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DETAILED CORE DESIGN AND FLOW COOLANT CONDITIONS FOR NEUTRON FLUX MAXIMIZATION IN RESEARCH REACTORS

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ABSTRACT

Following the design of the German research reactor, FRM-II, which delivers high thermal neutron fluxes, we have already developed an asymmetric compact cylindrical core with an inner and outer reflector. The goal was to maximize the maximum thermal flux to power ratio, which is a desirable characteristic of a modern research reactor. This design, for a 10 MW power, was analyzed using MCNP, ORIGEN2 and MONTEBURNS codes, considering a homogeneous mixture for the core material. Promising results showed that with standard fuel material of low enrichment uranium, high thermal fluxes are delivered by this core in the outer reflector. In addition, the life cycle, which is a limiting parameter regarding the compactness of the core, was calculated to be 41days.

In this paper, we report the results of recent developments in the design of this core. A detailed modeling of a suitable fuel element is performed with MCNP. Neutron fluxes are recalculated to assure that the same levels are achieved with this more detailed description. In this regard, three different zones with different flux levels were identified: a high neutron flux zone (4.0E14 $n{\cdot}cm^{-2}s^{-1}$), a moderate thermal neutron flux zone (2.5E14 $n{\cdot}cm^{-2}s^{-1}$), and a low thermal flux zone (1.0E14 $n{\cdot}cm^{-2}s^{-1}$). Heat deposition in the cladding, coolant and fuel material is also calculated to determine coolant flow rate, coolant outlet temperature and maximum fuel temperature for safe operating conditions.

INTRODUCTION

Neutrons are a powerful research tool, and are also useful in industrial and medical applications. Sources capable of producing high neutron fluxes are needed to perform material research, radiography and cancer therapy among other important uses. Research reactors (RR) are used to produce neutrons and to carry out related experiments. The flux and spectrum of these neutron sources and availability of irradiation facilities determine types of applications, and therefore, competitiveness of the reactor. Today, research reactors need to be considered not only as a tool to perform research, but also as an asset that satisfies industry's irradiation needs to assure sustainability of their operation. These facts continually demand development of new research reactors and improvements in the capability to produce neutrons and in facilities for their effective utilization. The aim of this study is to consider an alternative model for a new RR.

The simplest RR requires a core with a powerful and stable neutron source to be competitive. High neutron flux levels are associated with high thermal power levels (several MWs). This requirement adds to the complexity of the cooling system and the complexity of the reactor in general. Table I shows maximum thermal neutron fluxes in the reflector (MTF in units of $n{\cdot}cm^{-2}sec^{-1}$) and some other characteristics of some RRs that are operating, under construction or projected. Since MTF is proportional to the power of the reactor, the MTF normalized with power (MTF per Megawatt, MTF/MW) is a parameter that allows comparison of maximum fluxes for reactors with different power levels. As can be seen from Table I, the FRM-II German reactor [1] has the highest MTF/MW. Despite the fact that it was designed to operate with 93% enrichment, the compact construction of its fuel element is an attractive feature. This core produces a maximum thermal neutron flux at a distance of 12 cm from the surface of the fuel element: $8x10^{14}n{\cdot}cm^{-2}s^{-1}$ or $4x10^{13}n{\cdot}cm^{-2}s^{-1}MW^{-1}$ (undisturbed).

Table I. Maximum thermal fluxes, power and enrichment of some research reactors around the world.

Research Reactors	MTF $*10^{14}$	Enrich ment	Power (MW)	MTF/MW $*10^{13}$
HFIR (USA)	25.5	HEU	85	3.0
FRM-II (Germany)	8.0	93%	20	4.0
HANARO (Korea)	5.0	20%	30	1.7
JRR-3M (Japan)	3.0	20%	10	3.0
JHR[a] (France)	7.4	<20%	100	0.7
OPAL[b] (Australia)	3.2	20%	20	1.6

[a]Projected
[b]Under construction

In a previous work [2], we studied different homogeneous core configurations in a multipurpose pool type RR to maximize the intensity of the neutron sources or, in other words, the neutron fluxes delivered to the reflector. Specifically, our goal was a RR design with the following general features:

- Pool type reactor with a central core and heavy water as reflector (plenty of free space in the reflector),
- Annular core allowing irradiation positions with harder spectrum than in the reflector,
- Thermal power of 10 MW,
- Enrichment lower than 20%.

and with the following goals:

- Unperturbed thermal neutron flux peak greater than $4x10^{13}$ $n{\cdot}cm^{-2}s^{-1}MW^{-1}$,
- Life cycle greater than 40 days,
- Irradiation area in the inner reflector greater than 300 cm^2.

For a modern university RR, setting the power level at 10 MW seems to be appropriate. For this power level, the MTF/MW set as a goal allows obtaining neutron flux levels high enough to cover applications found in a modern multipurpose research reactor (see for instance OPAL RR). In addition, low power levels make the reactor simpler and its life cycle longer. Irradiation positions in the inner reflector is a desired feature that may allow irradiation of samples with harder spectrum than in the reflector pool without strongly perturbing the neutron fluxes there.

Setting the highest MTF/MW (FRM-II) to be our goal, we started our optimization studies from the core shape of the FRM-II (cylindrical fuel element). In the following sections of the paper we briefly discuss the steps followed to arrive at the basic design that fulfills the desired goals. A detailed modeling of the core follows.

HOMOGENEOUS AND AXISYMMETRIC CORE MODELING

In order to achieve the proposed goals, the first step was to develop different models with MCNP, Monte Carlo multi-particle transport code [3], to study neutron flux levels as a function of different parameters following the geometry of the FRM-II. In this step, we were not interested in a detailed description of the fuel element, and hence the core is modeled as a homogenous mixture of meat, cladding and coolant. It is assumed that a homogeneous representation of the core is sufficient to estimate thermal neutron flux levels in the reflector. To validate this claim, a homogeneous representation of the FRM-II core is modeled with MCNP and results are compared with those reported in literature.

The model is an annular cylindrical core (homogeneous mixture 17 v% fresh U_3Si_2, 21 v% cladding (aluminum) and 62 v% light water) with heavy water as external reflector and beryllium as internal one. The height of the active core region is 70 cm and a layer of 35 cm of light water was modeled over and below the core. The model is symmetric with respect to the $z = 0$ plane (z is the axial direction) and is azimuthally symmetric. Figure 1 (not to scale) indicates different mixtures modeled for the core and reflectors as well as their dimensions in centimeters [4, 5, 6]. Note that two different uranium densities were used in the core; 3.0 gr/cm^3 from 6.75 to 10.5 cm and 1.5 gr/cm^3 from 10.5 to 11.2 cm in the radial direction. The outer radius of the model was 250 cm and vacuum-boundary conditions were imposed over all external surfaces. No facilities were modeled in the reflector. All MCNP calculations showed in this paper are KCODE type calculations. In this particular case, FRM-II simplified model simulation, 500 cycles of 1000 particles per cycle were simulated (standard deviation in k_{eff} is less than 0.0015).

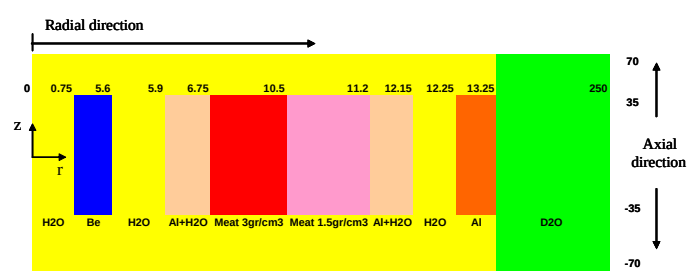

Figure 1. R-z view of the simplified model of the FRM-II (not to scale).

Fast and thermal (<0.625 eV) neutron flux distributions were calculated. The criticality condition at BOC was achieved by replacing the inner reflector (beryllium) by a control rod (CR) composed of aluminum (0.75-5.35 cm) and hafnium (5.35-5.6 cm). Figure 2 shows the radial variation of fast and thermal fluxes calculated at the central horizontal plane of the reactor (between $z = \pm 1$ cm) and rings of $\Delta r = 1$. Dots in Fig. 2 show non-perturbed thermal fluxes in the reflector for the FRM-II reactor as reported in Ref. 1. The purpose of this comparison is not to show quantitative agreement between fluxes reported in Ref. 1 and our simplified model. It is presented to show that a model based on a homogeneous description of the core yields enough details to allow parametric studies to maximize the thermal flux in the reflector for different "FRM-II core type" configurations.

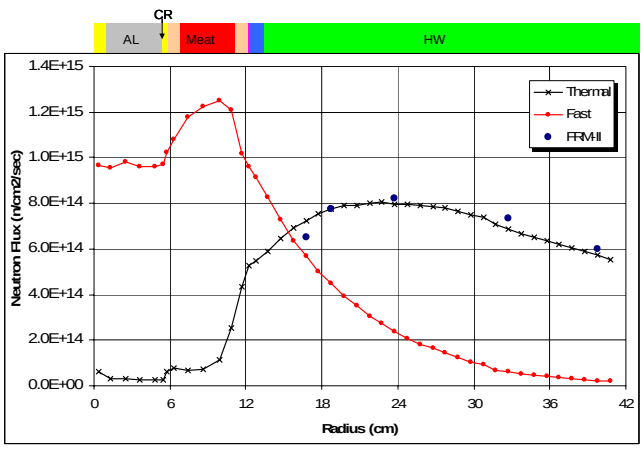

Figure 2. Radial profile of thermal and fast neutron fluxes in the simplified model of the FRM-II.

Elementary variations of the homogeneous FRM-II core model developed here were carried out to achieve the design goals with the specified constraints and to study neutron flux behavior. To allow an inner irradiation area, a central channel of 10 cm radius was modeled. The meat material was 20% enriched U_3Si_2 with a density of 4.8 gr/cm^3 (fresh). In addition, an inner reflector made of beryllium was modeled between the core and the central channel. All other dimensions are the same as in the FRM-II model.

The first set of simulations was carried out to study the effect of the thickness and the inner radius of the cylindrical fuel element (inner radius greater than 14 centimeters to allows space for inner irradiation positions) on the thermal neutron flux peak in the outer reflector. Recall that the first goal is to achieve a MTF of 4E14 n·cm^{-2}sec^{-1} (or 4E13 n·cm^{-2}sec^{-1}·MW^{-1}) in the heavy water reflector.

Table II shows the results of this parametric study. The amount of U5 is shown for comparison purposes (note that FRM-II has 0.325 KgU5/MW and a life cycle of about 52 days). Thermal neutron fluxes higher than 4x10^{14}n·cm^{-2}sec^{-1} were found with thinner fuel elements which allow greater neutron leakage to the reflector. However, the excess reactivity of these assemblies is not enough to sustain a critical core for an acceptable period of time or, in some cases, even to produce it. Note that the FRM-II core geometry is small enough to produce a leakage of more than 70% of neutrons produced in the core [1], and the amount of uranium 235 necessary to sustain a critical reaction for the desired core life is assured by a high enrichment.

Table II. MTF as a function of thickness and inner radius of the cylindrical fuel element.

Case	**I**	**II**	**III**	**IV**	**V**
Inner core radius (cm)	15	15	15	14	16
Outer core radius (cm)	20	19	18	17	20
Multiplication factor	1.15	1.10	1.03	1.02	1.12
U5 ratio to FRM-II (Kg/Kg) [a]	1.4	1.1	0.8	0.7	1.1
MTF (*10^{14})	2.31	2.74	3.06	3.11	2.58

Case	**VI**	**VII**	**VIII**	**IX**
Inner core radius (cm)	17	20	19	15
Outer core radius (cm)	20	23	20	16
Multiplication factor	1.07	1.12	0.87	0.78
U5 ratio to FRM-II (Kg/Kg) [a]	0.9	1.0	0.3	0.2
MTF (*10^{14})	2.77	2.55	4.13	4.47

[a]Normalized with power.

Clearly, for this annular, symmetric core shape, desirable thermal neutron fluxes and core life cannot be achieved simultaneously. In addition, the parametric study reported in Table II, as expected, shows that thick cores may have an acceptable life cycle (amount of U5 and multiplication factor) and thin ones produce the desired thermal neutron flux peak. Consequently, we varied the geometry and numerically experimented with asymmetric cores, seeking a compromise solution: a core that may produce a region in the reflector with high thermal neutron fluxes required for applications such as thermal beam and cold neutron sources, and a region in the reflector with moderate thermal neutron fluxes suitable for applications such as industrial processing. The core must also have an inner irradiation zone and an acceptable life cycle.

HOMOGENEOUS AND ASYMMETRIC CORE MODELING

The asymmetric core shown in Fig. 3 consists of two segments: a thinner part of angle θ, and a thicker part of angle (2π-θ). There is beryllium and light water in the center. Thinner meat section is padded with beryllium to make its inner and outer radii equal to those of the thicker meat section. A set of simulations were carried out to parametrically study the thermal neutron flux peak as a function of the inner and outer radii of both core sections and the aperture angle of the thinner part. As expected in this case, the neutron flux is not symmetric and the maximum of the thermal neutron flux is on a line passing through the center of the thicker core section. Results of this parametric study are given in Table III.

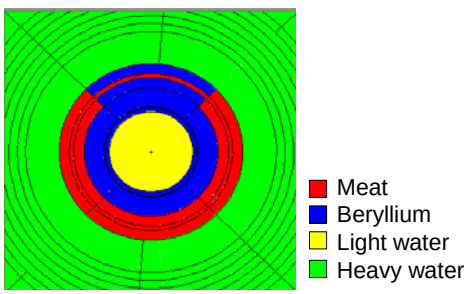

Figure 3. Asymmetric cylindrical core.

Results show that thermal neutron fluxes greater than $4x10^{14}$ $n{\cdot}cm^{-2}s^{-1}$ with acceptable life cycles (as indicated by multiplication factors at the BOC) are possible with this core configuration. For instance, case VI in Table III shows an excess of reactivity of 12% (Δk) and a MTF of $4.05x10^{14}{\cdot}cm^{-2}s^{-1}$. In this case, the minimum MTF on a line passing through the center in the thinner core section is $0.87x10^{14}$ $n{\cdot}cm^{-2}s^{-1}$. Due to asymmetry, instead of having an evenly distributed thermal flux in the reflector, this core produces both, a region with high thermal neutron fluxes and a region with low ones.

Table III. Parametric study for the asymmetric cylindrical core.

Case	I	II	III	IV	V	VI
Thicker core inner radius (cm)	15	15	16	15	16	16
Thicker core outer radius (cm)	20	20	22	20	22	22
Thiner core inner radius (cm)	17	17	18.5	17	18.5	16
Thiner core outer radius (cm)	18	18	19.5	18	19.5	17
Angular aperture thin section (°)	120	150	150	180	180	180
Multiplication factor	1.09	1.08	1.15	1.07	1.11	1.12
MTF center thin section (*10E14)	2.12	1.96	1.74	1.73	1.56	0.87
MTF center thick section (*10E14)	3.57	4.04	3.78	4.4	3.86	4.05

Case VI in Table III was selected to study the asymmetric model in more detail [2]. In the following section we briefly summarize the results of calculations performed to evaluate the core life of this design.

LIFE CYCLE OF ASYMETRIC HOMOGENEOUS CORE

Carrying out preliminary burnup calculation, it was found that Case VI in Table III must have an excess reactivity of about 14% (Δk / k) to produce the desired life cycle (note that the life cycle is desired to be greater than 40 days and the value shown in Table III for the multiplication factor yields an initial excess reactivity of 10.7% Δk / k). Therefore, it was decided to reduce the size of the light water center channel. Cores with increasingly smaller inner light water region were modeled. Light water in these models was replaced by beryllium. Results showed that a water hole with a radius of 7 cm, instead of 10 cm as in Case VI, leads to an excess reactivity of about 14% (Δk / k). The core of Case VI is shown in Fig. 4.

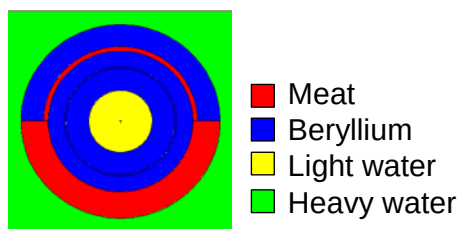

Figure 4. R-θ view of the MCNP model for Case VI of the asymmetric design.

Core life was estimated using ORIGEN2 [7] and MONTEBURNS 2.0 [8]. MONTEBURNS links MCNP transport code and ORIGEN2 burnup code. The MCNP model was 1/4 of the real core and each MCNP run involved 250 cycles of 1000 particles (standard deviation in k_{eff} was lower than 0.0025 and tallies in the core cells passed all statistical checks). As ORIGEN2 is a zero dimensional code, within MCNP, the core was divided in 36 different regions to account for the variation of the burnup with position. The thinner part of the core was divided in 9 regions of equal volume (3 angular and 3 axial intervals). The thicker part of the core was divided in 27 regions, as in the thinner case for θ and z, and additionally it was split in 3 different radial zones (16-18,

18-20, 20-22 cm). The number of days considered for the burnup was 50 and the neutron fluxes were actualized (MCNP calculation) every 2 days. The number of burnup steps performed in ORIGEN2 every 2 days was 40 and 1 predictor step was considered every 2 days. The PWRU library was used in ORIGEN2. Finally, the fractional importance in MONTEBURN 2.0 was chosen to be equal to 1.0E-5. Figure 5 shows the variation of k_{eff} as a function of time. Reference [4] sets the value of marginal reactivity due to temperature effects and facilities in the reflector equal to 7% $\Delta k / k$ for the FRM-II reactor. In our calculation, we have partially considered some temperature effects (MCNP calculation performed at 45 °C and heavy and light water densities evaluated at this temperature). Conservatively, we estimate the life cycle of the final asymmetric model to be 41 days, which gives us a marginal reactivity of 6.8% ± 0.3% $\Delta k / k$ (considering that the error in k_{eff} is twice its standard deviation). However, a longer cycle can easily be achieved by modifying the inner reflector or by using heavy water in the inner-irradiation area. In addition, this reactor and surrounding facilities may be less complex than the FRM-II, requiring a marginal reactivity due to facilities in the reflector lower than that in the German case (several beams). Note that error in k_{eff} (equal to two standard deviations; ±0.005) at day 27 is shown in Fig. 5.

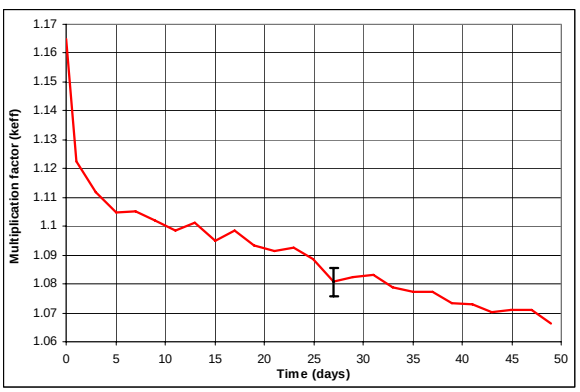

Figure 5. Multiplication factor for the burnup of Case VI (Table III).

DETAILED CORE MODEL

A necessary step to further evaluate features of the design is to model the core using a realistic geometry instead of a homogeneous one. This detailed design can be used to study thermal hydraulic parameters of the core, and carry out more accurate calculations of the neutron flux levels than in the homogenous case. MCNP capabilities allow us to model the core with great detail. The asymmetric core of Case VI from now on is modeled as a group of ten fuel elements (FE), five for each half of the core. Each FE of the thicker part of the core is made of 16 cylindrical fuel plates; the FE of the thinner part consists of 2 fuel plates (figure 6). Table IV shows the dimensions chosen for the fuel plates and the core itself (the same as in the fuel plates of the FRM-II). The fuel material is assumed to be the standard U_3Si_2-Al of 4.8 grU/cm^3 (20% enriched), and the cladding is chosen to be Al-6061. The addition of structural material in the detailed model, plus small changes in the ratio of the amount of light water, cladding and fuel in the core compared to the homogeneous case produce slight changes in the excess reactivity. Initial reactivity of this heterogeneous core is found to be lower than that calculated for the homogeneous one. For this reason, the inner light water channel is reduced again from a radius of 7 to 5 cm and replaced by beryllium (note that the inner radius of the core has not changed, and irradiation positions may be located in the inner reflector or beryllium). With this change, the initial excess reactivity of the heterogeneous model is the same as that in the homogeneous one. Criticality is achieved in the heterogeneous model by positioning a hafnium control rod in the shape of a cylindrical shell of inner and outer radii of 10.4 and 11.0 cm. The shell control rod spans the entire height of the FE.

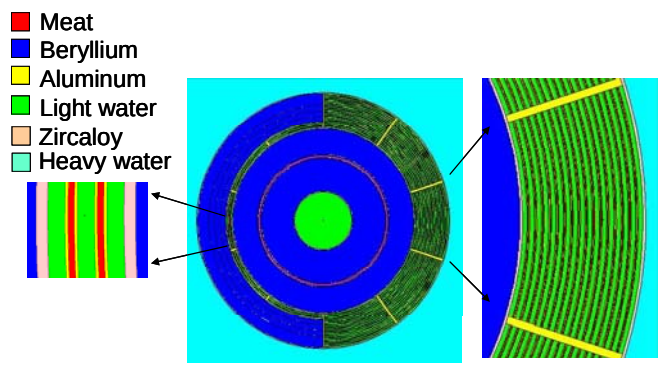

Figure 6. Detailed heterogeneous asymmetric model.

Table IV. Dimensions of the heterogeneous core and the FEs.

Core thick section (mm)	meat thickness	0.65
	Cladding thickness	0.38
	water gap	2.2
Core thin section (mm)	meat thickness	0.8
	Cladding thickness	0.38
	water gap	2.2
Core (cm)	Radii thick part	16-22
	Radii thin part	16-17
	Height	70

High neutron flux levels are desirable features in a modern pool type RR. For the asymmetric detailed model, we calculated thermal neutron fluxes at the $z = 0$ plane to visualize the asymmetry in the fluxes due to the core

configuration. The fluxes were calculated in cubes of 1 cm^3. Figure 7 shows the thermal neutron flux distribution as a color map together with the location of the core within the map.

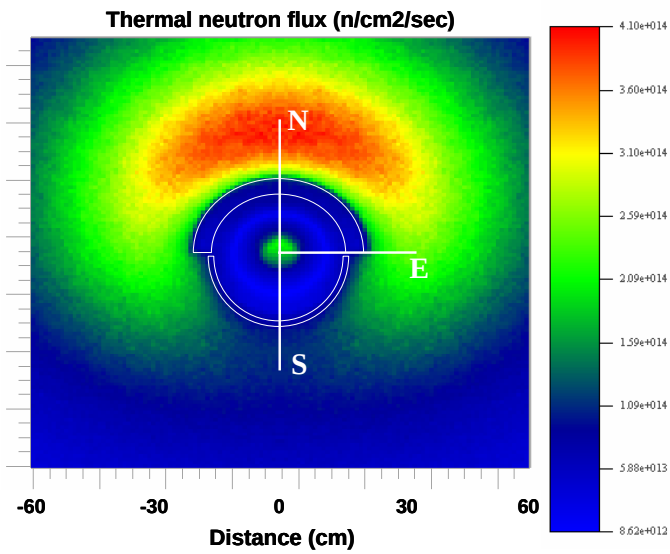

Figure 7. Thermal neutron flux in the z = 0 plane.

For a more quantitative feel, thermal neutron fluxes along three different lines from the center of the core in the *z = 0* plane (north, east and south lines in Fig. 7) are shown in Fig. 8. Also shown is the thermal neutron flux for the simplified model of the FRM-II (azimuthally symmetric). Over the north-line, results show a peak similar to the one calculated for the FRM-II case but located at 35 cm from the core center (4.0×10^{14} $n \cdot cm^{-2} s^{-1}$). A flux peak of 2.5×10^{14} $n \cdot cm^{-2} s^{-1}$ and 1.0×10^{14} $n \cdot cm^{-2} s^{-1}$ occurs along the east and south-lines, respectively.

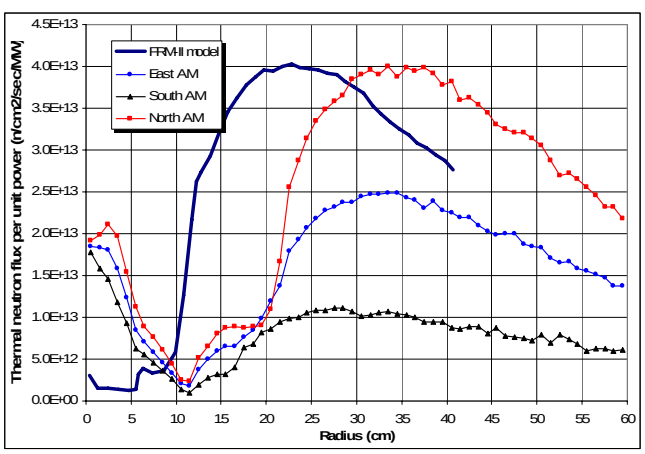

Figure 8. Thermal neutron fluxes for the heterogeneous design and for the simplified FRM-II model.

COOLANT OUTLET TEMPERATURE AND FUEL MAXIMUM TEMPERATURE

The asymmetry of the core suggests that different FEs may require different coolant conditions. To determine the coolant conditions necessary for this design it is necessary to calculate the energy deposited in the core material when the reactor is at full power. This can be readily done with MCNP using a particular tally to account energy deposited by particles and fission products in the material. For this purpose, a KCODE neutron-photon calculation of the asymmetric core was carried out (20000 cycles of 500 histories per cycle). Results show that 87% of the recoverable energy (~ 200 MeV/fission) is deposited in the core material (not including the inner reflector). That is, for the 10 MW core design, 8.7 MW have to be removed by the coolant that flows through the core.

With the help of a simple energy balance, knowing the energy deposited in the core per time unit and assuming steady-state conditions, it is possible to calculate for each coolant channel of each FE, the temperature rise of the coolant across the core. For each channel, we can apply the following energy conservation equation:

$$\dot{m} C_p \Delta T = \dot{Q} \tag{1}$$

where $\dot{m}$ is the coolant mass flow, C_p is the specific heat of the coolant, ΔT is the coolant temperature rise of the coolant across the core, and $\dot{Q}$ is the heat deposited in the core material per unit time. It is assumed that half of the energy deposited in each fuel plate is transferred to each of the two coolant channels that sandwich the fuel plate (Figure 9). The energy deposited in the coolant itself is also calculated however it is two orders of magnitude lower than the one deposited in the fuel plate.

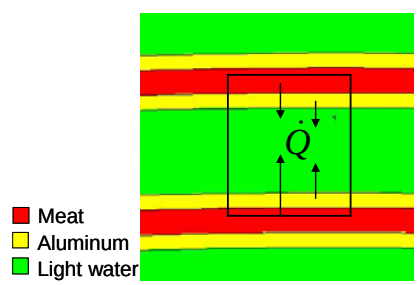

Figure 9. Energy removed for each coolant channel (coolant flowing perpendicular to the paper).

Assuming a coolant velocity of 8 m/s and an inlet temperature of 40°C, figure 10 shows the outlet temperatures of the coolant channels for each FE. Note that inner channels (lower radii) of each FE are numbered

with lower numbers. The FEs of the thinner part of the core has only 3 coolant channels. The thicker part of the other hand consists of 17 coolant channels. Results show that channel #16 in FE# 3 is the hottest one, presenting a temperature rise of 23 C. Additional FEs are not shown because of symmetry.

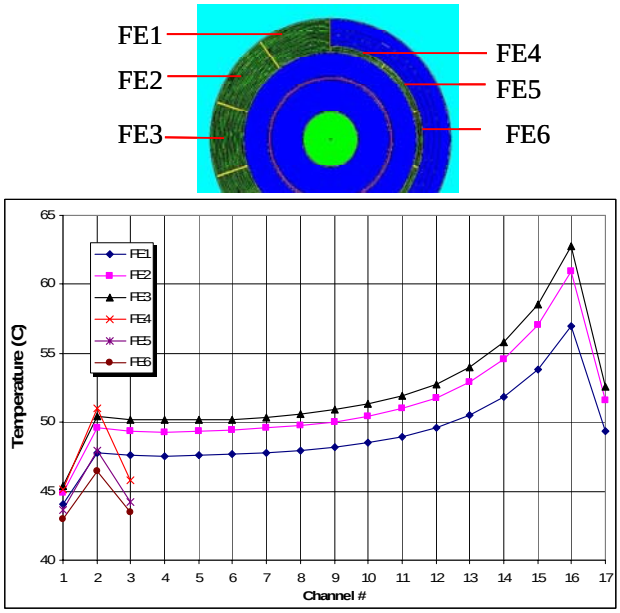

Figure 10. Outlet temperature for each cooling channel.

Figure 11 shows the average outlet temperature of each FE and that of the entire core. The temperature rise of the coolant trough the core is approximately 10°C and 13°C for the hottest FE or FE #3.

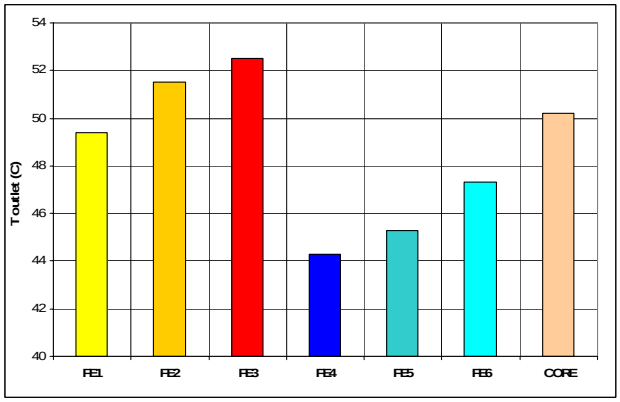

Figure 11. Outlet coolant temperature for each FE and the core.

The coolant flow velocity is an important parameter that directly influences the coolant outlet temperature. In operating research reactors such as the FRM-II, the coolant velocity is about 16 m/s [9]. That means that the value used in the previous calculation can increased to values that have been tested in actual facilities for the same cooling channel area.

For the hottest channel in our model or channel #16 in FE #3 (figure 10), the axial variation of the coolant temperature was calculated, taking the coolant velocity as a parameter. The height of the channel was divided in 14 elements (5 cm each) and the energy deposited in each of these elements was calculated with MCNP. Axial and azimuthal heat conduction in the fuel plate is neglected. Figure 12 shows the axial temperate profile for 3 different coolant velocities. The axial linear power distribution for this channel is also shown in figure 12.

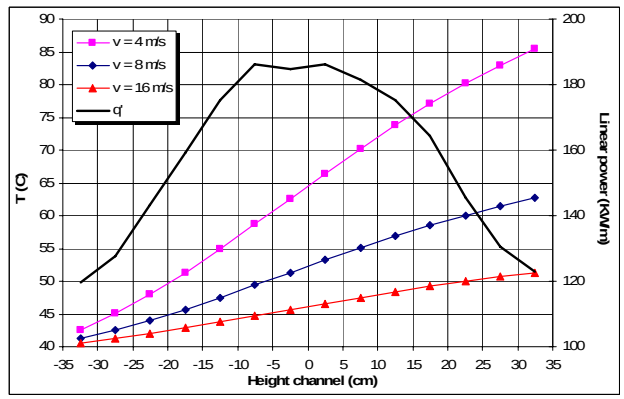

Figure 12. Axial temperature profile and linear power for channel #16 in FE#3.

Clearly, for coolant velocities typical of research reactors, the temperature rise in the core and in the hot channel remain safe. Compared to the FRM-II core, half of the power is being removed with approximately the same flow area and therefore, our thermo hydraulic design is less demanding. Although these are preliminary calculations, a flow velocity of around 8 m/s seems to be an acceptable value that results in safe coolant conditions and reduces pumping power.

Maximum fuel temperature is another important parameter that must be evaluated. To evaluate this parameter we solved the radial conduction equation in 1-D cylindrical geometry for the horizontal section of one fuel plate (figure 13). Axial and azimuthal heat conduction is neglected. Conservatively, the maximum volumetric heat source calculated for the axial heat distribution of the hottest fuel plate was considered (figure 12, center of the channel). Temperature distribution in the clad-meat-clad sandwich is calculated assuming the clad surfaces to be at the coolant temperature, T_c.

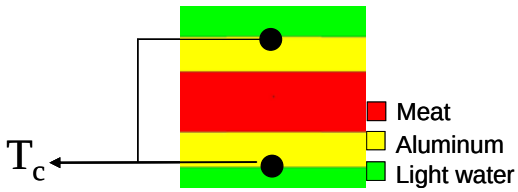

Figure 13. Horizontal section of the fuel plate.

One of the key parameters in the calculation of the fuel temperature is its thermal conductivity. This is a parameter that depends on the fuel fabrication process as well as on the burnup of the FE. Reference [10] gives an experimental curve for the U_3Si_2-Al matrix as a function of the void space or porosity in the fuel. Although for standard fresh fuel material we can consider a thermal conductivity of 180 W/m·K, for other porosities this value can drop to values as lows as 15 W/m·K. Therefore, as the porosity of the FE is not known a priori, the maximum fuel temperature is calculated as a function of the fuel thermal conductivity over the range of interest during the burnup cycle. Figure 14 shows the maximum fuel temperature ($T_{max} - T_c$) as a function of the fuel thermal conductivity for two different meat thicknesses (note that the thinner and thicker part of the core were modeled with different meat thickness, table IV). Even with very low values of the fuel thermal conductivity (~10 W/m·K), the maximum fuel temperature is only 30 C – 45 C above the cladding surface temperature.

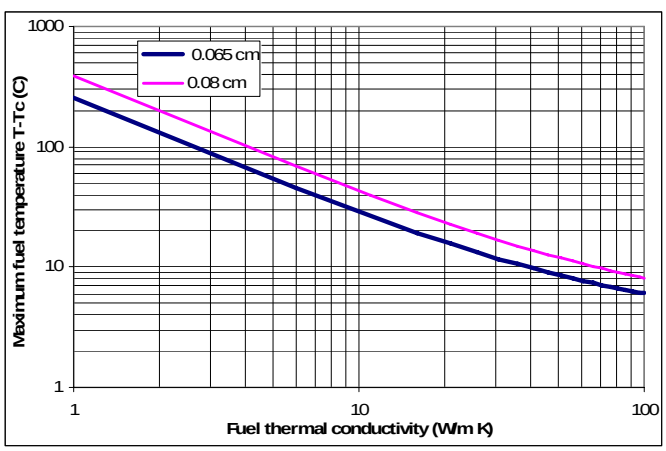

Figure 14. Maximum fuel temperature as a function of the fuel thermal conductivity.

CONCLUDING REMARKS

A detailed design of the core of a modern 10 MW research reactor was developed using MCNP. The asymmetric design, based on a core loaded with standard 20% enriched fuel, allows reaching an unperturbed thermal peak per unit power comparable to that in the FRM-II. The asymmetric design produces a high thermal neutron flux zone (4.0E14 $n \cdot cm^{-2}s^{-1}$), a moderate thermal neutron flux zone (2.5E14 $n \cdot cm^{-2}s^{-1}$), and a low thermal flux zone (1.0E14 $n \cdot cm^{-2}s^{-1}$) in the outer reflector. Moreover, an inner-irradiation area is provided. This region may be useful to irradiate material with harder spectrum than in the outer reflector without perturbing the neutron fluxes there.

Given the dimensions and the power developed by the asymmetric core, for coolant velocities existing in operating research reactors (e.g. 8 m/s), the coolant temperature rise through the core is about 10C. The rise in the coolant temperature in the hottest channel located in the center of the thicker part of the core was calculated to be around 23 C.

Given the design and material of the fuel plates commonly used in research reactors, the maximum fuel temperature does not present any challenge for this design. Calculated maximum temperatures exceed the coolant temperature by no more than 45 C, even when considering the worst case for the fuel thermal conductivity. Although the model employed to calculate this temperature is simple, the maximum fuel temperature is low enough to assure that steady-state thermo hydraulic analysis is unlikely to be the limiting factor for this design. .

Calculations carried out to evaluate thermal hydraulic parameters are preliminary and based on simple models. However, results show that conditions at steady-state are far from those that may compromise the safety of our design. Future work will include system analysis under loss of-pump conditions.

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