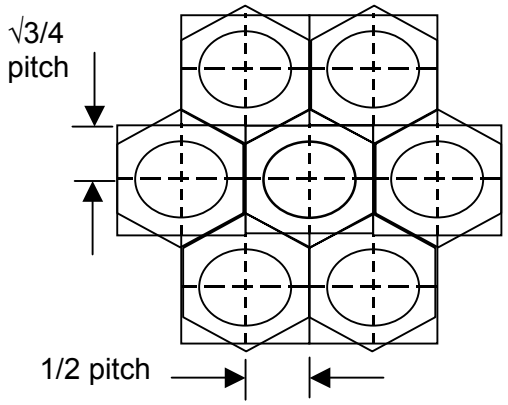

Figure 2 Schematic representation of the rectangular cell approximation of hexagonal lattice. Solid lines and circles define the fuel channels and surrounding hexagonal or rectangular cells. The dashed lines show the sub-cell boundaries.

Finer Cartesian mesh spacings were used in the radial direction at the boundary between the D_2O reflector and calandria, in order to better represent the circular boundaries of the D_2O, calandria and radial graphite reflector. The RFSP model of ZED-2 extended to the outermost physical boundary of the radial and bottom graphite reflectors. In the radial direction, a vacuum boundary condition was imposed. At the upper axial boundary, the RSFP model extended to the top of the moderator level. Above the moderator, either a vacuum boundary condition was imposed (VBC), or an extrapolation distance based on an experimentally determined flux distribution (EBC) was used. Both sets of results are presented here.

4. Results

4.1 Reactivity calculations

Results of MCNP and WIMS-AECL / RFSP reactivity calculations for heated channel experiments are summarized in Table 1. The value of k_{eff} from MCNP calculations does not show a significant change for either H_2O-cooled or air / CO_2-cooled channels over the range between 25 ^oC and 300 ^oC. The MCNP bias in k_{eff} was approximately –6.5 mk for H_2O-cooled channels and approximately –6.2 mk for air / CO_2-cooled channels. The MCNP mixed-lattice CVR bias ranged between approximately +0.2 mk and +0.45 mk. However, this variation was close to the uncertainty in the CVR bias, ±0.2 mk.

For RFSP calculations, two sets of results are presented, one using the VBC and the other using the EBC, as described in Section 3.3. Both the VBC and EBC results show a decrease in k_{eff} with

increasing channel temperature. However, there was some scatter in the k_{eff} values, suggesting competing reactivity bias contributions from the RFSP models. For the VBC-based calculations, the bias in k_{eff} ranged between approximately –5 mk to –7 mk for H_2O-cooled channels and was approximately -7 mk for air / CO_2-cooled channels. The corresponding biases in k_{eff} were consistently lower in magnitude for the EBC calculations, and ranged between approximately –4.5 mk to –5.5 mk for H_2O-cooled channels and was approximately –5 mk for air / CO_2-cooled channels. The CVR bias for the VBC calculations ranged from approximately –1 mk to –0.3 mk and between –0.4 mk and +0.4 mk for the EBC calculations. The tendency for RFSP was to show a slight increase in the mixed-lattice CVR bias with temperature, increasing by approximately 0.8 mk when going from room temperature to 300 °C.

Table 1 Calculated reactivities for mixed-lattice high-temperature substitution experiments.

Model	Temperature (^oC)	k_{eff}		CVR bias (mk)	$\sigma_{CVR\ bias}$ (mk) (approximate)
		Air / CO_2	H_2O		
MCNP	25	0.99384	0.99346	+0.39	±0.2
RFSP – VBC	25	0.99290	0.99388	-0.99	±0.2
RFSP – EBC	25	0.99507	0.99542	-0.35	±0.2
MCNP	200	0.99390	0.99346	+0.45	±0.2
RFSP – VBC	200	0.99301	0.99371	-0.70	±0.2
RFSP – EBC	200	0.99514	0.99516	-0.02	±0.2
MCNP	300	0.99382	0.99361	+0.21	±0.2
RFSP – VBC	300	0.99268	0.99293	-0.26	±0.2
RFSP – EBC	300	0.99480	0.99440	+0.41	±0.2

4.2 Neutron flux distributions

Representative plots of normalized experimental, MCNP-calculated and RFSP-calculated copper foil activation rates are shown in Figures 3 to 8. Absolute differences between normalized data and MCNP and RFSP calculations are shown in accompanying plots. Figures 3 and 4 show plots of radial and axial profiles, respectively, of Cu foil activation for the reference lattice. The different plots for the radial data in Figure 3 show differences of no more than 3% of the peak activation value for both MCNP and RFSP. For the axial activation profiles, differences are also approximately 3% of the peak activation value for elevations less than approximately 175 cm. Above 175 cm, the RFSP results deviate somewhat from the experimental results by approximately 8% of the maximum activation.

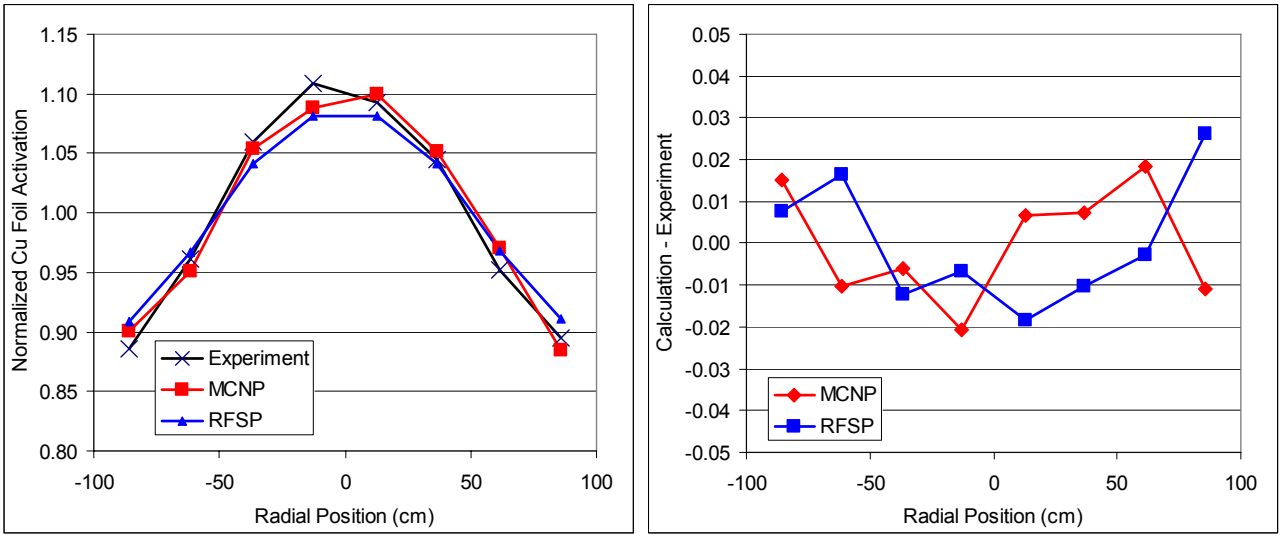

Figure 3 On the left are plots of radial Cu activation profiles determined experimentally and calculated via MCNP and WIMS / RFSP for the D_2O-cooled reference core. Plotted on the right are corresponding differences between calculated and experimental results.

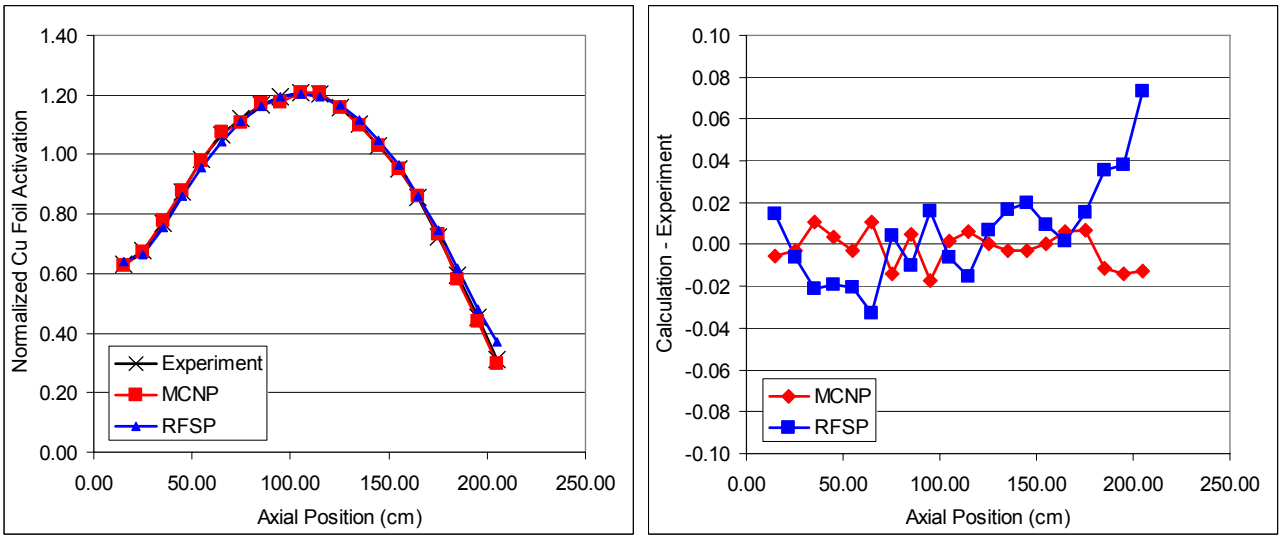

Figure 4 On the left are plots of axial Cu activation profiles determined experimentally and calculated via MCNP and WIMS / RFSP for the D_2O-cooled reference core. Plotted on the right are corresponding differences between calculated and experimental results.

Radial and axial profiles of Cu foil activation for the air-cooled CANFLEX-LVRF substitution core are shown in Figures 5 and 6, respectively. The different plots for the radial data in Figure 5 show differences of no more than 2% of the peak activation value for both MCNP and RFSP. For the axial activation profiles, differences are also approximately 2% of the peak activation value for elevations greater than 25 cm and less than approximately 175 cm above the calandria floor. Below 25 cm and above 175 cm, the RFSP results deviate from the experimental results by less than 10% of the maximum activation, respectively. This is not surprising since the accuracy of diffusion solutions deteriorates close to the upper vacuum boundary, where neutron transport is highly anisotropic, and also since 3-dimensional transport effects are prevalent near the interface between

the core and the bottom D_2O reflector. This is particularly true for the air-cooled case where there is non-trivial axial neutron streaming.

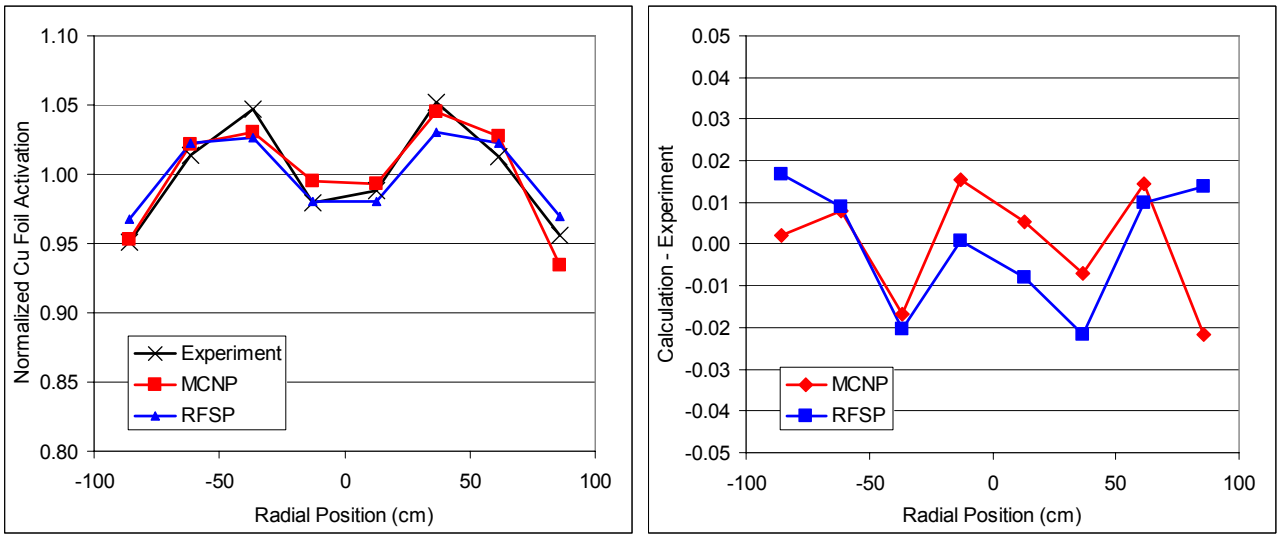

Figure 5 On the left are plots of radial Cu activation profiles determined experimentally and calculated via MCNP and WIMS / RFSP for the air-cooled CANFLEX-LVRF substituted core. Plotted on the right are corresponding differences between calculated and experimental results.

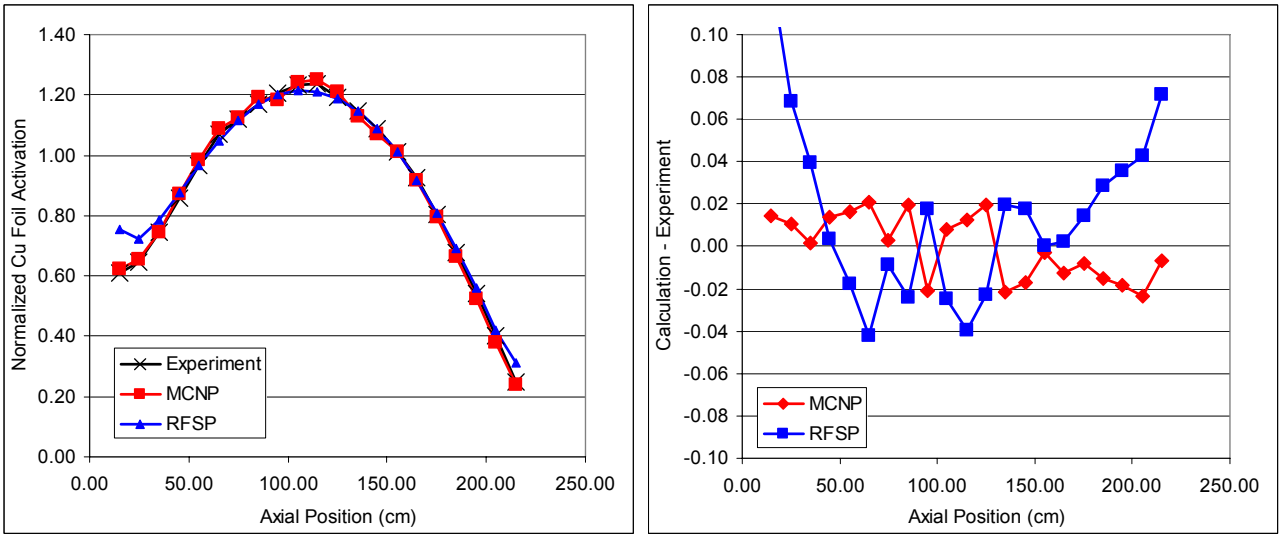

Figure 6 On the left are plots of axial Cu activation profiles determined experimentally and calculated via MCNP and WIMS / RFSP for the air-cooled CANFLEX-LVRF substituted core. Plotted on the right are corresponding differences between calculated and experimental results.

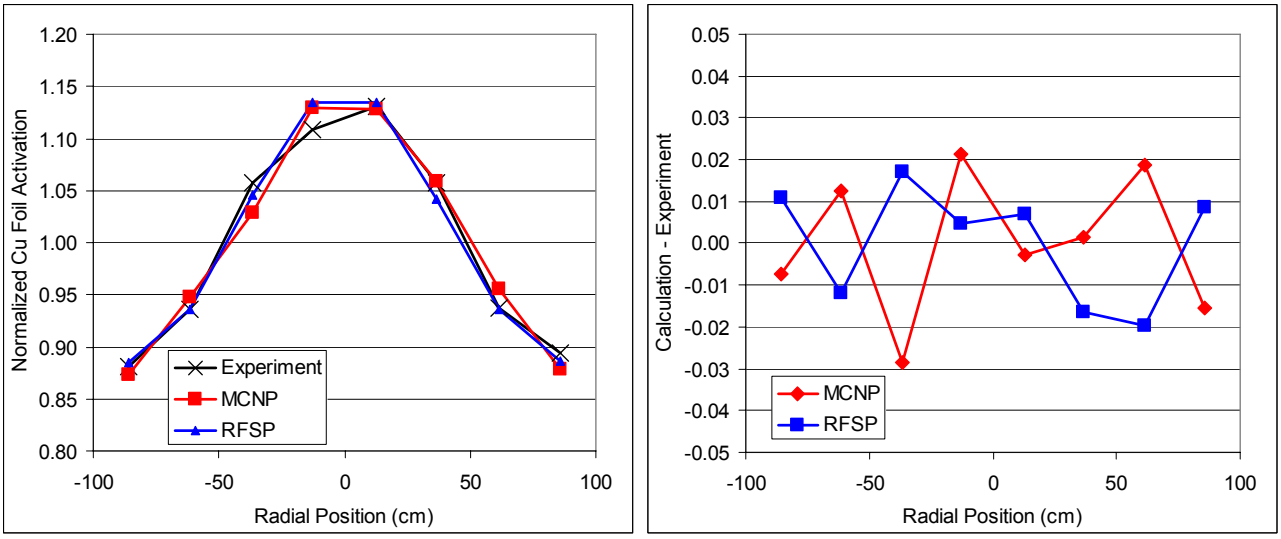

Figure 7 On the left are plots of radial Cu activation profiles determined experimentally and calculated via MCNP and WIMS / RFSP for an H_2O-cooled CANFLEX-LVRF substituted core. Plotted on the right are corresponding differences between calculated and experimental results.

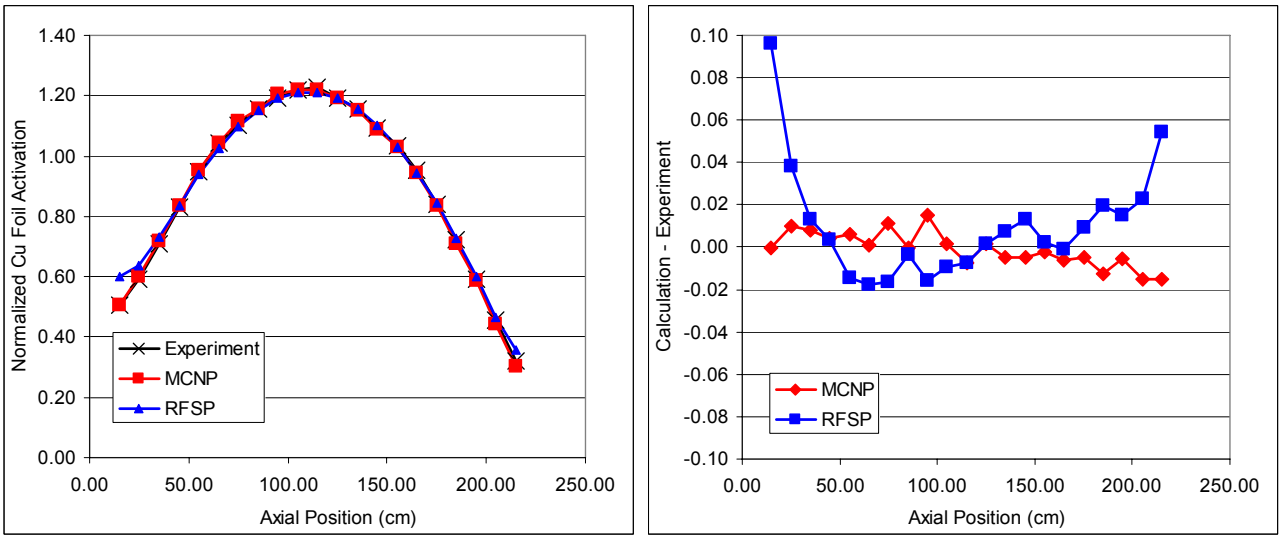

Figure 8 On the left are plots of axial Cu activation profiles determined experimentally and calculated via MCNP and WIMS / RFSP for an H_2O-cooled CANFLEX-LVRF substituted core. Plotted on the right are corresponding differences between calculated and experimental results.

Figures 7 and 8 show the radial and axial profiles, respectively, of Cu foil activation for the H_2O-cooled CANFLEX-LVRF substituted core. The difference plots for the radial data in Figure 7 are within 3% of the peak activation value for both MCNP and RFSP. For the axial activation profiles, differences are again approximately 2% of the peak activation value for elevations greater than 25 cm and less than approximately 175 cm. Below 25 cm and above 175 cm, the RFSP results deviate from the experimental results by approximately 10% and 6% of the maximum activation, respectively.

Discrepancies between the RFSP and experimental axial Cu activation results at low and high elevations arise from two independent sources. In both cases, the neutron flux calculated using RFSP is overestimated with respect to experiment. Near the bottom of the core the overestimate in neutron flux is due to an underestimate of neutron absorption in the reflector region, while at the top of the core, it is due to an underestimate of the extent of neutron leakage when the experimentally based upper extrapolation distance (EBC) is imposed on the RFSP model.

4.3 Bare core CANFLEX-LVRF simulations

To isolate the properties of the CANFLEX-LVRF from the reference core, substitution analyses were performed, using the method as described in [9], to determine the dimensions and bucklings of bare core critical lattices of CO_2-cooled and H_2O-cooled CANFLEX-LVRF. The critical dimensions and bucklings were then used in MCNP and stand-alone WIMS-AECL, respectively, to calculate k_{eff}. The results of these calculations are summarized in Table 2.

MCNP calculations of k_{eff} showed a small decrease (approx 1 mk) between 25 ^oC and 300 ^oC. The MCNP bias in k_{eff} ranged from approximately –2 mk to –3 mk for H_2O-cooled channels and from approximately –4 mk to –6 mk for the CO_2-cooled channels. The MCNP CVR bias ranged from approximately +2.0 mk to +3.7 mk with an uncertainty of approximately ± 2 mk.

For stand-alone WIMS-AECL calculations, the variation in k_{eff} with temperature was on the order of 1 mk. The bias in k_{eff} ranged from approximately -1.4 mk to -2.4 mk for H_2O-cooled channels and from -1.7 mk to -0.4 mk for CO_2-cooled channels. The stand-alone WIMS-AECL CVR bias ranged from approximately -0.3 mk to +1.7 mk with an uncertainty of approximately ±2 mk. Since the bucklings used in the WIMS-AECL calculations are determined from substitution analysis, the uncertainties in CVR bias for both WIMS-AECL and MCNP are approximately the same.

Table 2: Calculated Reactivities from MCNP Bare Core and Stand-alone WIMS-AECL Models

Model	Temperature (^oC)	k_{eff}		ΔCVR (mk)	±δCVR (mk) (approximate)
		CO_2	H_2O		
MCNP	25	0.99530	0.99820	+2.9	2
WIMS Stand-alone	25	0.99832	0.99865	-0.3	2
MCNP	200	0.99397	0.99768	+3.7	2
WIMS Stand-alone	200	0.99957	0.99885	+0.7	2
MCNP	300	0.99502	0.99701	+2.0	2
WIMS Stand-alone	300	0.99931	0.99758	+1.7	2

5. Conclusions

In this study, calculation results from MCNP5 and WIMS-AECL / RFSP were compared with experimental results obtained for CANFLEX-LVRF in heated channels, substituted into a reference lattice of 28-NU surrounded by 19-UM fuel drivers at a 24.5-cm hexagonal lattice pitch. MCNP-based substitution analyses were also performed, in order to isolate properties associated with the CANFLEX-LVRF. The substitution analyses results were then used in additional MCNP5 and

WIMS-AECL stand-alone calculations. For the mixed lattices, the results for both MCNP5 and WIMS-AECL / RFSP showed small biases in k_{eff}, ranging from –7 mk to –5 mk, small biases in coolant void reactivity, ranging from –1 mk to +0.5 mk, and good agreement for copper activation rate distributions (based on calculated neutron flux). Bare core MCNP and WIMS-AECL stand-alone results, based on substitution analysis, also show small biases in k_{eff}, ranging from –6 mk to -0.4 mk, and small biases in coolant void reactivity, ranging from –0.3 mk to +3.7 mk. Overall, the validation results show very good agreement between measurement and calculation and consequently help to build on confidence in the accuracy of the physics toolset for modelling of ACR-1000.

6. References

[1] J. Hopwood, "The AECL ACR-1000 Reactor", Proceedings of PHYSOR-2006, Vancouver, British Columbia Canada, 2006 September 10-14.

[2] B.P. Bromley, F.P. Adams, M.B. Zeller, D.G. Watts, B.V. Shukhman, and J. Pencer, "Validation of MCNP and WIMS-AECL / DRAGON / RFSP for ACR-1000 applications", PHYSOR 2008, International Conference on the Physics of Reactors, Nuclear Power: A Sustainable Resource, Interlaken, Switzerland, 2008 September 14 –19.

[3] A. Buijs, "The Design Process and Simulation Tools Used in the Neutronic Design of the ACR-1000 Core", Proceedings of the 23rd CNS Nuclear Simulation Symposium, Ottawa, Ontario Canada, 2008 Nov 2-4.

[4] P.G. Poczar and J.D. Sullivan, "Low Void Reactivity Fuel", Proceedings of the 25th Annual Conference of the Canadian Nuclear Society, Toronto, Ontario Canada, 2004 June 6-9.

[5] J.H. Lau, F.J. Doria, H. Chow, LK.H. Leung, S. Palleck, K.S. Sim, Z. He, and P. Purdy, "Design and Qualification of the Bruce CANFLEX® Low Void Reactivity Fuel Bundle – An Overview", Proceedings of the 25th Annual Conference of the Canadian Nuclear Society, Toronto, Ontario Canada, 2004 June 6-9.

[6] M.B. Zeller "Physics Experiments in the ZED-2 Reactor in Support of the Advanced CANDU Reactor", Proceedings of the 25th Annual Conference of the Canadian Nuclear Society, Toronto, Ontario Canada, 2004 June 6-9.

[7] X-5 Monte Carlo Team, "MCNP – A General Monte Carlo N-Particle Transport Code, Version 5", LA-UR-03-1987, Los Alamos National Laboratory, Los Alamos, New Mexico, 2003 April.

[8] D.V. Altiparmakov, "New Capabilities of the Lattice Code WIMS-AECL", PHYSOR 2008, International Conference on the Physics of Reactors, Nuclear Power: A Sustainable Resource, Interlaken, Switzerland, 2008 September 14 –19.

[9] B.P. Bromley; F.P. Adams; and G.B. Wilkin "Development and Testing of a MCNP-Based Method for the Analysis of Substitution Experiments", Transactions of the American Nuclear Society, Vol. 97, 2007, pp. 711-712.

[10] W. Shen, D.A. Jenkins, A. Connolly, I. Attieh, D. Buss, T. Liang, and S. Hart, "Benchmarking of WIMS-AECL / RFSP Multi-cell Methodology with MCNP for ACR-1000 Full-Core

Calculations", PHYSOR 2008, International Conference on the Physics of Reactors, Nuclear Power: A Sustainable Resource, Interlaken, Switzerland, 2008 September 14 –19.

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