HOMOGENEOUS SLOWPOKE REACTORS FOR REPLACING SLOWPOKE-2 RESEARCH REACTORS AND THE PRODUCTION OF RADIOISOTOPES

H.W. Bonin, J.W. Hilborn*, G.E. Carlin, R. Gagnon & P. Busatta*****
Royal Military College of Canada, Kingston, Ontario Canada K7K 7B4
bonin-h@rmc.ca
*Atomic Energy of Canada Ltd (retired)
** Ontario Power Generation
*** Officers in the Canadian Forces

Abstract

Inspired from the inherently safe SLOWPOKE-2 research reactor, the Homogeneous SLOWPOKE reactor was conceived with a double goal: replacing the heterogeneous SLOWPOKE-2 reactors when they reach end-of-core life to continue their missions of neutron activation analysis and neutron radiography at universities, and to produce radioisotopes such as ^{99}Mo for medical applications. A homogeneous reactor core allows a much simpler extraction of radioisotopes (such as ^{99}Mo) for applications in industry and nuclear medicine.

The 20 kW Homogeneous SLOWPOKE reactor was modelled using both the deterministic WIMS-AECL and the probabilistic MCNP 5 reactor simulation codes. The homogeneous fuel mixture was a dilute aqueous solution of Uranyl Sulfate (UO_2SO_4) with 994.2 g of ^{235}U (enrichment at 20%) providing an excess reactivity at operating temperature (40 ^{o}C) of 3.8 mk for a molality determined as 1.46 mol kg^{-1} for a Zircaloy-2 reactor vessel. Because this reactor is intended to replace the core of SLOWPOKE-2 reactors, the Homogeneous SLOWPOKE reactor core had a height about twice its diameter. The reactor could be controlled by mechanical absorber rods in the beryllium reflector, chemical control in the core, or a combination of both. The safety of the Homogeneous SLOWPOKE reactor was analysed for both normal operation and transient conditions. Thermal-hydraulics calculations used COMSOL Multiphysics and the results showed that natural convection was sufficient to ensure adequate reactor cooling in all situations. The most severe transient simulated resulted from a 5.87 mk step positive reactivity insertion to the reactor in operation at critical and at steady state at 20 ^{o}C. Peak temperature and power were determined as 83 ^{o}C and 546 kW, respectively, reached 5.1 s after the reactivity insertion. However, the power fell rapidly to values below 20 kW some 35 s after the peak and remained below that value thereafter. Both the temperature and void coefficients are significantly more negative than the corresponding coefficients in SLOWPOKE-2. The simulations of the reactor in transient states showed that the temperature and power levels attained never compromised the integrity of the reactor.

1. Introduction

In this work, a homogeneous nuclear reactor is proposed as an alternative for the production of radioisotopes such as ^{99}Mo which is chosen here as a representative of several radioisotopes that can be extracted from the irradiated homogeneous fuel mixture of the Homogeneous SLOWPOKE reactor for medical as well as non-medical applications. The extraction of radioisotopes from a homogeneous fuel-moderator mixture has many advantages [1] such as simplicity and minimal amounts of radioactive waste incurred. There are no requirements to produce and transport highly enriched uranium targets that represent safety and security issues. In heterogeneous reactors, the ^{99}Mo atoms produced in the driver fuel elements remain in the solid fuel, and only the ^{99}Mo produced in the targets is recovered. The reader may obtain a good description of a suitable ^{99}Mo extraction process from the reactor homogeneous fuel solution by consulting the work of Cheng *et al.* [2]. Homogeneous reactors are those reactors for which there are no physical barriers between the fuel and the moderator. Several advantages are provided by homogeneous reactors, such as the ease of construction of the reactor avoiding the need of fuel assembly design, fuel cladding and fabrication, the high fuel burn-up, and, more important, the possibility of processing the fuel mixture on-line to remove the fission products (and extract the wanted radioisotopes). Homogeneous reactors have a natural tendency to achieve inherent safety since these reactors most often display strong negative temperature coefficients of reactivity and very strong negative void reactivity coefficients.

2. The Homogeneous SLOWPOKE Reactor as Based on the SLOWPOKE-2 Nuclear Reactor

2.1. The SLOWPOKE-2 Reactor

"SLOWPOKE" is an acronym for Safe LOW-POwer (K) Critical Experiment. It is a 20 kW unpressurized, pool-type reactor, intended primarily for university research. In its current version, the SLOWPOKE-2 nuclear research reactor, depicted in Figure 1, is a small, low power, pool-type reactor based on a core assembly of nearly 200 fuel rods made of UO_2 fuel pellets with a 20% initial enrichment. The reactor core is essentially a large-diameter fuel bundle, its size being slightly larger than a 925-g can of coffee.

The SLOWPOKE-2 nuclear reactor is inherently safe. Inherent safety is defined by the International Atomic Energy Agency as follows: "*Inherent Safety refers to the achievement of safety through the elimination or exclusion of inherent hazards through the fundamental conceptual design choices made for the nuclear plant. Potential inherent hazards in a nuclear power plant include radioactive fission products and their associated decay heat, excess reactivity and its associated potential for power excursions, and energy releases due to high temperatures, high pressures and energetic chemical reactions. Elimination of all these hazards is required to make a nuclear power plant inherently safe. For practical power reactor sizes this appears to be impossible. Therefore the unqualified use of "inherently safe" should be avoided for an entire nuclear power plant or its reactor.*" [3].

The following safety principles must be adhered to. Any positive power transients (from causes such as the sudden withdrawal of the control rod) result in the reactor always remaining in a physical state such that the equipment and the fuel are not damaged. In particular, fission products and other radioactive contaminants are never released into the primary coolant and the environment. As an additional measure of safety, the SLOWPOKE-2 reactor design restricts a maximum excess reactivity to +4.0 mk. This ensures that a sudden and complete withdrawal of the control rod from the core cannot cause a prompt criticality excursion. Other safety principles include the following: no uranium fuel needs to be stored on site, strong negative temperature and void reactivity coefficients, natural circulation of

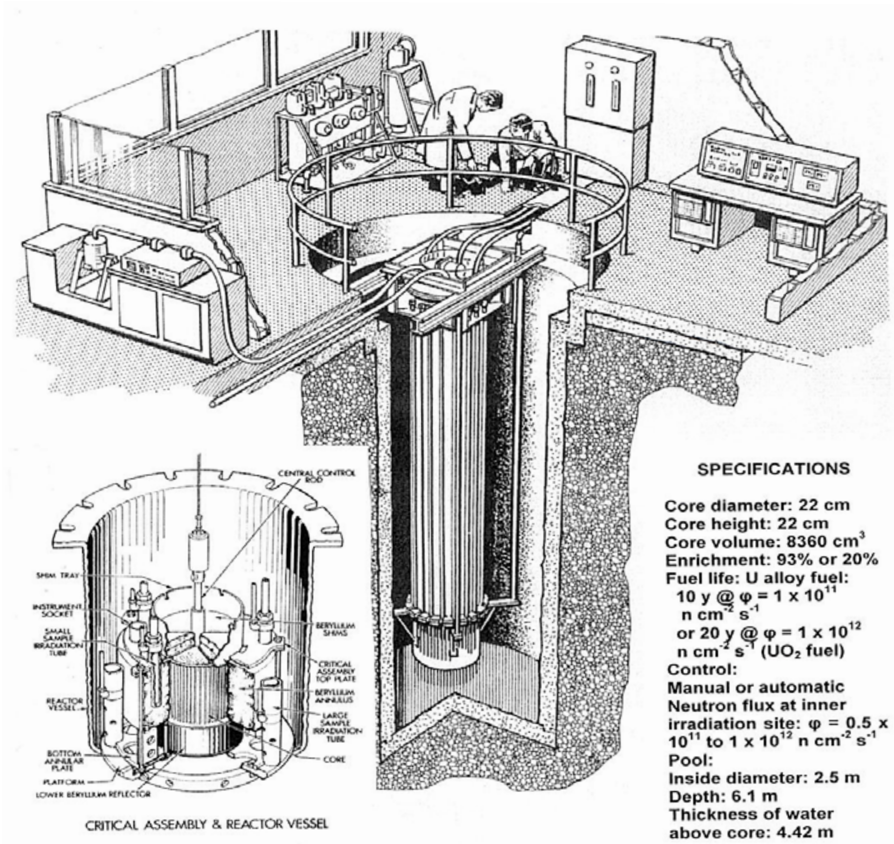

Figure 1: SLOWPOKE-2 reactor pool and facility [4].

water coolant through the core at atmospheric pressure, double containment of the reactor core, core sub-critical in water (i.e. to become supercritical, it needs both the beryllium reflector and the immersion in water), top beryllium reflector with adjustable thickness to keep the excess reactivity at or below +4.0 mk, and single motor-driven control rod operating between -1.5 mk and +4.0 mk. The fuel assembly is a single bundle, consisting of 200 Zircaloy-clad fuel rods containing 1100 grams of U-235 in the form of LEU ceramic pellets. An annular beryllium reflector surrounds the core, and beryllium plates are added periodically to the top reflector to compensate for uranium burn-up. Water coolant inside the aluminum container vessel is separate from the pool water. Orifices in the top and bottom reflectors allow water to circulate upwards through the core by natural convection. Heat from the circulating water in the container vessel is conducted through the container wall to the outer pool, and the pool water is cooled by an immersed coil supplied by city water. A single thermocouple monitors and records the core outlet temperature. The reactor is controlled, automatically or manually, by a single motor-driven cadmium absorber rod, located on the central axis of the fuel assembly. The control rod is worth 5.5 mk. Electrical signals from a self-powered neutron detector automatically control both startup and steady state operation over the range 5% to 100% full power. By physically limiting the excess reactivity to a small value less than prompt critical, and relying on the negative temperature and void coefficients as the primary safety mechanisms, it has been experimentally demonstrated that conventional electro-mechanical safety devices are unnecessary. Moreover, the skilled tradesmen normally required to test and maintain the safety devices are unnecessary; and because of the simple control system and inherent safety, there is no need for a full-time operator in the reactor room in automatic operation. This translates in significant savings in both capital and operating costs.

2.2 The Concept of the Homogeneous SLOWPOKE Reactor

The Homogeneous SLOWPOKE core consists in a cylindrical vessel filled with a homogeneous aqueous uranyl sulfate solution, as shown in Figure 2. This vessel is referred to as the inner reactor vessel. The inner vessel has the same diameter as of the SLOWPOKE-2 reactor core in order to fit within the cavity in the annular beryllium reflector annulus left when the core of a SLOWPOKE-2 reactor is removed at its end-of-core life. However, the height of the core is about twice its diameter. The Homogeneous SLOWPOKE reactor can be located on university campuses or within nuclear research centres and can fulfill and continue the same mission as for the SLOWPOKE-2 reactor, i.e. being a source of neutrons for Neutron Activation Analysis (NAA) and neutron radiography, among others, in support of educational programs and research. Moreover, the Homogeneous SLOWPOKE reactor can accomplish the extra mission of producing radioisotopes for applications in nuclear medicine and many other domains. For the Homogeneous SLOWPOKE nuclear reactor, particular attention is given to the homogeneous fuel mixture and associated potential corrosion and erosion problems for the reactor vessel and the heat transport conduits. The extraction of the wanted radioisotopes and other fission products must be done outside the reactor core. A significant amount of the irradiated fuel mixture thus represents radiation protection hazards which can be addressed using well known and proven technology and methods. In the present study, an aqueous uranyl sulfate solution (UO_2SO_4-H_2O) is proposed and the corrosion power of this solution as well as the possibility of precipitation are addressed. The radiation-induced decomposition of the moderator (water) is also considered as it may lead to the accumulation of gases which could become explosive mixtures of hydrogen and oxygen. The Homogeneous SLOWPOKE reactor has been designed as an inherently safe nuclear reactor, like the present SLOWPOKE-2 research reactors. Therefore, it can be operated continuously in automatic mode by crews of Certified Operators with relatively minimal training, resulting in substantial savings in operation costs.

3. Design Methodology

The full details of this research may be found in three theses written by graduate students at the Royal Military College of Canada [5-7]. The Homogeneous SLOWPOKE reactor was simulated on computer with the MCNP 5 (Monte Carlo N-Particle) code [8] which is a general-purpose, continuous energy, generalized-geometry, time-dependent, coupled neutron/photon/electron Monte Carlo transport code. For fissile systems, MCNP 5 has the ability to calculate the effective multiplication factor k_{eff} as an eigenvalue. The cross sections used with MCNP 5 were provided by the ENDF/B-V library and were adjusted to the temperatures by interpolating within the database and also by calculating the homogeneous mixture density for the temperature modeled. Table 1 provides the dimensions of the reactor. The results obtained with the MCNP 5 code were then compared with the results obtained by running a computer model of the reactor with the deterministic code WIMS-AECL Release 2-5d [9].

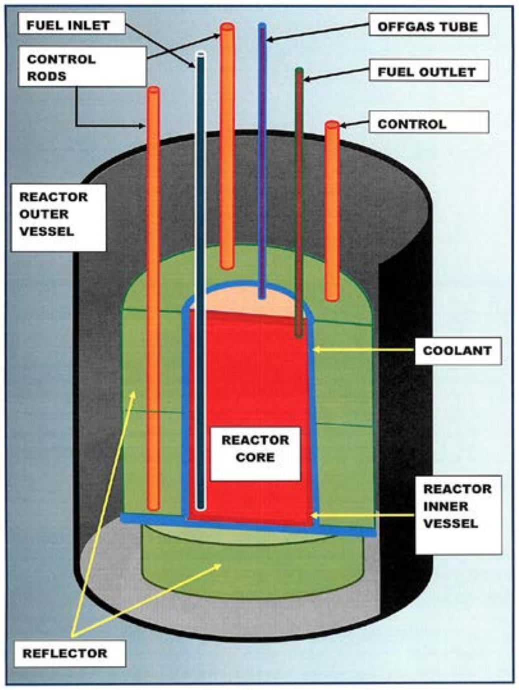

Figure 2: Simple representation of the Homogeneous SLOWPOKE reactor.

Because WIMS-AECL performs the simulation in two dimensions only, the axial buckling needs to be adjusted until the value of the effective multiplication factor obtained with WIMS-AECL matches the value of k_{eff} calculated by MCNP 5. Both codes then do agree very well on other results, such as the thermal and fast neutron distributions, the power density distributions and the nuclear reaction rates including the production rates of isotopes such as ^{99}Mo. The analysis then concentrated on the effects of the temperature on the reactivity of the reactor, on the control rod worth and the determination of the potential for inherent safety for this reactor concept. The simulations were carried out by varying the temperature of the reactor using both MCNP 5 and WIMS-AECL, using the ENDF/B-VI library and 89 groups of neutron energy. Figure 3 presents the effects of the temperature on the excess reactivity of the reactor and confirms the large negative coefficient of reactivity due to the temperature at high temperatures, which is an essential, but not sufficient, requirement for inherent safety. A more elaborate analysis was then performed with MCNP 5 and with the WIMS-AECL Version 3.1 [10]. Several materials compositions were investigated, with Table 1 presenting the retained selections. The fuel solution molality was adjusted within the final reactor configurations in order to yield excess reactivity values similar to those of the original SLOWPOKE-2 reactor (~3.8 mk). In order to determine the molality of the uranyl sulfate aqueous solution, the maximum allowable 20% uranium enrichment was maintained throughout this study. For that excess reactivity of 3.8 mk, the solution molality obtained was 1.46 mol kg^{-1} and 1.70 mol kg^{-1} for the model with the Zircaloy-2 reactor inner vessel and the model with the 347 Stainless Steel reactor inner vessel, respectively.

Water within the reactor vessel was maintained at the expected reactor operating temperature of 313 K (40 ^{o}C), this being the maximum value for the temperature foreseen in normal operation at steady state. As for the pool water temperature, a value of 293 K (20 ^{o}C) was assigned for this

Table 1: Characteristics of the final Homogeneous SLOWPOKE design.

Characteristic	Zircaloy-2 model type (Mark IIa)	Stainless Steel Type 347 model type (Mark IIb)
Core height	48.8 cm	48.8 cm
Core radius	10 cm	10 cm
Core vessel thickness	3 mm	3 mm
Vessel material	Zircaloy-2	Stainless Steel Type 347
Control rod material and radius Control rod cladding thickness	Cadmium 1.50 mm Al 2 mm	Cadmium 2.25 mm Al 2 mm
Control rod worth: 5 x 2 mm rods within inner sites 2 x 3.8 mm within inner sites	3.4 mk* - (-6.6 mk**) =10.0 mk 4.0 mk* - (-3.5 mk**) =7.5 mk	4.5 mk* - (-3.5 mk**) = 8.0 mk 4.0 mk* - (-3.3 mk**) =7.3 mk
Fuel volume (in core) at 40 ^{o}C	15.331 L	15.331 L
Reflector annulus: inner radius	11 cm	11 cm
Reflector annulus: outer radius	21 cm	21 cm
Beryllium radial reflector height	49.75 cm	49.75 cm
Beryllium/Graphite radial reflector height	22.0 cm (Be) + 27.0 cm (C)	-
Beryllium radial reflector thickness (extension part)	10 cm	10 cm
Graphite radial reflector thickness (extension part)	10 cm	-
Bottom reflector dimensions	Diameter: 28 cm; Height: 11 cm	Diameter: 28 cm; Height: 11 cm
Fuel	uranyl sulfate solution in water	uranyl sulfate solution in water
Fuel enrichment	20%	20%
Fuel concentration	1.4575 ± 0.0061 mol kg^{-1}	1.7058 ± 0.0033 mol kg^{-1}
U-235 Mass (in core)	994.2 g	1149 g
k_{eff}	1.0039 ± 0.001	1.0038 ± 0.001
Thermal power	20 kW	20 kW
Operating temperature at steady state	40 ^{o}C	40 ^{o}C
* : Control rod fully withdrawn	**: Control rod fully inserted	

water. An air-filled plenum was included in the reactor inner vessel above the fuel mixture surface and was modeled as water-saturated air at 313 K (40 ^{o}C). As a result of continuous operation of the reactor, the composition of this air plenum changes since hydrogen and oxygen accumulate in significant concentrations following radiolysis of fuel-water mixture. Gaseous radioisotopes also gradually accumulate in the air plenum such as ^{41}Ar and ^{135}Xe. The design of the reactor addresses the issue of removing these gases by including a gas removal system.

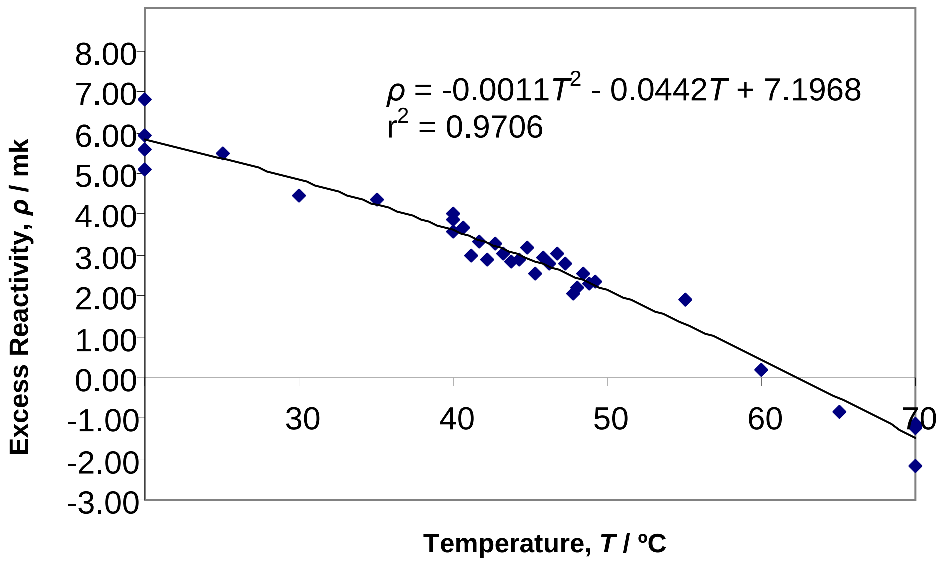

Figure 3: Excess reactivity estimated by individual Zircaloy-2 MCNP-5 models with a molality of 1.4575 mol kg^{-1}, normal operating fuel solution level, at various temperatures.

Two final models were retained for further studies: one with the reactor inner vessel made with Zircaloy-2 (Mark IIa) and the other with 347 Stainless Steel (Mark IIb). The characteristics of both models are presented in Table 1. While the stainless steel inner vessel appears to be the least expensive alternative for the reactor design, additional beryllium reflectors and higher fuel solution concentrations would be needed in addition to more cadmium control rods. The control rod system was designed as a cluster of cadmium rods that would slide in guide tubes located in the radial reflector. This option is justified since the thermal neutron flux in the radial reflector is almost equal to the value at the axis of the core. The fabrication of the inner reactor vessel is then much easier and the fuel mixture volume is increased. Several control rod configurations were then investigated. The control rod locations were chosen at the inner irradiation sites selected such as to ensure as much azimuthal uniformity as possible. In some of the models, the radius of the control rods was increased in order to limit the number of inner sites used for reactor control, thus freeing up more inner sites for NAA. The best configurations obtained had 2 x 3.8 mm rods and 3 x 3.2 mm rods for the reactor with a vessel made of Zircaloy-2 and 347 Stainless Steel, respectively. A 5-rod cluster arrangement minimized the variations of the thermal neutron flux in the azimuthal direction and the results for this configuration appear in Table 1.

4. Safety Analysis

A safety analysis was implemented aiming, among other goals, at assessing the possibilities of the Homogeneous SLOWPOKE reactor being inherently safe. The reactivity coefficients due to the temperature and the void fraction were determined from a series of simulations of the reactor using MCNP 5. Distributions of the excess reactivity of the reactor as a function of the temperature of the fuel-moderator solution were obtained through multiple runs of MCNP 5. For some key temperatures such as 20 ^{o}C, 40 ^{o}C, 70 ^{o}C and 90 ^{o}C, multiple runs were carried out and the reactivity recorded. Figure 3 (obtained from 36 MCNP 5 runs) presents a typical distribution for a reactor with a Zircaloy-2 reactor inner vessel under normal operation.

The reactivity coefficient due to the void fraction was determined using various approaches to simulate voiding. One of these approaches, referred to as "microbubble voiding", consisted in mimicking the void fraction by simply decreasing artificially the density of the homogeneous fuel solution. A second mode of voiding consisted in inserting two spherical bubbles within the reactor core according to a centred geometry, and then in varying the radii of these bubbles. A variant consisted in using 49 actual spherical bubbles and in varying their radii. Yet another mode consisted in using a single void cylinder of varying dimensions and centered within the reactor core. A sample of the results are presented in Table 2 for the case of the microbubble model at different temperatures, as the investigation determined that the microbubble model was the most conservative method for determining the reactivity coefficient due to the void fraction. All of the results show that voiding produces a strong negative reactivity, bringing the reactor to a subcritical state well before the temperature of the homogeneous fuel mixture becomes close to levels that would damage the reactor and its components.

Table 2: Tabulated estimation of the reactivity coefficient due to the void fraction (α_V) at various temperatures with calculation performed for a several fuel solution void fractions, using the "microbubble voiding" model.

Model Type	Temperature (ºC)	Curve-fitted Polynomial $\rho(x)$ (mk) (Note a)	x at $\rho = 0$ (Note b)	$\alpha_V = \frac{\partial \rho}{\partial x}$ (mk void^{-1})	α_V at x = (mk void^{-1}) 0.05	0.20
Zircaloy-2	20	$-0.0654\,x^2 - 4.5055\,x + 5.755$	0.0130	**-0.1308x - 4.5055**	-4.51	-4.53
	40	$-0.0622\,x^2 - 4.6007\,x + 3.6695$	0.0080	**-0.1244x - 4.6007**	-4.61	-4.63
	70	$-0.0674\,x^2 - 4.5479\,x - 1.5541$	n/a	**-0.1348x - 4.5479**	-4.55	-4.57
Stainless Steel Type 347	20	$-0.0677\,x^2 - 4.7863\,x + 5.7520$	0.0120	**-0.1254x - 4.7863**	-4.79	-4.81
	40	$-0.0720\,x^2 - 4.7391x + 3.7298$	0.0080	**-0.1440x - 4.7391**	-4.75	-4.77
	70	$-0.0788\,x^2 - 4.6455\,x - 1.9309$	n/a	**-0.1576x - 4.6455**	-4.65	-4.68

Notes: a) - the void fraction (**x**) used for Table 2 is expressed simply as the fraction of the volume occupied by void. Since the Homogeneous SLOWPOKE reactor core volume is 15.331 L, then a volume fraction of 0.01 (1%) is simply 0.15331 L or 153.31 cm^3. Therefore, the reactivity per void fraction unit, if the unit is 1% void fraction, would be, for example, -4.51 mk void^{-1} = -4.51 mk (153.31 cm^3)$^{-1}$ = -2.94 x 10^{-2} mk cm^{-3}.
b) - neglecting the negative void fraction solution, except at T=70ºC, where both solutions are negative. Void solutions which are **x**< 0 have no physical meaning, therefore at 70ºC, the reactor is already subcritical, hence the use of "not applicable" (n/a) for the void fraction for which the reactor becomes subcritical.
c) - Of course, for reactor temperatures of 20, 40 and 70 ºC, there is no voiding due to ebullition. The data presented here are obtained from the aqueous uranyl sulfate solution's density for the various temperatures and used in the "microbubble voiding" model. The interpretation is that of the reactivity that would be obtained if a void fraction (**x**) is created artificially by means other than natural boiling in the homogeneous fuel mixture.

A third component of the safety analysis consisted in examining the thermal-hydraulics of the reactor system in order to provide answers to questions such as whether natural convection remains sufficient to ensure the cooling of the reactor in normal and abnormal operating conditions. In order to fully confirm inherent safety, modelling of the Homogeneous SLOWPOKE reactor through several transient states was carried out [7] and the results demonstrate that the Homogeneous SLOWPOKE reactor is indeed inherently safe from simulations in non-steady state conditions. Using the software COMSOL Multiphysics [11] to carry out the thermal-hydraulics simulations, this study has shown that natural convection is sufficient to cool down the reactor, even following severe transients. Figure 4 illustrates the results of several transients to which the reactor was submitted and shows how the reactor responds to these transients without developing excessive temperatures. The pronounced self-regulating nature of the Homogeneous SLOWPOKE reactor is demonstrated, with the temperatures kept well below the boiling point of the aqueous uranyl sulfate solution.

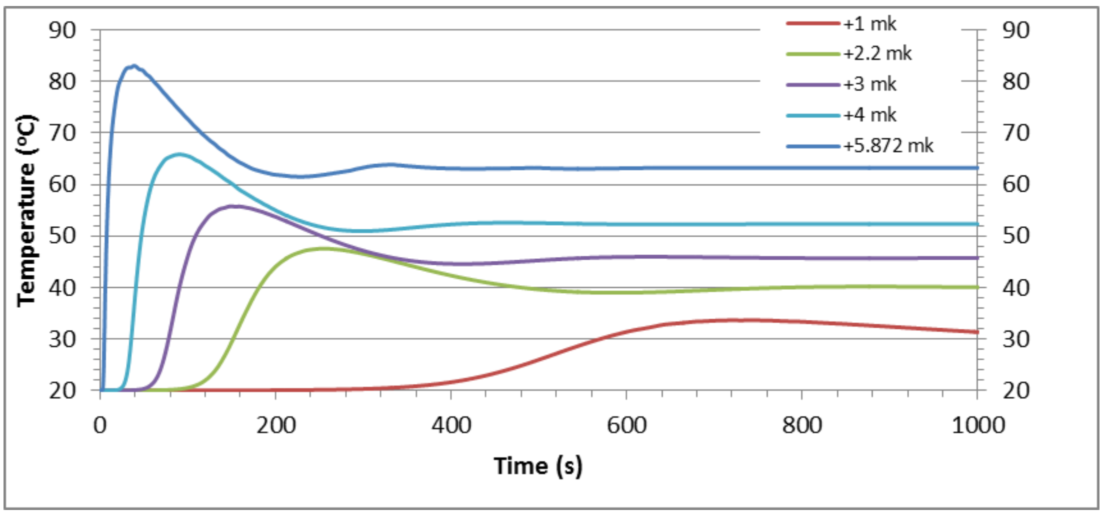

Figure 4: Average core temperature during various power transients initiated at t = 0 s. (Mark IIa model).

5. Discussion and conclusion.

In order to validate the results of the reactor physics calculations, an actual Homogeneous SLOWPOKE reactor needs to be built and tested, which is not (yet) the case here. The next best approach to acquire confidence in the results of the simulations is to use two very different reactor simulation codes considered reliable in the nuclear industry: MCNP 5 and WIMS-AECL. MCNP 5 is an improved version of MCNP 4A which was used by Pierre [12] for his successful model of the heterogeneous LEU-fueled SLOWPOKE-2 reactor at RMC, with the results of his simulations in very good agreement with the commissioning data obtained by the AECL commissioning team when the reactor was installed in 1985. As for WIMS-AECL, this code is an Industry Standard Tool currently used in the industry for the support of existing CANDU reactors and the development of Generation III and IV reactors. The authors' experience [13] with WIMS-AECL allows them to estimate the accuracy within ±5% (at most), with an extra 1% added for the fact that here the simulation is carried out with a two-dimension model.

In order to further verify the numerical results, MCNP 5 was tested against two benchmark homogeneous reactor cases found in the Nuclear Energy Agency's International Handbook of Evaluated Criticality Safety Benchmark Experiments [14-15]: the LEU-SOL-THERM-002 and the LEU-SOL-THERM-004 cases. The LEU-SOL-THERM-002 system considers a 174 L (34.3990 cm inner radius) sphere of low enriched (4.9%) uranium oxyfluoride (UO_2F_2) solution. The experiments using this system to calculate k_{eff} were part of a series of measurements performed at Oak Ridge National Lab during the 1950's. The spherical reactor is constructed of 1/16 in. thick aluminum and supported by only the top and bottom overflow and feed tubes. Experiments 1 and 2 determined the uranium concentration and chemical composition of the fuel solution with which the sphere is critical when completely filled with and without a water reflector, respectively. Experiment 3 used the same fuel solution as Experiment 2, however, a water reflector was employed and the fuel solution was brought to the level of criticality. In this case, the sphere was 83% filled when criticality was achieved.

For the MCNP model, the feeder and overflow tubes were not included to simplify the geometry and the space outside of the 15 cm thick water reflector is considered to be a void space. The temperature of the solution is assumed to be 298K (25 ^{o}C) and the temperature of the water reflector is assumed to be 300K (27 ^{o}C). Table 3 presents the results of the simulations for the benchmark cases [16]. The second column of Table 3 lists the "experimental k_{eff} values" reported. The actual k_{eff} values for these experiments were not recorded, however, in each case it was recorded that the reactor was critical and was held there for at least ten minutes. When the experimental errors are accounted for, the multiplication factors found for the three experiments are the ones reported in the second column of the table. For each experiment tested using MCNP 5, the resulting k_{eff} value is found to be within 1% error of the benchmark k_{eff} value. The slight discrepancy between the MCNP 5 values found here and the MCNP values found in the report may be explained from a different number of Monte Carlo iterations used in the benchmark report or from slight differences in nuclide values used in the report calculations.

The LEU-SOL-THERM-004 case was experimentally tested on the Static Experiment Critical Facility (STACY) in 1995 at the Tokai Research Establishment of the Japan Atomic Energy Research Institute. The STACY reactor is a cylindrical stainless steel tank (150 cm length by 60 cm diameter) filled with 10% enriched uranyl nitrate solution and is water reflected. The fuel solution level in the tank was adjusted to find criticality with various uranium concentrations in the solution ranging from 225 to 310 gU L^{-1}. The core is submerged in a tank of light water which acts as a reflector for the reactor core. The size of the water tank is 202.0 cm wide by 402.0 cm long by 240.0 cm high. The system's criticality is controlled by both control rods and solution level. The results for the simulations using MCNP 5 are shown in Table 3 (shaded bold column) along with the results found experimentally and from simulations with other software. All calculated MCNP 5 results here are again within 1% error of the benchmark experimental results, providing good agreement. The discrepancy between the MCNP 5 and the benchmark results comes likely from the inability to use the Continuous-Energy JENDL-3.2 cross section for MCNP 5 results calculated here. However, using the Continuous Energy ENDF60 W-184 cross section table, more accurate results were found.

Table 3: Comparison of the MCNP 5 results for LEU-SOL-THERM benchmark cases.

Simulation results for the LEU-SOL-THERM-002 benchmark case.

Code (Cross Section Set)→ Experiment Number ↓	Benchmark k_{eff}	KENO (Hansen-Roach)	KENO (27-group ENDF/B-IV)	MCNP (Continuous-Energy ENDF/B-V)	ONEDANT (27-Group ENDF/B-IV)	**MCNP5 (Continuous-Energy ENDF/B-V)**
1	1.0038 +/- 0.0040	1.0077 +/- 0.0011	0.9952 +/- 0.0011	1.0006 +/- 0.0004	0.9981	**1.0029 +/- 0.0004**
2	1.0024 +/- 0.0037	1.0039 +/- 0.0012	0.9930 +/- 0.0011	0.9963 +/- 0.0004	0.9946	**0.9973 +/- 0.0004**
3	1.0024 +/- 0.0044	1.0094 +/- 0.0012	0.9937 +/- 0.0011	1.0012 +/- 0.0004	---	**1.0016 +/- 0.0004**

Simulation results for the LEU-SOL-THERM-004 benchmark case.

Code (Cross Section Set)→ Experiment Number ↓	Benchmark k_{eff}	MULTI-KENO (137-Group MGCL based on JENDL-3.2)	MCNP (Continuous-Energy JENDL-3.2)	**MCNP (Continuous-Energy ENDF60 W-184)**
1	0.9994 +/- 0.0008	0.9959 +/- 0.0007	1.0072 +/- 0.0006	**0.9992 +/- 0.0006**
29	0.9999 +/- 0.0009	0.9976 +/- 0.0007	1.0075 +/- 0.0006	**0.9996 +/- 0.0006**
33	0.9999 +/- 0.0009	0.9949 +/- 0.0006	1.0053 +/- 0.0006	**0.9984 +/- 0.0006**

Replacing the solid fuel core of the SLOWPOKE-2 reactor with liquid fuel for the Homogeneous SLOWPOKE reactor affects the reactivity calculation in five important ways: no self-shielding in the liquid fuel/moderator; no separate temperature reactivity coefficients for fuel and moderator/coolant; an additional power-dependent void coefficient from radiolytic hydrogen and oxygen gas bubbles; neutron absorption in the inner vessel wall and the light water gap between the inner vessel wall and the inner wall of the reflector; and increased neutron leakage because of increased height and core volume. The neutronic analysis is considerably more complex and better tools than the 2-D WIMS-AECL exist and should be used for the continuation of the present research. A challenging follow-up of the present research would consider the development of a simulation tool coupling the neutronic calculations with a thermal-hydraulic model producing accurate temperature/void/delayed neutron distributions. This research on the design of the Homogeneous SLOWPOKE reactor has produced indeed encouraging results such that the construction of a prototype reactor may now be considered. A companion presentation will presents the results of an analysis of the ^{99}Mo production rates from a Homogeneous SLOWPOKE reactor.

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