DRAGON MODELLING OF THE SLOWPOKE-2 REACTOR AT ÉCOLE POLYTECHNIQUE

G. MARLEAU, S. NOCEIR, R. ROY AND D. ROZON

Institut de génie nucléaire
École Polytechnique de Montréal, Montréal, Québec H3C 3A7
e-mail: marleau@meca.polymtl.ca

ABSTRACT

Predicting the behaviour of the SLOWPOKE-2 reactor at École Polytechnique de Montréal poses a major challenge from the point of view of transport calculations since it has properties which forbid the use such simple schemes as those considered for power reactor analyzes. For example, the use of a totally reflected 2–D cell containing a single fuel pin with its associated coolant is not good enough to generate the fuel averaged properties required in a full core diffusion calculation. Here we present the DRAGON cell model which was used to generate the fewgroup macroscopic cross sections needed for full core diffusion calculations. We also discuss how DRAGON can be used to evaluate and explain the effect on reactivity of temperature and local voiding perturbations.

I. Introduction

The use of cell models in reactor calculations generally embodies three different concepts. For a reactor core which is made up of a regular array of identical heterogeneous subregions, the cell represents the smallest of these subregions. The cell model also refers to the fact that these heterogeneous subregions can be replaced in a fewgroup full core calculation by an equivalent homogeneous cell. Finally it refers to the technique used for the computation of the multigroup neutron flux distribution required for the previous homogenization and condensation process.

For CANDU reactors, the cell model generally refers to a 2–D cut of a fuel bundle surrounded by part of the moderator. Since the fuel channels are arranged in a very large regular array, which can be assumed to extend to infinity in all spatial directions, the fine structure neutron flux distribution is obtained by solving the multigroup transport equation in a totally reflected cell. Then a unique set of homogenized and condensed cross sections is generated for each cell using the transport flux distribution.

On the other hand, for the small and highly heterogeneous SLOWPOKE-2 reactors, the process of defining a coherent cell is much more difficult. In fact, it is impossible to define a unique cell satisfying all the above requirements.[1] For example, even if the fuel pins in this reactor are located in a uniform 2–D hexagonal array, all the possible fuel locations are not necessarily occupied. This fuel array is also inserted in an annular calandria containing a Beryllium and water reflector (both being annular). Finally the fuel in the core, which has a finite height, is located between the top and the bottom Beryllium reflector plates.

In this paper we will present the DRAGON calculation models which have been used to generate the macroscopic cross sections required for a full core diffusion calculation.[2] We will also present some in-depth DRAGON calculations which were used to evaluate the effect on reactivity of local temperature variations and voiding perturbations. These are important to obtain a better knowledge of the physical process dominating various regions of the reactor and to understand the coupling between these perturbations.

II. The DRAGON model

The description of a SLOWPOKE-2 cell for the code DRAGON is divided into three parts. First, we present the geometry that will be treated using DRAGON as well as the various assumptions resulting from this specific choice of cell. In the second part we describe the DRAGON sequence of calculations, including a description of the library, the self-shielding and leakage methods considered. Finally, we discuss the homogenization and condensation models that can be used to generate a reliable reactor database.

II.A. The Geometry

We will consider here a cell calculation model for the SLOWPOKE-2 which represents a two dimensional cut of the full reactor core. This passage from 3 to 2 dimensions can be justified by the fact that the axial neutron flux distribution is relatively uniform inside the core due to the presence of the top and bottom Beryllium reflectors. In addition, we think that most of the errors resulting from this approximation can be compensated by the use of an adequate leakage model in the DRAGON calculation. However, further simplifications such as the use of a 2–D model for a single fuel pin (or a set of pins) surrounded by part of the coolant as our reference cell would not produce a sufficiently realistic description of the core to justify their use.[3]

Many different variations of this cell can be considered. However, because of the restrictions in the types of geometries that can be used with DRAGON, we will consider a 7 region annular cell where the central water region (without the control rod), the control rod guide tube, the coolant region, the Beryllium reflector, the water reflector, the calandria and part of the water in the pool which extends to 35.56 cm (see Fig. 1) are explicitly represented.[3] In addition, these annular regions will be further subdivided into respectively 5, 1, 48, 50, 41, 4 and 25 subregions of equal thickness in order to ensure the convergence of the radial flux distribution inside the core.

The 5 detector sites located in the Beryllium reflector are also represented explicitly in our model by inserting them in the form of pin cluster in the global annular geometry.

For the fuel sites, two different options were considered. First we chose to simulate explicitly only the sites which are fueled (see Fig. 2 for the LEU fueled reactor and Fig. 3 for the HEU fueled reactor). First, a two region annular cluster model was considered, the central part of the pin containing the Uranium mixture and the outer part representing the sheathing. As a consequence both the gap and the sheathing surrounding the Uranium fuel region were combined by reducing the effective density of the sheathing material. Simulations were also performed for a more precise three region fuel pin, where the gap region was treated independently. However, the results we obtained showed that the

previous approximation was entirely adequate. We also considered the case where the empty fuel sites are modeled explicitly (see Fig. 2) using a cluster model. In this case additional annular pins, having the same dimension of the fuel pins but containing coolant water were also treated explicitly in the model.

Finally, we have considered a totally reflected cell in such a way that the neutrons reaching the outer core limit are uniformly reflected towards its center.

Now let us consider the effect of these successive approximations on the expected precision of the model. Because the coolant, the Beryllium and the water reflector are represented in this model by annular regions, the neutron flux in these regions will be independent of the angular position. Accordingly, all the pins located at the same radial position with respect to the center of the cell will be surrounded by coolant region of identical thermal flux irrespective of their explicit angular location. Because the main source of thermal neutrons in the core is located in the coolant, the fission rate in the fuel pins will be controlled by an angularly uniform neutron source. As a result, it would be difficult to determine which fuel pin in a given ring sees the highest neutron flux. We will therefore associate a single flux to all the pins located at a given radial position, even if their explicit location in the core will be tracked by the collision probability model. For the same reason, the neutron flux inside each of the 5 irradiation sites will be assumed identical.

The reflective boundary condition we selected may also look strange since the full reactor was considered and one would expect that void boundary conditions would be more appropriate. However, because we have extended the cell in such a way as to reach the pool region where the flux is quite low, both types of boundary conditions should produce nearly identical results.

II.B. The Calculation Options

In order to generate a complete DRAGON input file, one has to select, in addition to the cell model, specific calculation options. These include the choice of a specific cross section library and a collision probability integration technique. Here we will try to describe and justify the main calculation options which were used for all of our analysis.

Our calculations were performed using the ENDF/B-V WIMS–AECL format library since it is a more recent library and it contains temperature dependent cross sections for the Beryllium.[4]

The self-shielding calculations were performed using the same geometry as that used in the transport calculation with the following difference: the annular discretization described above for the coolant, Beryllium reflector region, etc. was not taken into account for these calculations.

On the other hand, the collision probabilities required for both the transport and self-shielding calculations were integrated using the same technique, namely the isotropic collision probability method provided by the EXCELL module of DRAGON. The same integration parameters were also used for both cases.

The final option we need to discuss is the leakage option that was selected for the flux calculation. Because the SLOWPOKE-2 reactor is a small reactor with large leakage both in the radial and axial directions we felt that a K_∞ eigenvalue calculation would not

be reliable. Accordingly, we first performed a flux calculation with buckling search using the B_1 homogeneous leakage method on a reference core at a uniform temperature of 20 C. The remaining calculations (perturbations) were then performed for a K_{eff} eigenvalue search with an imposed buckling corresponding to that which makes the reference cell critical. For all our calculations involving a LEU fueled reactor core a constant buckling of $B^2 = 7.67 \times 10^{-3}$ cm^{-2} was used while for the HEU core studies we selected $B^2 = 6.90 \times 10^{-3}$ cm^{-2}.

II.C. The Homogenization and Condensation Procedure

As we mentioned above, in the case where cell calculations are performed for power reactors, a two group condensation of the completely homogenized cell cross sections leads to reactor properties which are more than adequate for core calculations. For the SLOWPOKE-2 reactor such an approximation is no longer valid because of the large variation in the flux spectrum as one moved from one region of the core to another.

In fact the condensation of the core properties to two energy groups is insufficient and a minimum of 6 carefully selected energy groups must considered to ensure that the variations of the flux spectrum in the core is correctly represented. In addition, it is important to generate at least one set of averaged fuel cross sections (including some of the coolant) as well as two sets of independent reflector properties (one for the Beryllium and one for the water reflector). Even the generation of a single set of fuel properties is highly questionable, and one should preferably consider at least three or four sets of fuel properties to account for the radial variation of the neutron flux spectrum in this core. This has not been done for the presimulation of the commissioning tests.[7, 8]

III. Perturbation Calculations with DRAGON

A large number of simulations were performed using DRAGON to examine the effect of various type of perturbations on the cell reactivity. These were prompted by the need to better understand the current HEU fueled SLOWPOKE-2 reactor as well as the proposed LEU core.[5] Here we will first present the effect of varying the temperature of various material regions on the reactivity of this cell. We will also discuss the effect on the reactivity of inserting different void volumes at various locations inside the core.

III.A. Temperature Reactivity Coefficients

The first reactivity effect we will investigate results from changing the temperature of four different regions in the core. These temperature variations run from 20 C to 80 C for the coolant, the Beryllium and water reflector while the fuel temperature variations range from 20 to 180 C. Note that the effect of increasing the temperature of a given material can also affect its density. In our studies we only took this effect into account for the regions containing water, namely the coolant and water reflector regions.

The results are presented in Table 1 for both the LEU and HEU fueled reactors. In the case of the LEU core, one can see that both the coolant and the fuel temperature reactivity coefficients are negative while it is positive for the water reflector and negligible for the Beryllium reflector . For the HEU core, the same general trend is observed except

that the fuel temperature reactivity coefficient is now very small and slightly positive. This last difference is mainly due to the relatively large concentration of ^{238}U in the LEU fuel as compared to the HEU fuel. The increased absorption in ^{238}U resulting from the Doppler broadening at higher temperatures is therefore more evident in this type of fuel.

For the LEU core we also observe that the reactivity coefficient resulting from variations in the coolant temperature is somewhat smaller than that computed for the HEU core. This is because the LEU core contains fewer fuel pins (more coolant water). Accordingly an increase in the temperature of the coolant which results in a reduction in its density has a smaller effect on reduction of the global reactivity of the core. This is due to the fact that the water in a SLOWPOKE-2 plays two opposing roles. First it is used to slow-down the neutrons which will have for effect to increase the reactivity of the core. However the relatively large absorption cross section of the Hydrogen in water is such that it can also be considered as a neutron absorber. This can be easily appreciated by looking at the reactivity effect resulting from an increase in the temperature of the water reflector which has a different sign than that observed for the coolant water. The reduction in the density of the reflector water is dominated by the reduction in the background absorption while its effect on the coolant water is dominated by the reduction in neutron thermalization. In the case where the amount of coolant water is more important, as for the LEU core, the reduction in the thermalization, while still being the dominant contribution is partially compensated by the reduction in neutron absorption in the additional water.

A final comment concerns the fact that the reactivity effects due to the individual temperature perturbation are nearly independent as can be observed in Table 1 where the sum of individual reactivity contribution corresponding to the maximum temperature perturbation (Sum) is compared to the effect observed for a calculation where the maximum temperature are applied simultaneously on the core (All).

From these observations we can derive the following reactivity temperature relation for the fuel ($\Delta K^{\mathrm{F}}_{\mathrm{eff}}$ in mk) in a LEU core:

$$\Delta K^{\mathrm{F}}_{\mathrm{eff}}(T) = [0.01 \pm 0.01] - [0.01 \pm 0001](T - 20)$$

while for the coolant ($\Delta K^{\mathrm{C}}_{\mathrm{eff}}$ in mk) and water reflector ($\Delta K^{\mathrm{W}}_{\mathrm{eff}}$ in mk) we obtained

$$\Delta K^{\mathrm{C}}_{\mathrm{eff}}(T) = [0.01 \pm 0.05] - [0.095 \pm 0.004](T - 20) - [0.00128 \pm 0.00007](T - 20)^2$$

and

$$\Delta K^{\mathrm{W}}_{\mathrm{eff}}(T) = [0.00 \pm 0.01] + [0.036 \pm 0.001](T - 20) - [0.00026 \pm 0.00002](T - 20)^2$$

These relations were obtained using a regression analysis for a linear or quadratic fit.[6] For completeness, the statistical errors on each of the fit are also provided.

The above reactivity coefficients are only approximate since they were obtained using a 2–D model of the core. The actual reactivity coefficients to be used in point kinetic calculations[7] will in fact be obtained by carrying out, using DONJON, 3–D diffusion calculations.[8]

III.B. Void Reactivity Coefficients

The main reason for investigating the effect on reactivity of void insertion in the core is that during power excursions in a SLOWPOKE reactor, sub cooled nucleate boiling

could take place around the fuel pins. A related problem is that one does not necessarily know if the voided regions (bubbles) will stay localized around the fuel pins or if the bubbles will detach and condense thereby distributing the void more uniformly in the coolant. Moreover, as we observed above, an indirect effect of increasing the temperature of the coolant region is to decrease its density which can be interpreted as an increase in the void in the core. Here we would like to characterize independently the effect of void insertion in the core. Accordingly, we would like to know if a given amount of void localized at various locations inside the core has the same effect as an equivalent reduction in the coolant water density representing a distributed void.

We investigated the effect of inserting inside the reactor core a volume of void equivalent to that occupied by the water located in the control rod guide tube. This results in a reduction of the coolant volume of 1.214 % for the LEU core and 1.292 % for the HEU core. The results we obtained from such a numerical experiment are presented in Table 2 where Center represent the case where the void is localized in the guide tube, Pin is for the case where it is distributed uniformly around each fuel pin and Density represents the case where it is distributed uniformly in the coolant via a reduction of the coolant density.

The first observation is that the reactivity of the reference cell is lower by 1.5 mk than that obtained in the previous calculations. The reason for this behaviour is that all the calculations were performed in this case using a slightly different cell model where an additional annular region was defined around each of the fuel pins while the imposed buckling remains constant. This new model was selected to ensure that the three different numerical experiments could be performed using exactly the same geometry therefore reducing the discretization errors.

The following conclusions can be reached:

- The void reactivity effect is relatively large even for a such a small amount of void;
- Localizing this amount of void in the guide tube is nearly equivalent to distributing it in the coolant by a reduction of its density;
- When the void is concentrated around the fuel pins rather than being distributed uniformly in the coolant, large differences in the void reactivity effect can be observed.

This last observation can be explained by the fact that the additional ring of void around the fuel pins is so thin that the code has difficulties simulating the problem and in fact only sees part of the void volume thus inserted.

Finally one can find in Fig. 4 a plot of the change in reactivity as a function of coolant density for the LEU core where three of the points were taken from Table 1 and therefore also include temperature corrections in the coolant cross sections while the fourth point was taken from Table 2 for calculations at the reference temperature. Apart from a slight deviation for the result at the reference temperature, all these points nearly lie on a straight line. This confirms that the interpretation of the temperature corrections mainly in terms of density correction in the previous section is justified for the LEU core.

IV. Conclusions

We think that this 2–D cell model for the SLOWPOKE-2 reactor, while being far from perfect, is adequate for the purpose of generating a fewgroup database for core diffusion calculations. In fact, our main concern is that it cannot be used to identify the hot fuel pins directly in the cell calculation. This model is also useful to get a better understanding of the differences between a HEU and LEU fueled reactor core.

ACKNOWLEDGMENTS

This work was supported in part by grants from the Natural Sciences and Engineering Research Council of Canada (NSERC).

REFERENCES

[1] B.M. Townes and J.W. Hilborn, *"The SLOWPOKE-2 Reactor with Low Enriched Uranium Oxide Fuel"*, Annual conference of the CNS, Ottawa, June 1985.

[2] G. Marleau, A. Hébert and R. Roy, "New Computational Methods Used in the Lattice Code DRAGON", Topical Meeting on Advances in Reactor Physics, Charleston, SC, March 8–11 1992; see also G. Marleau, A. Hébert and R. Roy, "A User's Guide for DRAGON", Report IGE–174 Rev.1, École Polytechnique de Montréal, March 1996.

[3] S. Noceir, E. Varin and R. Roy, *"HEU and LEU Fueled SLOWPOKE Reactor Modeling"*, Report IGE–216, École Polytechnique de Montréal, October 1996.

[4] J.V. Donnelly, *"WIMS-CRNL, A User's Manual for the Chalk River Version of WIMS"*, AECL-8955, Atomic Energy of Canada Limited, January 1986.

[5] G. Kennedy and G. Marleau, *"Refueling the SLOWPOKE-2 Reactor at École Polytechnique: Procedure and Proposed Experiments"*, 18th Annual Conference of the Canadian Nuclear Society, Toronto, Ontario, June 8–11 1997.

[6] Wolfram Reaseach, *"Mathematica 3.0 Standard Add-on Packages"*, Cambridge University Press, Cambridge, U.K. (1996).

[7] D. Rozon and S. Kaveh, *"SLOWKIN: A Simplified Model for the Simulation of Reactor Transients in SLOWPOKE-2"*, 18th Annual Conference of the Canadian Nuclear Society, Toronto, Ontario, June 8–11 1997.

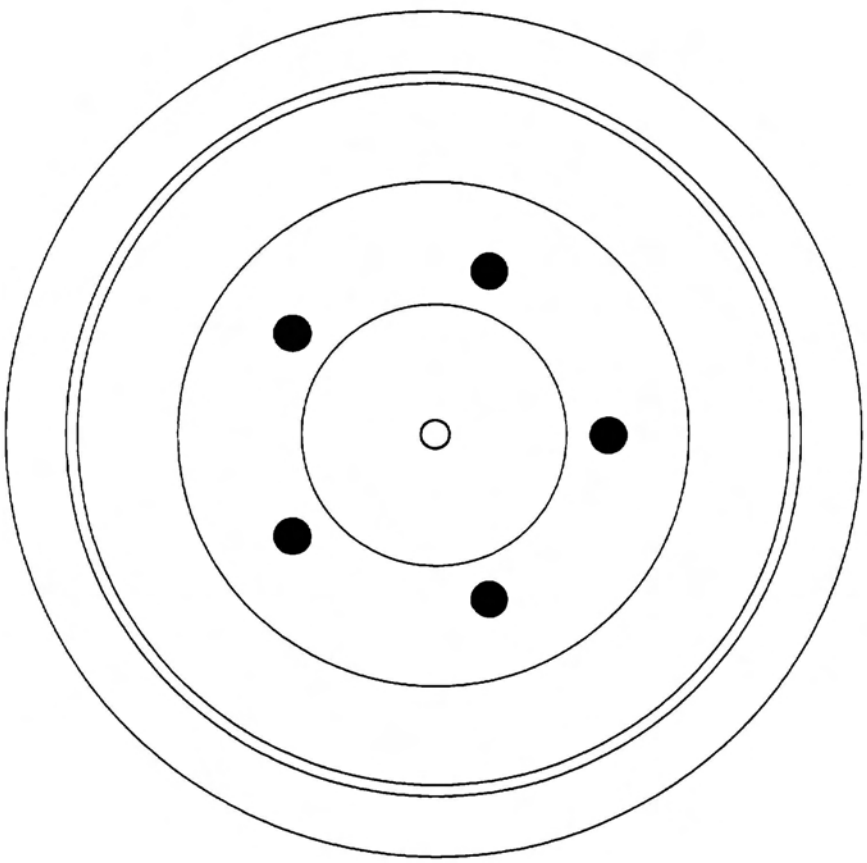
[8] S. Noceir, O. El Hajjaji, E. Varin, R. Roy and D. Rozon, *"Diffusion Calculations for the SLOWPOKE-2 Reactor Using DONJON"*, 18th Annual Conference of the Canadian Nuclear Society, Toronto, Ontario, June 8–11 1997.

Table 1: Reactivity as a function of temperature for a LEU and HEU core.

Type	ΔT (C)	LEU core		HEU core	
		K_{eff}	ΔK_{eff} (mk)	K_{eff}	ΔK_{eff} (mk)
Refe		1.00003	0.00	1.00004	0.00
Fuel	40	0.99960	-0.42	1.00024	0.20
	80	0.99919	-0.83	1.00039	0.34
	120	0.99883	-1.20	1.00039	0.34
	160	0.99843	-1.60	1.00036	0.31
Coolant	20	0.99766	-2.37	.99681	-3.24
	40	0.99415	-5.88	.99255	-7.50
	60	0.98974	-10.29	.98743	-12.61
Beryllium	20	1.00007	0.04	1.00017	0.12
reflector	40	1.00007	0.04	1.00020	0.15
	60	1.00009	0.06	1.00024	0.20
Water	20	1.00066	0.63	1.00073	0.68
reflector	40	1.00105	1.02	1.00125	1.20
	60	1.00126	1.23	1.00154	1.50
All		0.98948	-10.55	0.98942	-10.62
Sum		0.98943	-10.59	0.98944	-10.60

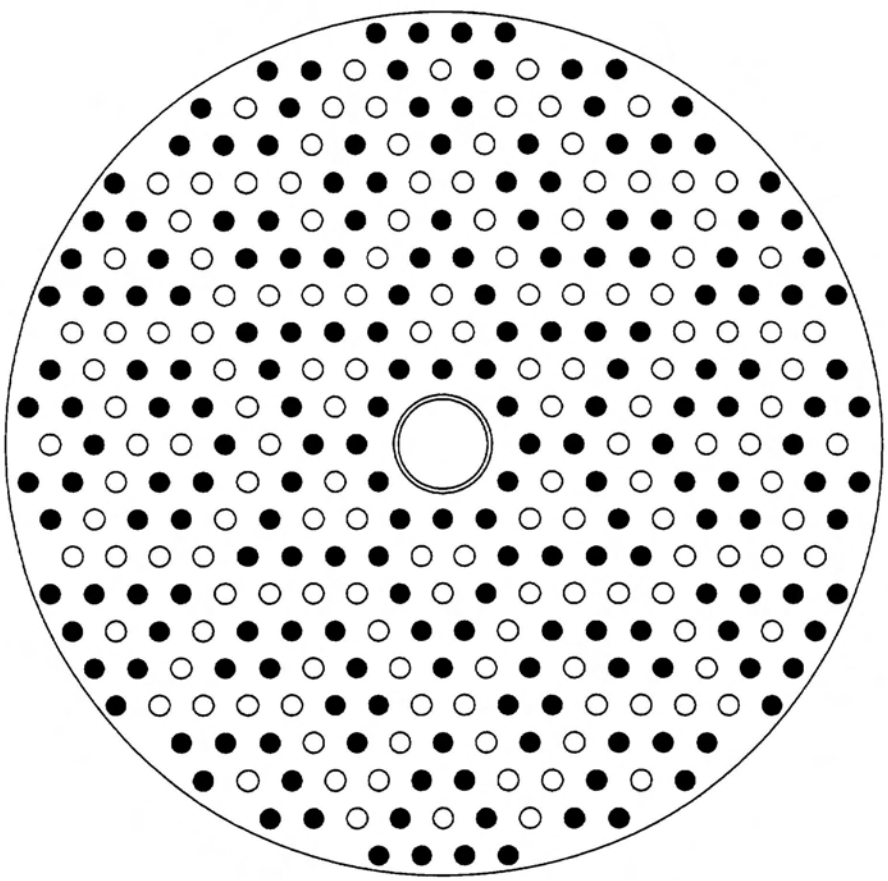
Table 2: Void effect for a LEU and HEU core.

Type	LEU core		HEU core	
	K_{eff}	ΔK_{eff} (mk)	K_{eff}	ΔK_{eff} (mk)
Ref	0.9989	0.0	0.9985	0.0
Density	0.9949	-3.9	0.9944	-4.1
Pin	0.9958	-3.1	0.9959	-2.6
Center	0.9948	-4.1	0.9948	-3.6



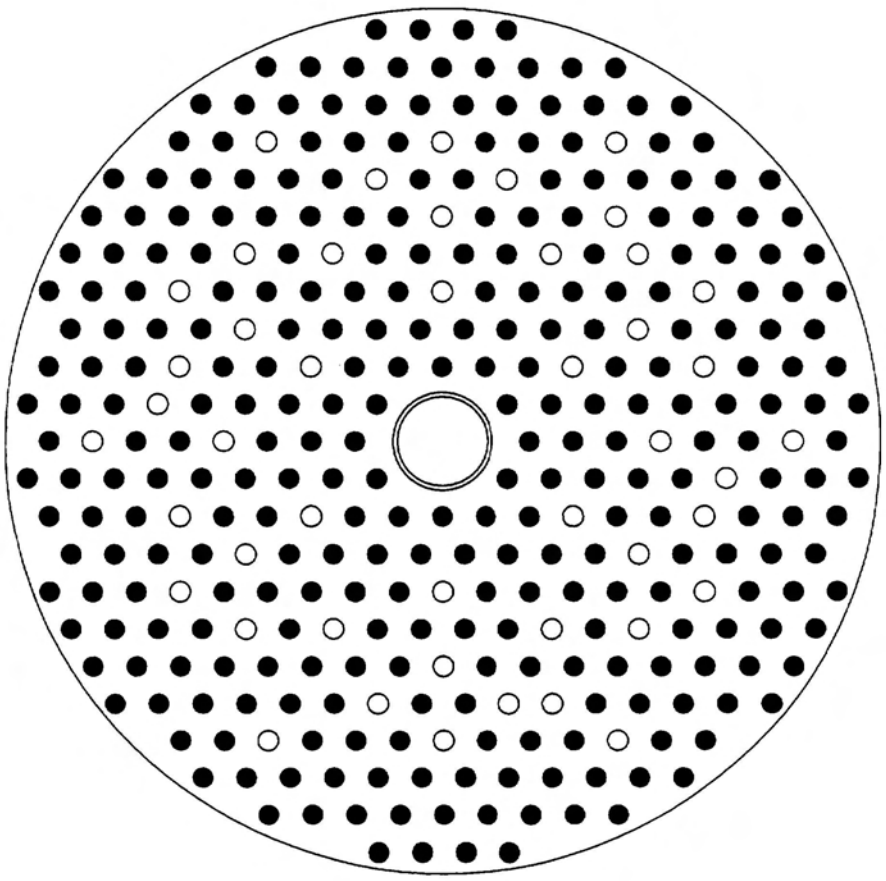
The filled circles in the above represent the detector sites while the central empty circle indicates the location of the control rod guide tube.

Figure 1: Overall SLOWPOKE geometry.



The filled circles in the above represent the loaded fuel sites while the empty circle indicate the location of the empty fuel sites.

Figure 2: Core geometry for the LEU fueled SLOWPOKE-2 reactor core.



The filled circles in the above represent the loaded fuel sites while the empty circle indicate the location of the empty fuel sites.

Figure 3: Core geometry for the HEU fueled SLOWPOKE-2 reactor core.

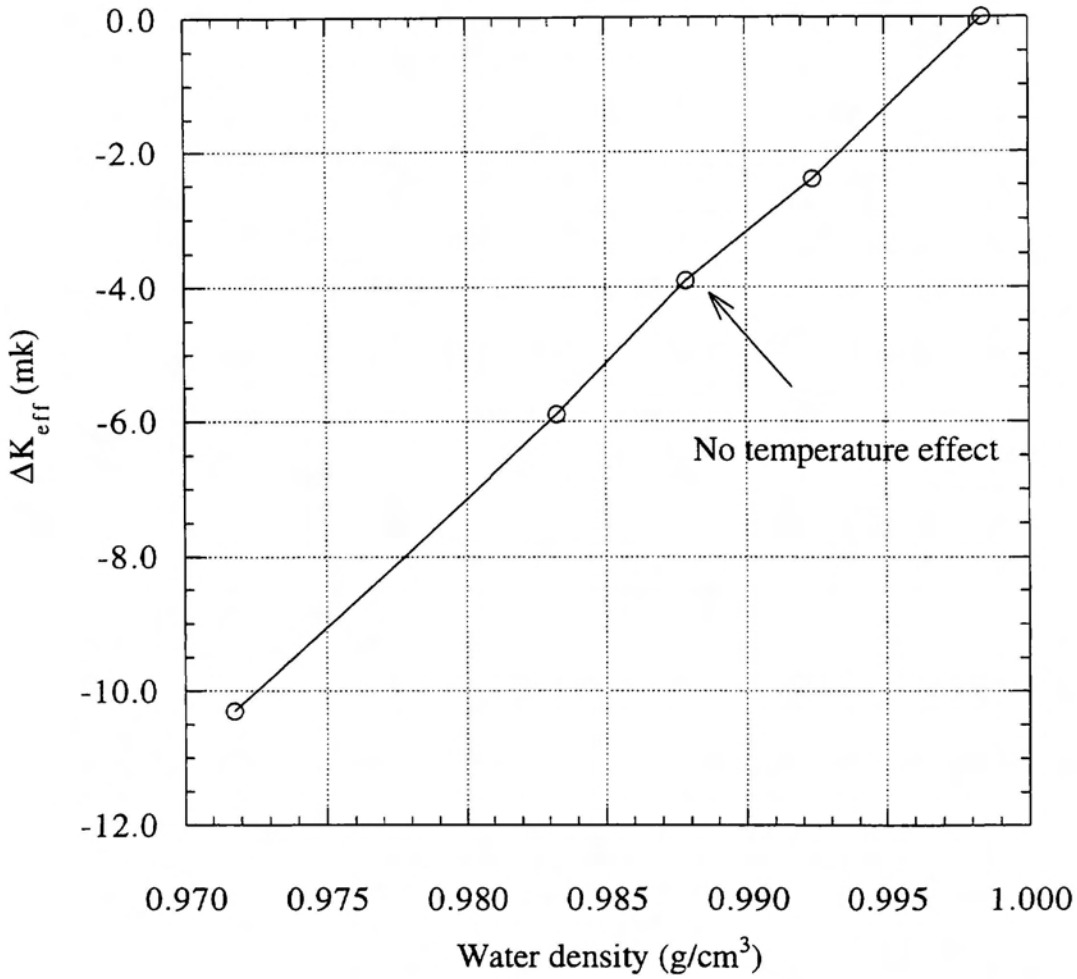


Figure 4: Change in reactivity as a function of the coolant water density for the LEU core.