Safety Analysis of a Homogeneous SLOWPOKE Reactor (Part 2)

A Paper Submitted for

the 31st Annual Student Conference

of the Canadian Nuclear Association and the Canadian Nuclear Society

St. John, New Brunswick, Canada

3-6 June 2007

by

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ABSTRACT

An opportunity is presented with the approaching end of some of the SLOWPOKE-2 research reactors' design life, which consists of replacing the current heterogeneous core with a homogeneous assembly that would allow the extraction of radioisotopes (such as ^{99}Mo) for nuclear medicine and other applications. Preliminary investigation has demonstrated the feasibility of this concept and produced an initial design of a Homogeneous SLOWPOKE reactor that demonstrates the potential for inherent safety based on calculations of a strong negative reactivity coefficient due to temperature.

The present research aims at continuing the safety analysis of the proposed homogeneous assembly, most notably in assessing its inherent safety characteristics with respect to factors such as void fraction of the moderator and additional thermalhydraulic effects.

Modelling of the reactor is accomplished using both the deterministic WIMS-AECL and the probabilistic MCNP 5 for the determination of the reactor's reactivity and flux shape, while COMSOL Multiphysics is used for thermalhydraulics modelling. In pursuing the safety analysis, the design of the reactor is improved. One example of the improvements is the proposed substitution of the required additional reflector material in the initial design with graphite instead of beryllium. Another improvement is the replacement of the unique control rod in the core centre with a cluster of control rods within the radial reflector. This would not only simplify the construction of the reactor vessel, but increase the core volume for added radioisotope production capacity. The latest results of reactor simulations and the safety analysis will be presented.

INTRODUCTION

This work is a continuation of the feasibility study performed by Lt(N) Paul Busatta [1]. Busatta's work focused on investigating the possibility of replacing an existing SLOWPOKE-2 fuel assembly with a container filled with a homogeneous aqueous solution of uranyl sulphate, mostly in terms of neutronics. The design reactor is intended mainly for the production of medical isotopes with the possible side benefits of maintaining the ability to continue neutron activation analysis and other research activities currently performed by SLOWPOKE-2 facilities. The possible financial and educational benefits to institutions operating SLOWPOKE-2 reactors make this a very attractive proposition.

The paper submitted at the 30th Annual Student Conference [2] described the general theory of the homogeneous reactor, and also the envisioned approach to the safety analysis. The research carried out since then and additional data availability have lead to a change in the approach employed, which shall be explained in detail. This paper will discuss steps taken to date in the safety analysis of the homogeneous SLOWPOKE design through computer modelling using mainly Monte Carlo Neutron Particle-5 (MCNP-5), with some discussion of current and future work with Winfrith Improved Multi-group Scheme – Atomic Energy of Canada Limited 3-1 (WIMS-AECL 3-1) and COMSOL Multiphysics modelling programmes.

Due to the lack of an actual Homogeneous SLOWPOKE nuclear reactor from which to obtain experimental data, the validation of the experimental models still represents a serious challenge. While the computer modelling of an existing homogeneous reactor design (i.e.- ARGUS) could demonstrate the effectiveness of the modelling approach through comparison of the simulated data with actual experimental observations, the required level of reactor data has so far not been available. Some possible alternate validation approaches will be discussed.

METHODOLOGY

Homogeneous Reactors and ^{99}Mo Production. The general theory of homogeneous reactors and their use in the production of ^{99}Mo is discussed at Reference [2], and is only summarized here. Aqueous homogeneous reactors have been described as "daunting design and material challenges" [3], due to the following characteristics intrinsic to their nature [4]: Corrosiveness of the fuel/moderator solutions, gas formation through water dissociation, and the requirement for external circulation of fuel. Advances in material science and technology since the development period of nuclear reactors has mitigated some of these factors and can allow for the exploitation of other compelling strengths of homogeneous reactors [5]: High specific power, simplified core design, high neutron economy, large negative temperature coefficient of reactivity, and continuous removal of fission products. last characteristic makes possible the extraction of ^{99}Mo, a precursor isotope for ^{99m}Tc, which is a widely used radioisotope in nuclear medicine. Successful extraction of material of high purity from the Russian ARGUS reactor has provided valuable precedent [6], while the conventional methods of producing ^{99}Mo that are both inefficient and wasteful [7].

MCNP-5 Criticality Calculations. Monte Carlo N-Particle 5 is a general-purpose, continuous-energy, generalized-geometry, time-dependent, coupled neutron/photon/electron Monte Carlo transport code which has the ability to calculate k_{eff} eigenvalues for fissile systems [8]. MCNP-5 has been used as the main code to optimize the features of the homogeneous SLOWPOKE reactor. During criticality determinations, the code tracks individual lives of virtual neutrons in the geometry and materials defined by the user. Each virtual neutron has a pre-set weighting: representing a set number of individual neutrons in an actual reactor. The weighting is decreased through probabilistic means as the virtual neutrons react in the model geometry, just as new virtual neutrons are created in a probabilistic fashion for the next generation. An averaged effective multiplication factor for a given reactor model is achieved in this manner after a user set number of generations are completed to achieve steady state.

Safety Analysis. The safety analysis approach that was originally envisioned is described in the first part of this series[2]. In the absence of detailed data and observations from other existing homogeneous reactors, the work has proceeded with the evolution of the reactor itself based on a consultation with Drs John Hilborn and Hugues Bonin [9]. Extensive modeling has occurred using MCNP-5 to determine the effective multiplication factor of various reactor models with modifications to materials and configurations, as summarized in the actual steps taken to date in the safety analysis of the homogeneous reactor:

1. Busatta's final MCNP-5 model was run to ensure consistency of the reactor models, and for comparison against the author's future models.

2. A benchmark model was created. An extensive library of cross sections are available in MCNP-5, and nearly all naturally occurring isotopes were modelled. Many other simplifications to material composition in the original model were also eliminated, such as with the use of actual nuclide weight fractions for Zircaloy-2, including allowance for naturally occurring isotopes, in place of pure ^{90}Zr for the container material. A minor configuration change to enhance the generic applicability of the concept to actual SLOWPOKE-2 reactor facilities was also implemented, involving the recreation of a fifth outer-irradiation site. Irradiation site #5 was never installed on RMC's SLOWPOKE-2 reactor in order to accommodate a Neutron Beam Tube system as part of a Neutron Radiography Facility. Finally, density formulas for both the fuel solution and the zircaloys at the operating temperature of 313 K were amended according to formulas from the literature [10,11].

3. Various material modifications (detailed later) were executed individually, based primarily on the previously mentioned coordination meeting [9]. Desired modifications were grouped and several configurations of reactors were advanced based around the container alloys used in the particular model: Zircaloy-2, 347 Stainless Steel and Ledeburitic Steel [12].

4. Various additions and modifications to the reflector configuration and material were implemented, as well as minor reactor and pool water temperature and density changes. Impurities were added to reflector materials, due to significant impact on the effective scattering cross section noted in the literature [13]. Models were developed incorporating graphite in all reflector additions in an attempt to reduce the cost (with due concern given to the galvanic interaction between the two reflector materials). The original beryllium reflector from the SLOWPOKE-2 reactor remains in all models.

5. Fuel solution molality was adjusted with the final configurations until excess reactivity was similar to that of the original SLOWPOKE-2 and Busatta's homogeneous reactor, within one standard deviation (~3.84 mk) [1].

6. The central container orifice and control rod were removed and replaced by various rod clusters in the irradiation sites. Inner and outer irradiation site clusters were modeled, as well as various numbers and diameters of rods, until shutdown could be assured.

7. Infinite multiplication factors for each of the final reactor configurations were determined using two different complementary methods.

8. Finally, 2-dimensional deterministic WIMS-AECL models were constructed based on cross sections of the final MCNP-5 models they represent.

Determining k_∞. Two methods were used in determining the infinite multiplication factor for the final homogeneous reactor models:

1. First Principles. The four-factor formula ($k_\infty=\eta\varepsilon pf$) was employed using output from the MCNP-5 models to determine each factor from its basic definition, which is summarized for each in Table 2. This method is possible because of the "bottom-up" nature of a Monte-Carlo simulation details virtual neutron counts, as well as most interactions at steady state of the model.

Table 1. Basic Definitions of the Four Factors (references as indicated, n = neutron)

Symbol	Factor	Definition	Reference	Remarks
η	Thermal Regeneration Factor	$\nu \bullet \dfrac{thermal\ n\ absorbed\ in\ fission}{total\ thermal\ n\ absorbed\ in\ fuel}$	[14], Section 2.56	ν = 2.42 (for ^{235}U)
f	Thermal Utilization Factor	$\dfrac{thermal\ n\ absorbed\ in\ fuel}{total\ thermal\ n\ absorbed}$	[14], Section 3.87	
p	Resonance Escape Probability	$1-\dfrac{n\ absorbed\ from\ E\ to\ E_o\ (resonance\ region\ ^{238}U)}{source\ strength\ of\ n\ in\ core}$	[14], Section 3.62	
ε	Fast Fission Factor	$\dfrac{n\ produced\ from\ fission\ by\ all\ incident\ n}{n\ produced\ from\ fission\ by\ thermal\ incident\ n\ only}$	[15], p. 287	

2. 3-Dimensional Model Scaling. k_∞ was also determined by scaling the reactor models in all 3 dimensions by various increasing factors to construct a distinctively logarithmic-shaped plot of the resultant k_{eff}. The horizontal asymptote of these graphs provides the infinite multiplication factors for each of the reactor models (see Figure 3).

Using WIMS-AECL 3.1. WIMS-AECL is a two-dimensional multigroup neutron transport code routinely used for reactor lattice cell calculations [16]. In order to represent the final 3-dimensional homogeneous reactor models 2-dimesionally in WIMS-AECL, the separate dimensional buckling components are required code inputs. In order to make this possible, unidimensional scaling of the final models in each the radial and axial dimensions was performed in MCNP-5 to determine, respectively, the axial and radial components of the buckling. Overall geometric buckling for a bare finite cylindrical reactor can be seen at Table 2, and serves as a simplified analogue for the effect exploited here. Radial and axial components of the buckling can be determined when spatial separability of the buckling is assumed. The separate components can be determined by increasing the model scale unidimensionally to effectively achieve ideal infinite reactor shapes as also shown for comparison in Table 2. Theoretically the addition of these unidimensional buckling components should equal the overall buckling calculated using k_∞ from the models increased in scale 3-dimensionally, but more importantly, the correct ratio of axial to radial buckling can be determined for the WIMS-AECL input file.

Table 2. Buckling for Critical Bare Reactors – Analogue for Determination of Model Buckling Components (adapted from Reference [15], Table 6.2, with remarks inserted by the author)

Geometry	Dimensions	Buckling	Remarks
Finite Cylinder	Radius – R Height – H	$\left(\frac{2.405}{R}\right)^2 + \left(\frac{\pi}{H}\right)^2$	3-dimensional expansion of this model approaches k_∞
Infinite Cylinder	Radius – R	$\left(\frac{2.405}{R}\right)^2$	Unidimensional axial scaling As H→∞
Infinite Slab	Thickness – a	$\left(\frac{\pi}{a}\right)^2$	Unidimensional radial scaling As R→∞, with a = H

Error Analysis

Precision and Accuracy. MCNP5 computed standard deviations refer to the precision of the model runs, and are not a measure of the accuracy of the calculated k_{eff} compared to that of an actual reactor. There are many reasons why only a precision can be determined [17], with the main two reasons applicable in this case being: The geometry and material approximations employed by the user in constructing the model, and user error. While the models have been improved since the first homogeneous reactor was conceived [1], the accuracy of the runs cannot be determined in the absence of an actual homogeneous SLOWPOKE.

Handling of Accuracy in this Paper. The MCNP-5 calculated standard deviation is used as a measurement error in the absence of any other figure. Separate runs of the same model are treated like independent measurements, and their values are averaged. The standard error propagation formula is then used to determine a new standard deviation for this average, assuming the form of the measured k_{eff} for the model form a Gaussian distribution [18]:

$$\frac{1}{\sigma^2_{<X>}} = \sum_{i=1}^{N} \frac{1}{\sigma^2_{x_i}}$$

RESULTS AND DISCUSSION

Initial Excess Reactivity Determinations

Figure 1 presents a comparison of the excess reactivities of the initial models. While maintaining constant fuel-solution molality, the sum of the changes does result in a significant increase in excess reactivity, as evidenced in Figure 1. The effects of the initial actions performed on Busatta's model are confounded in the Benchmark model, and it is not possible to separate them from the current set of data. The information would be of academic interest only, and the cost of running extra experiments where the changes are introduced individually was deemed unjustified. The comparison serves only to show the relative effects of the materials involved under the modeled conditions, and in no way demonstrates the overall superiority of any material in itself.

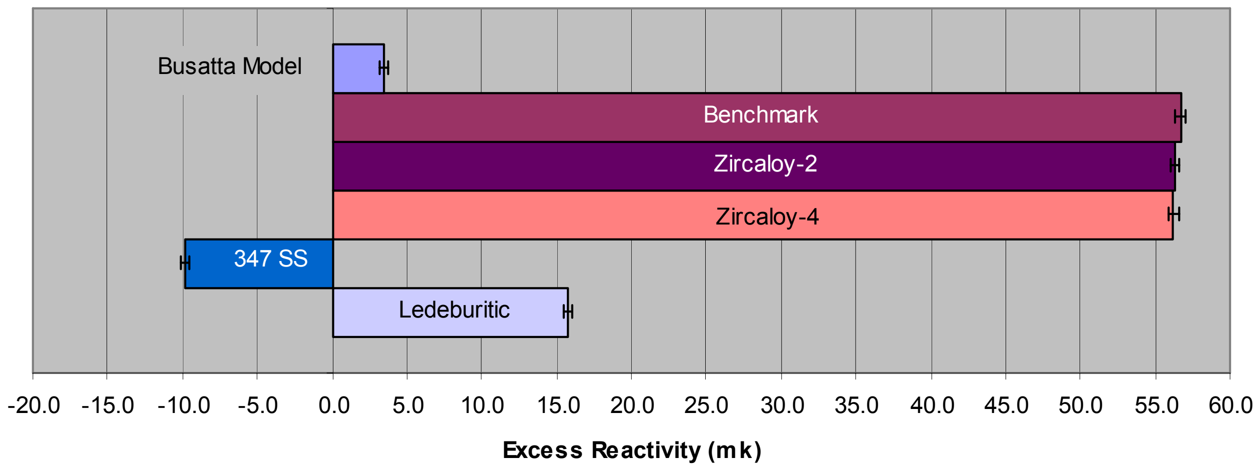

Figure 1. Comparative Excess Reactivities of Base MCNP-5 Models.

Choice of Fuel Container Materials

General. The results of a literature review conducted at Reference [19] determined that a dilute aqueous solution of Uranyl Sulphate (UO_2SO_4), H_2SO_4 and $CuSO_4$ with an oxygen overpressure was the fuel of choice for a small homogeneous reactor in terms of materials chemical compatibility with both Type 347 Stainless Steel and Zircaloy-2. The Homogeneous SLOWPOKE does not contain H_2SO_4, $CuSO_4$ or an oxygen overpressure – as these are intended to amplify protection of the materials at higher temperature applications (i.e.- approximately 300ºC). The discussion beyond this section will only include points relevant to the two final container material options, Zircaloy-2 and 347 Stainless Steel.

Zircaloy-2 and Zircaloy-4. A comparison of Zircaloy-2 and Zircaloy-4 elemental compositions and properties leads to the conclusion that Zircaloy-2 (as opposed to Busatta's proposed tank material [1]) is the more suitable material for the homogeneous SLOWPOKE tank. At the operating temperatures envisioned, the effects of both corrosion and hydrogen embrittlement by the mechanisms experienced in a typical CANDU reactor are not expected [20]. Modelling of the reactor with MCNP-5 supports the fact that there is little difference between these materials in terms of neutronic behaviour (compare k_{eff} for Zircaloy-2 and -4 models at Figure 1).

Ledeburitic Steel. Another high-silicon steel alloy referred to as Ledeburitic Steel was also modeled, and showed great promise due to its nearly inert quality in a high temperature and high molar sulphuric acid solution. This steel was also of interest because of its high silicon and carbon content (together 18% weight fraction [12]) and consequent low neutron absorption cross section, as evidenced by the modeled excess reactivity in Figure 1. Unfortunately, upon further investigation, this novel alloy was dropped from further consideration due to its expected highly brittle physical nature [21], poor suitability for welding and a simple lack of past experience with applications in the nuclear industry.

347 Stainless Steel. There is a long history of the use of 347 Stainless Steel in the nuclear industry [22,23] in contrast to Ledeburitic Steel. Its relative cost in comparison to the exotic zirconium alloys makes an attractive choice, but the challenge in its practical use lies with steel's larger neutron absorption cross section, requiring more fissile material to achieve criticality. As seen in Figure 1, with no change in fuel solution molality, the model of the Homogeneous SLOWPOKE nuclear reactor with a 347 Stainless Steel container does not even achieve criticality.

Choice of Reflector Additions

Configuration. As Busatta has shown [1], an addition to the annulus reflector is required to match the height of the new core container. A top reflector with a similar thickness to that of the base reflector, as well as a donut-shaped reflector to fill the void around the present base reflector, were both added in the present models to further improve the performance of the core. These additions aided in making the 347 Stainless Steel model critical at low fuel solution molality. Considering the very large excess reactivities predicted in some of the cases studies and shown in Figure 1 above, the top reflector may well be designed with an adjustable thickness to limit the excess reactivity for safety reasons, just as practiced in the present SLOWPOKE-2 reactors.

Material Impurities. The previously mentioned reflector coatings were not modeled in this work, although reasons for their requirement and possible material choices for the coatings are discussed later. Reactivity effects due to reflector material impurities were studied and are summarized at Figure 2, below, which demonstrates the difference in excess reactivity when various compositions of beryllium and graphite are incorporated into the benchmark model. The reflector material in the Benchmark model is pure beryllium, and the addition of impurities in beryllium can be seen to have a significant impact (of –10 mk and more). The excess reactivity of the benchmark dropped to less than half with the introduction of graphite reflector additions, a property that later proved very useful in reducing the undesirably high excess reactivity of the Zircaloy-2 models. Reference [24] gives impurity composition for actual SLOWPOKE-2 beryllium: Supplier-provided measurements (by unspecified means) are from the supplier of beryllium for all SLOWPOKE-2 reactors (Brush Wellman Inc.), and the NAA based measurements conducted by École Polytechnique are from the beryllium used in its own SLOWPOKE-2. Graphite Impurities are based on pre-irradiation values of nuclear grade graphite from a Japanese study [25], and had an insignificant positive effect on excess reactivity. The results on graphite are not exhaustive, and significant variation in quality and composition in graphite are noted in the literature [26].

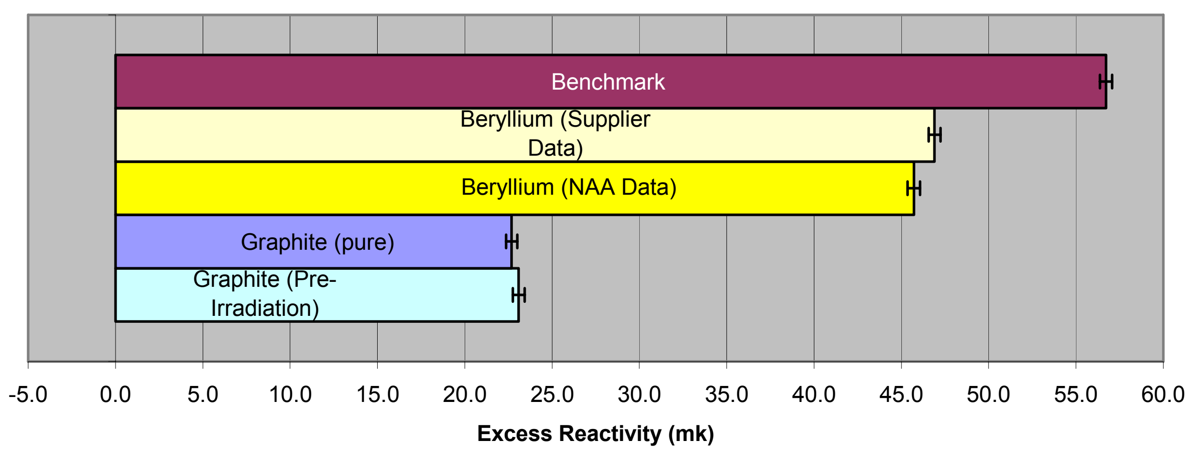

Figure 2. Comparative Excess Reactivities of Differing Reflector Materials/Impurities.

Material Choices. The galvanic potential between beryllium and graphite is extreme and steps such as coating the graphite with a thin layer of metal, epoxy or Poly-Ether-Ether-Ketone (PEEK) [27] would be required in any composite reflector. These coatings were not modeled in the present work. In the final MCNP-5 homogeneous models, beryllium reflectors were used in all additions for the 347 Stainless Steel container models, and graphite additions were incorporated into the Zircaloy-2 models.

Fuel Solution

The maximum 20% enrichment allowed under non-proliferation agreements was maintained in all models. The molality of the solution was varied in order to achieve an excess reactivity similar to both the SLOWPOKE-2 and Busatta's model reactors (approximately 3.8 mk). Final values were 1.46 molal for the Zircaloy-2 model, and 1.70 molal for the 347 Stainless Steel model.

Water

All models are designed with light water. For safety and costs reasons, heavy water has not been modeled mainly because of concerns related to tritium generation from the absorption of neutrons in deuterium, and consequent possible contamination hazards. Water within the reactor shell on all final models was kept at the expected reactor operating temperature (40ºC), which is also the likely worst-case scenario in terms of steady state operation. The pool temperature outside of the reactor shell was set at 20ºC, and could easily be modified in later runs.

Air

Air above the fuel solution in the container is modelled as water vapour-saturated air at 40ºC. Radiolysis of water could contribute a significant fraction of elemental hydrogen and oxygen gas to this air in an actual reactor, and the fission process could add gaseous products as well (such as ^{41}Ar), both dependent on the power and the temperature of the reactor's operation. The reason for a gas removal tube as part of the reactor's design is in fact to remove these gases. This will have to be studied in terms of composition and effects on reactivity at a later time, as the quantities are not presently known and the radiolysis and fission product gases were not included in the reactor's model. Air inside the irradiation sites is modeled as dry air at 25ºC, and both composition and temperature could easily be modified in later runs.

Control Rod Configuration

Placement. The central orifice and control rod were removed and replaced with rods outside of the core in an effort to reduce the fissile material required for the design, and lessen the complexity of the reactor container. Various configurations were attempted in the inner and outer irradiation sites. The criteria for the final control rod properties was simply set at achieving a reverse in the sign of the excess reactivity with rods fully inserted compared to when fully retracted. Outer irradiation sites proved completely ineffective at shutting down the reactor, so inner sites within the reflector annulus were used. The rods when used in cluster were placed as symmetrically as possible about the core to achieve maximum effect, as depicted in Figure 3. The placement of the control rods within the reflector has been influenced by the importance of the thermal neutron flux within the reflector, and would have to be mapped out at various power levels and continually monitored to ensure known fluxes at the irradiation sites for uses such as NAA.

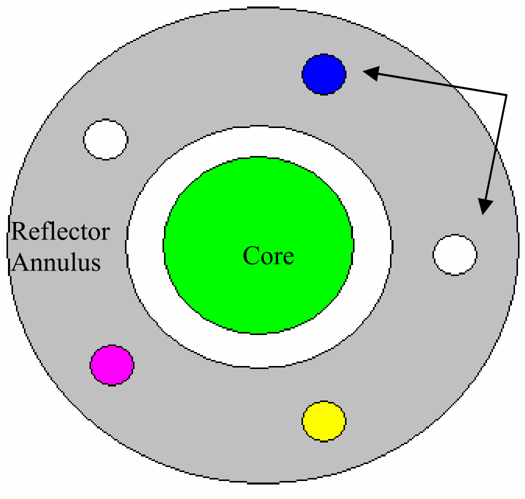

Figure 3. Symmetrical placement of the control rods leads to a more even power distribution in the core.
*** Zircaloy-2 models have a cluster with rods in the blue and yellow sites.**
*** 347 Stainless Steel models have a cluster with a single rod located in each of the three coloured sites (blue, yellow and pink).**

Size and Number. Clusters of up to five control rods were modeled, with some notable results presented at Table 3, below. In some runs, the radius of these rods was increased to limit the number of valuable inner irradiation sites used, in order to save these for Neutron Activation Analysis. In these final models 2 x 3.8 mm rods are used for Zircaloy-2 models and 3 x 3.2 mm rods are used for 347 Stainless Steel models. The rods seemed to have less effect on the Stainless Steel model, even in the inner irradiation sites, requiring a greater mass of inserted Cd to achieve shut down. This is likely because the less neutron-transparent container reduced the amplitude of the thermal neutron flux in the reflector, hence its importance, and effectively shielded the core from the effects of the control rods. In order to achieve perfect symmetry, a cluster of five control rods is preferred here, provided that sample irradiation for NAA studies could be carried out using the outer irradiation sites at the expense of longer irradiation times.

Table 3. Comparison of Excess Reactivities for Various Control Rod Configurations (Fully Retracted and Fully Inserted)

Model Type	Rod Cluster Configuration	MCNP-5 Excess Reactivity (mk)	
		Fully Retracted	Fully Inserted
Zircaloy-2	5 x 2 mm rods Outer Sites	+4.34	+2.75
	1 x 2 mm rod Inner Site #1	+3.88	+1.79
	5 x 2 mm rods Inner Sites	+3.39	-6.62
	2 x 3.8 mm rods Inner Sites #1,3	+4.04	-3.45
347 Stainless Steel	5 x 2 mm rods Outer Sites	+4.19	+3.62
	1 x 2 mm rod Inner Site #1	+4.08	+2.29
	5 x 2 mm rods Inner Sites	+4.49	-3.51
	3 x 3.2 mm rods Inner Sites #1,3-4	+4.02	-3.34

Determination of the Infinite Multiplication Factor

General. Values for k_∞ from two methods are at Table 4. The initial values calculated using the four factor formula are suspect for reasons explained below.

Table 4. Infinite Multiplication Factor Values for Final Reactor Models

Model Type	Four Factor Initial Calculation	3-Dimensional Modelling
Zircaloy-2, 1.46 molal	1.00	1.59917 ± 0.00016
347 Stainless Steel, 1.70 molal	1.01	1.62486 ± 0.00018

Issues with the four factor formula. Special runs are being conducted to provide neutron tallies for various neutron energy bins and at locations throughout the core and reflector. In the meantime, more limited data from standard MCNP-5 output files have been used to conduct preliminary calculations for k_∞. The form of the modeling output is such that the error could only be arbitrarily assigned to these calculations, as the values exist in the absolute virtual world of the MCNP-5 model (hence no error in Table 4). Two other outstanding issues remain with the calculated results:

1. Undermoderation. Undermoderation occurs in small reactors (and most light water reactors), and leads to significant epithermal neutron-induced fissions [28]. This is contrary to one of the assumptions of the four-factor formula, which is that only thermal neutrons induce fission in ^{235}U. This assumption is accounted for in the thermal regeneration factor, whereas the fast fission factor accounts for fissions (in ^{238}U) by high-energy neutrons. The possibility of a correction factor for the undermoderation was not investigated in this work.

2. Inaccurate ^{238}U Resonance Region Neutron Count. The initial calculations are based on ratios of fissions produced by three energy groups of neutrons (as reported in standard MCNP-5 output files), the energy windows for these groups do not perfectly match those generally used for the formula. Also, the measure of neutron-induced fissions provided by the standard MCNP-5 output file gives a biased account of the actual presence of the inducing neutrons due to differing average cross sections for each group (generally much smaller at high energies). The second issue is being rectified by executing special runs that will tally neutrons in various areas of the core to produce a more accurate picture of the population ratio between the three energy groups. However, these runs take longer to accomplish by several orders of magnitude (approximately a week each), and are not yet completed at this time. When they are, the resonance escape probability in particular is expected to increase, with a consequent increase in k_∞.

Asymptotes from Three-Dimensional Modelling. Figure 4 denotes how increasing the size of the models leads to a horizontal asymptote in graphed values of k_{eff}, giving k_∞. The size of the reactor was increased (by using scaling factors) until the graph had essentially a zero slope for four points. This last scale model (in each case 600 x all dimensions) was rerun 4 more times for a total average of five assumed-independent runs, with the error (deviation at 95% confidence interval) reported at Table 4.

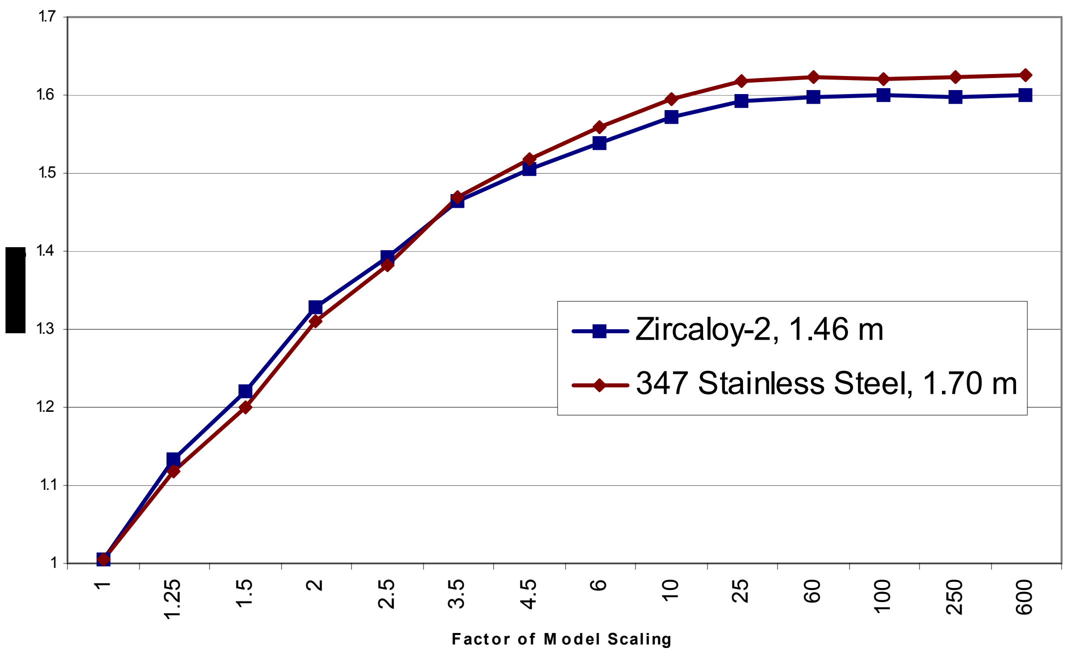

Figure 4. Determination of Infinite Multiplication Factor through 3-Dimensional Model Scaling.

Explanation of Behaviour in Figure 4. An explanation of the behaviour seen in Figure 4 is instructive, and shows how confidence can be ascribed to the resultant k_∞ values. Both models start out at nearly identical k_{eff} values, engineered to be that way through modification of the fuel solution molality. At low values of scaling, the increase in Stainless Steel in the core (with identical increase in all other materials) causes the k_{eff} value to increase more slowly for that model than that of the more neutron-transparent Zircaloy-2 container model. At approximately 3 factors of scale, the trend lines cross, and the material properties of the fuel solution become the final determinants of k_∞. The higher molality of the 347 Stainless Steel model's fuel solution determines that that model will have a higher k_∞, because after approximately 25 factors of scale, this material is pushed out so far from the core it becomes insignificant.

CONTINUATION OF SAFETY ANALYSIS

The Way Ahead. The following is an outline of the next steps that will be taken in the safety analysis of the homogeneous SLOWPOKE reactor models:

Temperature Coefficients of Reactivity. In the case of the homogeneous reactor, α_{fuel} or α_{prompt} is equivalent to $\alpha_{moderator}$, because there is no barrier between the fuel and moderator in solution. This means that the moderator can be assumed to heat up instantaneously with an increase in power. Two methods will be used to determine α:

1. Directly, using the definition [29]: $\alpha_{prompt} = \frac{1}{k^2}\frac{dk}{dT}$, and MCNP-5 models modified for different core temperature conditions;

2. Indirectly, using updated WIMS-AECL figures for the four factors, and the already calculated buckling components (to determine the leakage probability) [30]:

$$\alpha_{moderator} = \alpha_T(f) + \alpha_T(p) + \alpha_T(P)$$

Coefficient of reactivity due to void formation. This coefficient of reactivity is to be determined with the modelling of the reactor at high temperatures with the fuel-mixture density adjusted accordingly and the insertion of empty spheres representing boiling in the MCNP-5 model. This would lead to the confirmation of the inherent safety characteristics of the proposed Homogeneous SLOWPOKE.

Flux Shape. Determination of the Flux Shape Using WIMS-AECL, and preferably confirmation of k_{inf} through the inputs of buckling components.

Raise Operating Power. Higher operating power/temperature would result in an increased rate of ^{99}Mo production. Some possible operating powers to investigate: 50 kW and 100 kW.

Heat Transfer Mechanisms. Modification and use of Busatta's COMSOL Multiphysics model to confirm that convective heat transfer still works for all the above changes – even if minor cooling of the input water is required.

RECOMMENDATIONS

Model an Existing Homogeneous Reactor. Future modelling of an existing homogeneous reactor to better validate the safety analysis approach, when data can be gathered in sufficient detail to create a similarly detailed MCNP-5 model.

Volumetric Control of Reactivity. Possible elimination of the need for a control rod through variation of the volume of Uranyl Sulphate Solution within the reactor tank (requiring extra piping and a subcritical holding tank). One negative aspect of this modification, added complexity aside, would be the extra volume of Uranyl Sulphate solution required, however, this requirement already exists with the required circuit to allow removal of the fission products.

Protective Coating. Model a graphite reflector coating such as PEEK – model a thin-layer coating for the graphite reflector additions to protect the pre-existing beryllium reflector components (anode) from galvanic interaction.

Neutron Beam Tube. Neutron Beam Tube modelling if this capacity is to be maintained for a given installation so equipped, or added in concert with the homogeneous core installation.

Fuel Composition. Possible changes to the fuel composition include using thoriated fuel for purposes of breeding Uranium-233, thus improving the reactor performance and possibly allowing a lower solution molality.

CONCLUSION

The present work continues as a first step in the design of the Homogeneous SLOWPOKE reactor, intended mainly for the production of radioisotopes for nuclear medicine. Busatta's initial model has evolved, and now two main configurations with many possible minor modifications have been proposed. Time-dependent modelling of thermalhydraulics will further confirm that the natural convection cooling mechanism for the homogeneous SLOWPOKE core in these models remains adequate. The promise of Mo-99 isotope production while maintaining research activities such as neutron activation analysis, all within existing SLOWPOKE-2 facilities, makes the homogeneous SLOWPOKE research reactor a concept worth pursuing.

ACKNOWLEDGEMENTS

The author and his supervisor acknowledge the appreciated contribution of Dr. John W. Hilborn (AECL retired) who suggested and supports this work, and the financial assistance of the Academic Research Programme of the Department of National Defence of Canada.

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