THE NEUTRONICS OF INCREASED POWER DENSITY PWR CORE BASED ON ANNULAR FUEL RODS USING THE MONTE CARLO METHOD

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الملخص

تم في هذه الدراسة التحقق وإثبات بارامترات التصميم لأعمدة الوقود المجوفة بواسطة طريقة مونتي كارلو. أثبتت الدراسة أن الأداء النيتروني لهذا النوع من الوقود قريب جدا من أداء أعمدة الوقود الصلب التقليدي حيث إن الفرق بين عامل التضاعف اللانهائي لكل منهما أقل بكثير من 1% والفرق مع الدراسة المنشورة في حدود 1% [1]. تم اقتراح فكرة التصميم الجديدة للأعمدة المجوفة المصففة في تجميعات (assemblies) 13×13 في معهد ماستشوستس التقني (MIT) لغرض رفع قدرة المفاعل إلى 150 % من القدرة الأصلية (3411 ألى 5111 ميجاوات حراري) بدون تجاوز لهوامش الحرارة والأمان. بعد التأكد من بارامترات التصميم لخلية الوقود المجوفة تم إجراء نمذجة مفصلة للتجميعات والقلب. المهمة الصعبة تتلخص في إدارة السم القابل للاحتراق (burnable poison) من الجادولينيوم حيث أنه ليس كل الأعمدة مسممة وليست بنفس النسبة الوزنية كما أن التجميعات مخصبة بنسب متفاوتة عالية ومنخفضة وعليه فقد اعتمدت عملية التكرار للتقليل من عدم الانتظام في توزيع القدرة والحصول على كمية كافية من المفاعلية (reactivity) السالبة لإلغاء المفاعلية الزائدة الضرورية لاستنفاد الوقود (Fuel depletion). البقية الباقية من المفاعلية الزائدة وهي 9% تم إلغاؤها بكمية من البورون السائل في حدود 370 جزء لكل مليون جزء(ppm). وعليه فقد تم الحصول على توزيع الفيض الحراري والسريع والقدرة في القلب وفي كل تجميعة أفقيا بالإضافة إلى توزيع القدرة رأسيا في كل من القلب وأسخن تجميعة وخلية وقود وكذلك في متوسطي التسخين. هذه البيانات أساسية للقيام بالدراسات الهيدرو- حرارية وبالتالي إثبات إمكانية زيادة القدرة المزعومة.

ABSTRACT

In this study the design parameters of the annular fuel cell are investigated and verified using the Monte Carlo method. The annular design shows a comparable neutronic performance to the solid fuel design as the difference in the infinite multiplication factor (k_∞) is far less than 1% and the difference from published work is within 1% [1]. This new design concept of annular fuel pins arranged in a 13 by 13 fuel assemblies has been suggested first at the Massachusetts Institute of Technology (MIT) where they have claimed a 150% power increase (3411 to 5111 MWt) without affecting the thermal and safety margins. Having established the confidence in the fuel cell design parameters, a detailed assembly and core model is described. The difficult part of the procedure is the Gadolinium burnable poison management where not all rods are poisoned and not all have the same mass fraction besides that some assemblies have highly enriched fuel rods and others have low enrichment ones. This lengthy iterative

process is conditioned by less power fluctuations and by having enough amount of negative reactivity to cancel out the excess reactivity required for the fuel to deplete. The remaining 9% of excess reactivity is matched by 370 ppm of soluble boron to make the core critical. Therefore the X-Y fast and thermal flux and power map of the core and assembly in addition to axial power profiles in the core and hottest and average assembly and the hottest and average fuel cell are obtained. Such data are essential for thermal hydraulic analysis and hence provide a proof for the power output increase.

KEYWORDS: Power Density; PWR; Monte Carlo; MCNP Code; Annular Fuel; Brain Tumor; Flux; Criticality; Reactivity Control; Neutronics

INTRODUCTION

The last few years have marked a period of rethinking of nuclear reactor technology because of a changing economic environment reflecting stronger competition with other power sources coupled with a deregulation trend in the electric utility industry. The new and highly competitive environment put all nuclear plants under increased pressure to significantly reduce total power cost. Therefore, reduction of total generation cost while maintaining excellent safety is a key challenge facing the nuclear industry today and in the future [1].

One attractive approach to improve the economy of both the operating and new plants is to increase their power density and extract more energy from a given system volume. One of the key components affecting the allowable power density in the nuclear island is nuclear fuel. In fact, the safety limits in a nuclear power plant are largely related to its fuel. Evolutionary improvements in fuel design and cladding quality allowed a remarkable reduction of failure rate, and fuel assembly design changes allowed both power increases and performance improvement at steady state and during accidents.

Thermal hydraulic studies proposed three types of fuel design, which are:

1. Conventional solid cylindrical fuel rods.
2. Internally and externally cooled annular fuel rods.
3. Spiral cross-geometry fuel rods.

The proposed annular fuel departs from the traditional solid rod design by introducing internal cooling of each fuel pellet. The idea of annular fuel pellets is not new in reactor technology. In pressurized water reactors, annular fuel pellets are used in the short axial blankets of western PWRs and BWRs and in Russian VVER cores. Annular fuel with both internal and external cooling was proposed for high temperature gas cooled reactors where a compact fuel element of annular shape was conceived to be enclosed in an inner tube and an outer tube. The annular fuel proposed here uses UO2 fuel and is intended for all core assemblies with the prime objective being to significantly increase fuel burnup (to 80-90MWd/kg) and core power density (up to 50%), while increasing or at least maintaining safety margins. The new internally and externally cooled Annular Fuel (AF) for a PWR has been proposed to substantially increase power density while retaining or improving safety margins. The geometry of this annular fuel is shown schematically in Figure (1), where the traditional solid fuel rod is also drawn for comparison. The annular fuel rods are of significantly larger diameter than the typical solid rods to accommodate an inner coolant channel allowing for sufficient coolant flow [2].

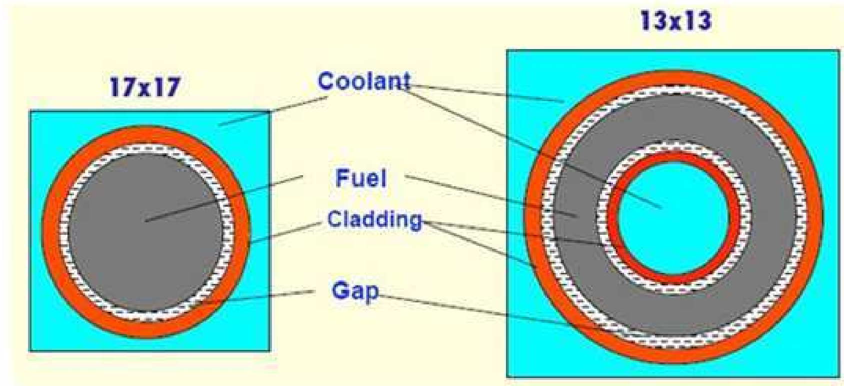

Figure 1: Schematic of solid and internally and externally cooled annular fuel rod and associated coolant cell

A transition from solid to annular geometry has two important implications that allow higher power density:

(1) Reduction of the thickness of the heat conduction path, which improves the margin from peak fuel temperature to melting.

(2) Increased heat transfer surface area (in spite of a reduction of the number of fuel rods), which enlarges the Departure from Nucleate Boiling Ratio (DNBR) margin.

In addition to peak fuel temperature and DNBR limits, the new fuel has to satisfy a number of other safety limits and performance constraints. The internally and externally cooled annular fuel concept exhibits significantly lower fuel temperature than solid fuel, hence it is expected that fission gas release will be smaller allowing higher burnups.

The Nuclear Energy Research Initiative (NERI) at the Massachusetts Institute of Technology (MIT) undertook the neutronic and thermal hydraulic design of the reactor core based on annular fuel rod design. The optimization of rod parameters were carried out using deterministic codes benchmarked by the Monte Carlo (MC) code MCNP-4C.[3] They have used the cell code CASMO-4 and the cross section data tables code TABLES-3 and the global code SIMULATE-3.[4] Of course the modifications carried out to the codes during benchmarking have their uncertainties. The deterministic codes fall short in describing the details of the heterogeneous nature of the reactor core. They usually homogenize the X-section before carrying out global calculations in addition to approximations to physics and numerical models. They have relied on such codes because of speed and the ability to perform detailed fuel depletion calculations, an advantage over MC codes. However, using the MC method one can describe all the fuel cells, assemblies, and the core with their actual details and retrieve cross sections from the point wise tables directly.

Throughout this paper, the following calculations are performed using MCNP-4C aiming at producing radial and axial power distributions for the hottest and average channel in the core at 150% power density increase to be used in future thermal hydraulic calculations. Prior to this, cell calculations are verified by comparing to those obtained by the fore mentioned deterministic codes. The comparison is done through the infinite multiplication factor in step changes from the solid fuel design to the annular design.

UNIT CELL MODEL

A 50% core power density up rate for a Westinghouse 4-loop PWR may be achieved based on the AF concept with comparable or better thermal margins than conventional solid fuel pins. Although the AF concept may be thought of as a neutronically inferior option due to increased U-238 resonance absorption due to decreased spatial self shielding and doubling of cladding parasitic absorption, there are other differences from the solid pins such as lower fuel temperature (hence a Doppler reactivity gain) that may be beneficial. Therefore, a detailed, step-by-step study (with a unit cell model) has been performed to assess the infinite multiplication factor (k_∞) changes of the AF concept using MCNP-4C.

From Solid to Annular: A Step-by-Step Variation Approach

The reference solid fuel pin (Case 0) is taken from a standard Westinghouse PWR 17×17 lattice assembly to serve as the starting point. The end point is the newest AF design with a 13×13 lattice (Case 6). Thus, in order to understand the fundamental neutronic differences between solid and annular fuel, five intermediate hypothetical cases (step-by-step variations) were created as a logical connection between the reference solid pin (Case 0) and the AF fuel (Case 6) as shown in Figure (2) Note that except for the difference specified, all other parameters are kept the same. These variations in geometry and material composition and ratios will alter the physics of the cell such as resonance absorption, moderating power, parasitic absorption, etc.

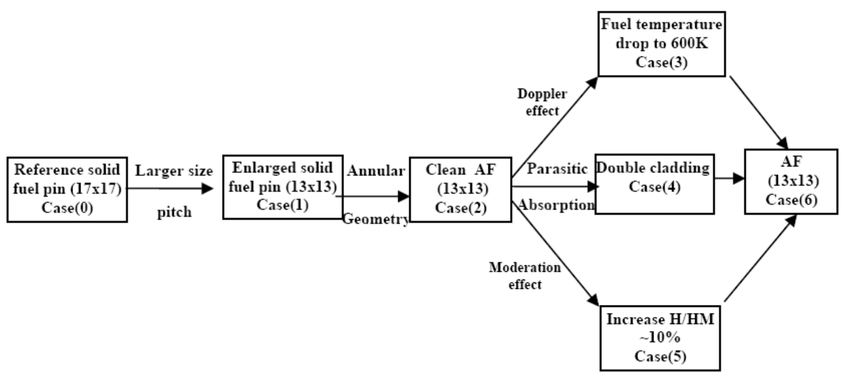

Figure 2: Illustration of step by step transformation from solid to annular design

The definition of the representative fuel cell is not clearly stated in the reference [4]. Therefore, two approaches of cell calculation are proposed in the step-by-step variations from the solid rods in the 17×17 assembly to AF rods in the 13×13 assembly. The first approach is to extract the cell from the assembly as it is (fuel, clad, moderator). The second approach is to make a representative cell for the whole assembly with the following constituents: fuel, clad, moderator, in addition to stainless steel (SS) and water. The last two materials are the share of the cell from the surrounding assembly structure material and water separating assemblies. MC calculations for the cases 0 through 6 are carried out and the results for the infinite multiplication factor are listed in Table (1). Figures (3) and (4) illustrate the steps graphically.

Table 1: k_∞ for step-by-step changes (% difference from reference in brackets)

Case	Case (0)	Case (1)	Case (2)	Case (3)	Case (4)	Case (5)	case (6)
First approach	1.44597 (3.%)	1.2569 (11%)	1.36559 (<1%)	1.37794 (<1%)	1.35334 (<1%)	1.41669 (2.7%)	1.42944 (3.6%)
Second approach	1.398 (<<1%)	1.24016 (12%)	1.32903 (2.3%)	1.34020 (2.5%)	1.31681 (2.2%)	1.37742 (<<1%)	1.38826 (<<1%)
Reference	**1.397**	**1.414**	**1.361**	**1.375**	**1.347**	**1.380**	**1.380**

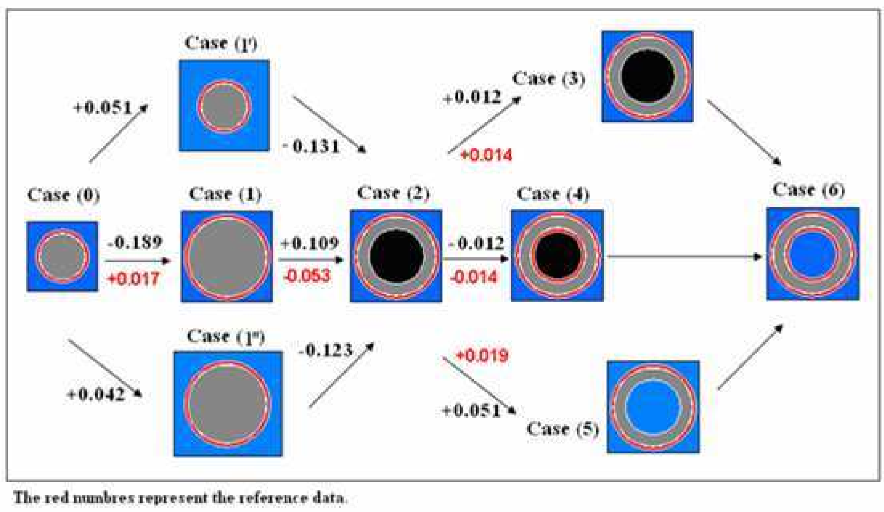

Figure 3: Changes in k_∞ for step by step changes (first approach)

The values of k_∞ obtained for the cases 0, 5, and 6 in the second approach are comparable to the reference case. Cases 2, 3, and 4 (unrealistic fuel rods) for both approaches are not greatly different, however, case 1 for both approaches k_∞ is far from the reference value. This is due to the ambiguity associated with the understanding of the phrase "*enlarged solid fuel pin*". Hence, the following routes have been proposed. The first route (case 1') is to assume a cell with solid fuel type 17×17 having an amount of moderator equal to that for the fuel type 13×13 (an over moderated cell). The second route (case 1") is to assume a solid fuel type 13×13 with the same moderator to fuel ratio of the 17×17 fuel cells. Referring to Table (2) case 1' (second approach), the value of k_∞ is the nearest to the reference one.

Table 2: k_∞ for cases 1' and 1''

	Case (1)	Case (1')	Case (1'')
First approach	1.2569	1.49671 (6%)	1.48835 (5%)
Second approach	1.24016	1.42742 (1%)	1.45517 (3%)
Reference	1.414		

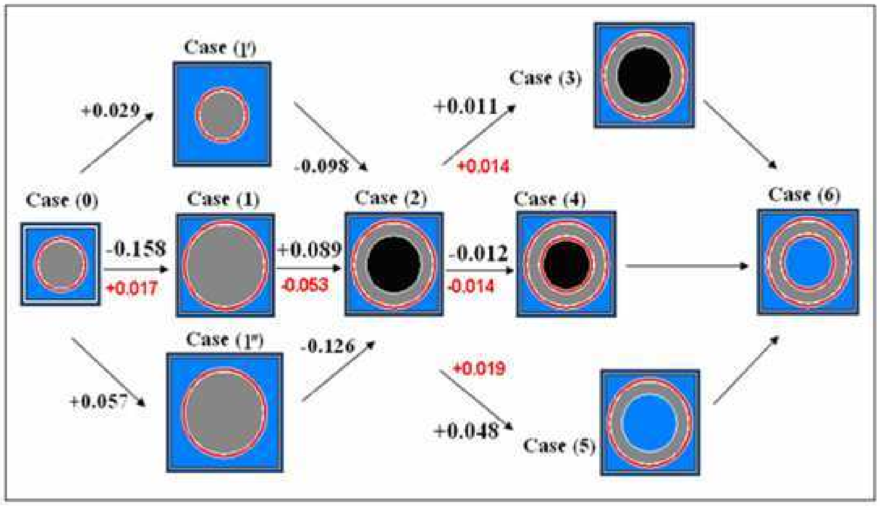

Figure 4: Changes in k_∞ for step by step changes (second approach)

Core description

The standard Westinghouse pressurized water reactor core operating at 3411MWth with 193 (17×17) fuel assemblies with solid fuel rods is replaced with 193 (13×13) fuel assemblies with AF rods operating at 5111MWth (150% increase in the power output). The AF assembly has the same dimensions as the solid fuel assembly. The new design at higher power density introduces changes of the core operating parameters. For example, the nominal fuel temperature is 900 K for the solid fuel, 600 K for the annular fuel at 100% power level, and 800 K for the AF at 150% power level according to preliminary estimates. Therefore, fuel temperature feedback parameters are different for the AF [5].

The reference core is an equilibrium 18-month-cycle, 3-batch PWR core, consisting of 72 fresh fuel assemblies, 72 once-burnt fuel assemblies, and 49 twice-burnt fuel assemblies. The 72 reload assemblies are subdivided into two enrichment levels: 48 with higher and 24 with lower enrichment. The sub-batch with higher enrichment stays in the core for 3 cycles whereas the lower-enriched fuel remains for 2 cycles, but at higher flux regions to achieve comparable burnup. The letters H and L designate high (9.0%) and low (8.1%) enrichment, the first two numbers after a letter stand for number of rods with gadolinium (Gd) and the last two numbers designate the weight percentage of Gadolinia (Gd2O3) in these rods (for example 08 stands for 8.0wt%). The assembly fuel pin layouts for the annular-fueled, 150%-power PWR core are present in Figure (5) [1].

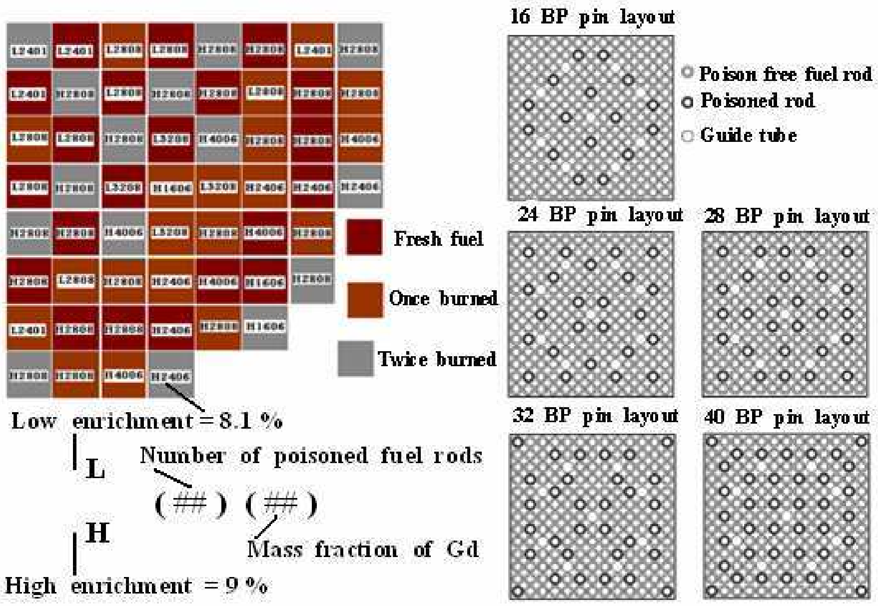

Figure 5: Reference core loading pattern with 13×13 assemblies at 150% power up rate

MCNP CORE MODEL

The core under study, unlike the reference core, is loaded with fresh fuel at the beginning of the fuel life (BOL). The modeled core data departs from the reference one with a slight fuel composition changes in a number of assemblies. The changes comprise Burnable Poison (BP) mass fraction in the fuel material in a trend to obtain acceptable near uniform power distribution not to exceed the maximum core peaking factor of 2.5 such that minimizing power fluctuations between neighboring fuel assemblies. The Gadolinium mass fraction is varied from assembly to another. In some assemblies all rods are poisoned and in others are partly poisoned. Successive iterations of about 12 steps (employing trial and error and engineering judgment) have been executed in order to optimize the BP content in the fuel rods and to assign the high and low enrichments to the chosen assemblies. Multiple runs of the MCNP code have shown that the number of histories per neutron cycle in the KCODE calculation is about 8×10^4. This is a requirement for statistically acceptable results. The remaining excess reactivity ($\Delta k / k$ %) in the presence of BP is about 9 %. There is a complete symmetry in the geometry and material composition of the fuel assemblies axially and in the transverse directions, hence only a quarter core is considered. The annular rods and the fuel assembly details making up the core are included, whereas, in the reference core calculations are carried out using deterministic codes where homogenization of fuel cells, and assemblies is a routine procedure. Figure (6) shows MCNP graphics of the fuel rod, assembly, and core (different colors indicate different fuel composition). The amount of shim control required to bring the core to criticality $k = 1.0026$ at the BOL has been calculated in the range of 370 ppm. The investigation of results includes flux and power distributions in the core, hottest and average assembly, and the hottest and average channel.

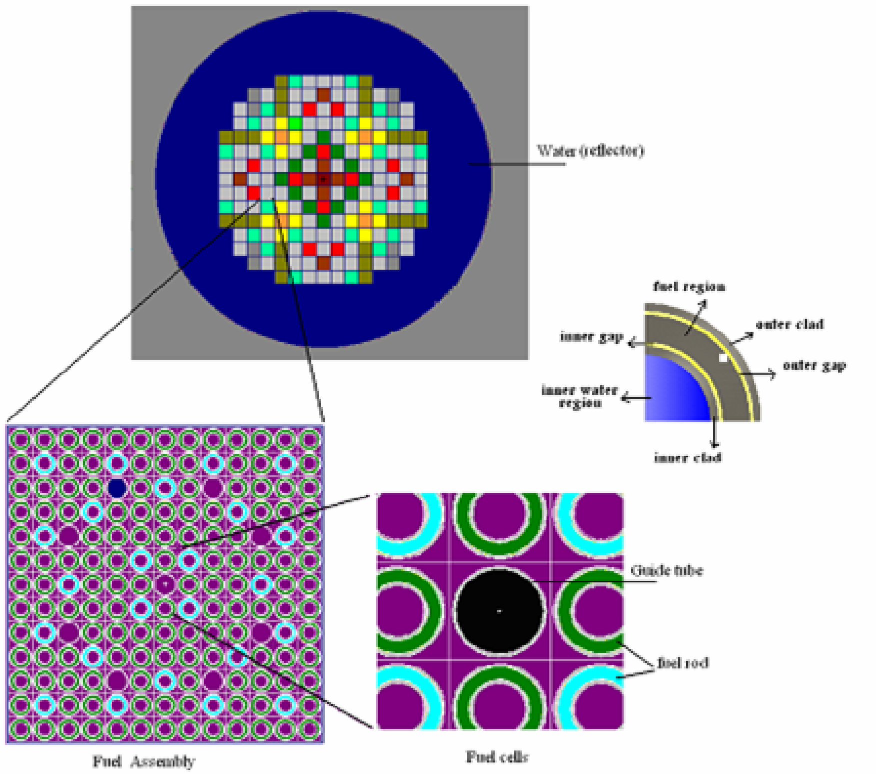

Figure 6: MCNP drawing of the core and assembly

RESULTS AND DISCUSSION

Based on the above modeling, calculations are carried out on three levels: the core, the fuel assembly, and the fuel rod. From core calculations, the hottest and the average fuel assemblies are designated and from which the hottest and average channel are also designated. It should be noted that axial calculations of flux and power in the reactor core, fuel assembly, or fuel rod in Monte Carlo are done through segmenting in the axial direction to create cells for tallying (Figure 7). Unlike in the deterministic numerical methods, there is no need for treating the fuel assembly or rod independently from other assemblies or rods using boundary or reflecting conditions. Instead the hottest fuel assembly or the hottest fuel rod is segmented in the presence of the rest of rods and assemblies during the computer run. Therefore, the neutronic effects of neighboring assemblies or rods are accounted for precisely. This is a great advantage of the MC method.

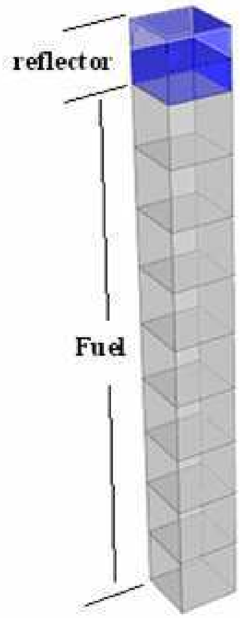

Figure 7: Axial segmentation

Core flux and power profiles

Figure (8) shows the core fast and thermal fluxes in each assembly in the quarter core. Figure (9) shows the core power in each assembly where one can easily designate the hottest assembly generating about 55MWt and the average assembly generating about 26MWt, the nearest to the calculated average. Figures (10) and (11) show the core radial fluxes and power profiles at core center. Figures (12) and (13) show the core axial fluxes and power profiles. The axial variation of flux and power is symmetric because no consideration of axial temperature variation in fuel and coolant. More power is produced in the MIT model. This is may be attributed to a lower amount of BP inserted in the fuel rods allowing for more soluble boron injection. Another reason is the different methods in both calculations. Fluctuations are also apparent in the MIT calculations. The thermal flux peaking in the reflector is also apparent both radially and axially. Notice that the hottest assembly contains the least number of BP rods.

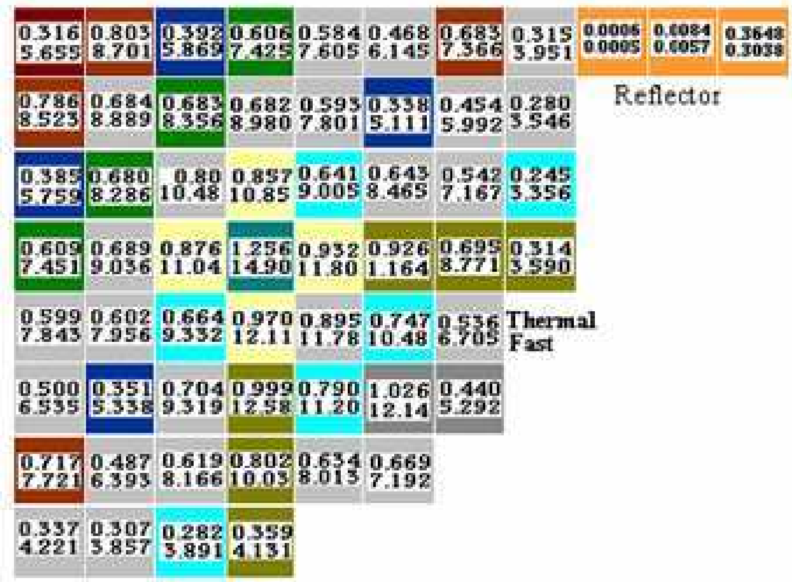

Figure 8: Core fast and thermal flux (x10^{14})

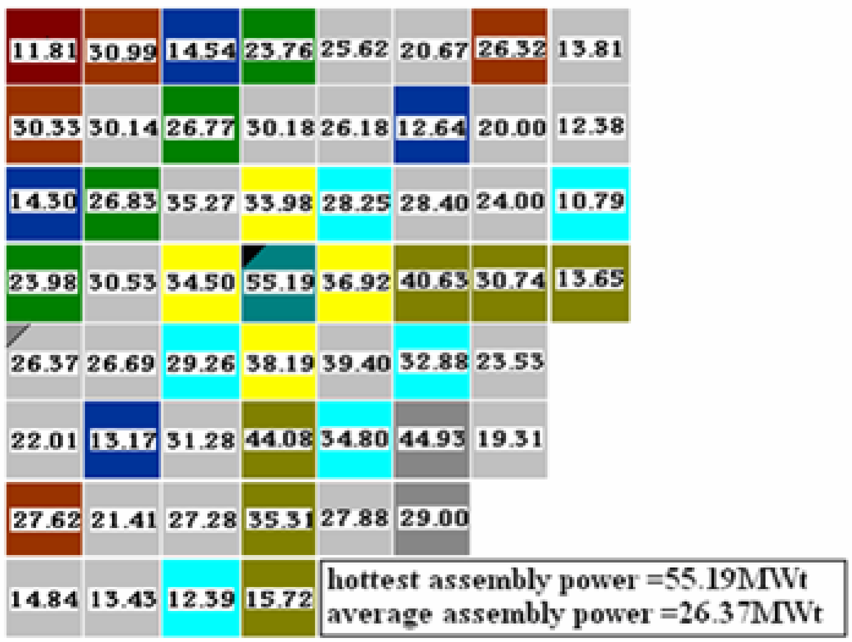

Figure 9: Core power X-Y map (MWt)

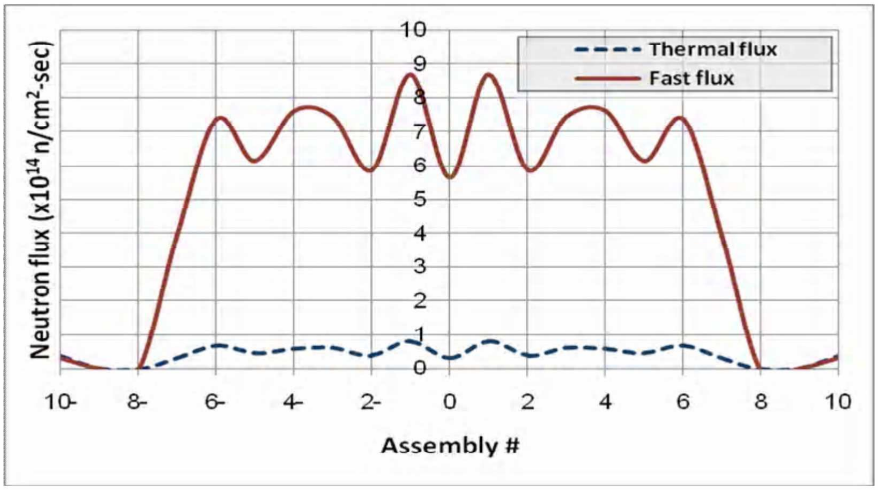

Figure 10: Core radial flux profile

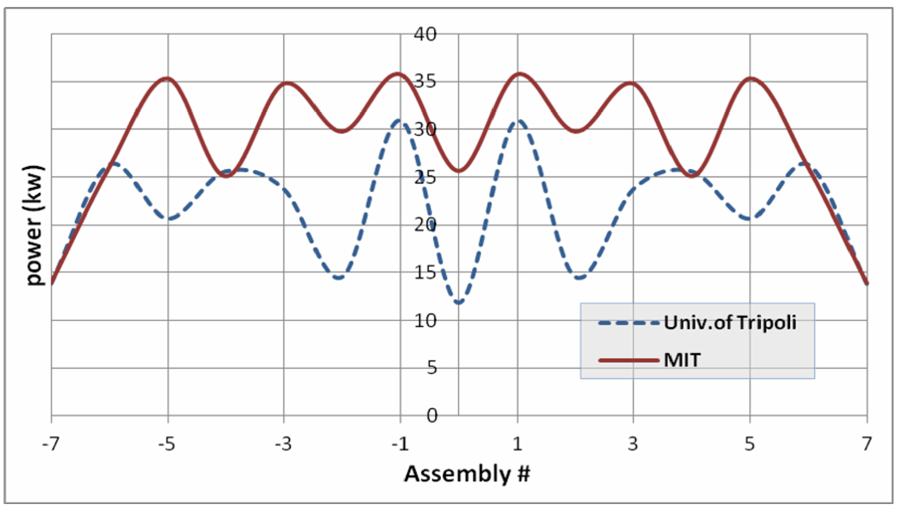

Figure 11: Comparison with MIT calculation of core power profile

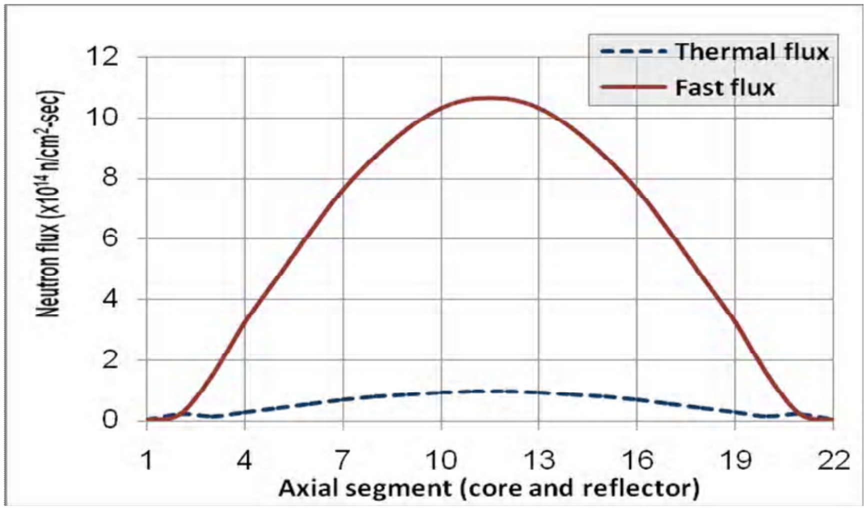

Figure 12: Core axial flux profile

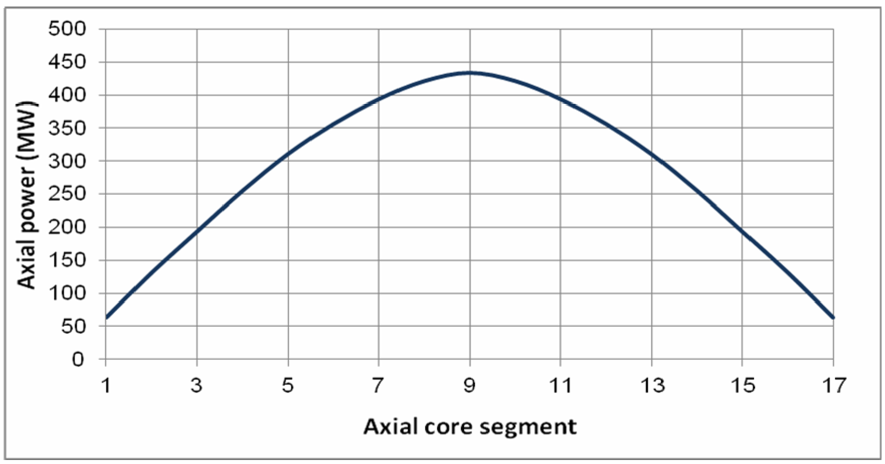

Figure 13: Core axial power profile

Fuel assembly calculation

Once the hottest and average assemblies are designated (figure 9) detailed analysis on the level of the fuel rod is performed. The detailed composition of the assembly is accounted for in the MC model which includes the locations of the poisoned rods, clean rods and guide tubes. Rod powers are presented in Figures (14) and (15) for the hottest and the average assemblies. Figures (16) and (17) show the axial variation of flux and power in both types of assemblies.

315	376	357	363	346	316	330	337	354	383	365	331	306
364	375	391	376	374	135	301	135	361	384	374	352	348
384	392	378	390		363	355	353		350	360	384	362
391	388	363	137	368	381	355	372	346	138	360	378	362
362	388		373	385	383	148	360	364	363		357	357
350	143	383	389	375	401	393	400	343	358	347	135	326
357	325	376	370	144	374		403	141	346	352	292	332
363	149	385	386	359	387	389	376	349	374	362	136	339
391	385		373	384	374	143	358	379	370		365	350
400	414	397	149	369	388	349	366	363	141	361	377	362
404	416	415	399		381	371	370		364	375	390	370
376	408	403	387	367	145	314	141	340	370	375	375	346
345	408	403	413	390	336	354	336	361	382	381	357	308

Figure 14: The hottest assembly power X-Y map (shaded areas are the BP positions)

204	194	207	205	205	213	192	189	191	208	225	194	230
218	80	194	197	77	183	079	178	080	204	209	081	206
224	202	199	192		206	180	197		209	199	215	216
229	200	190	78	199	203	77	186	185	77	206	195	208
204	81		196	195	189	184	193	208	199		77	181
199	175	206	174	174	77	174	73	179	180	181	157	186
175	73	159	69	164	156		168	168	71	163	73	182
177	151	178	177	169	71	163	71	175	182	185	161	177
175	75		165	172	166	164	174	176	171		72	177
185	170	179	67	166	165	70	167	184	67	176	173	194
199	186	177	164		173	165	188		171	167	168	179
168	70	169	169	68	157	68	162	71	184	172	71	168
179	169	190	184	166	159	170	186	177	184	178	166	164

Figure 15: The average assembly power (kW) X-Y map

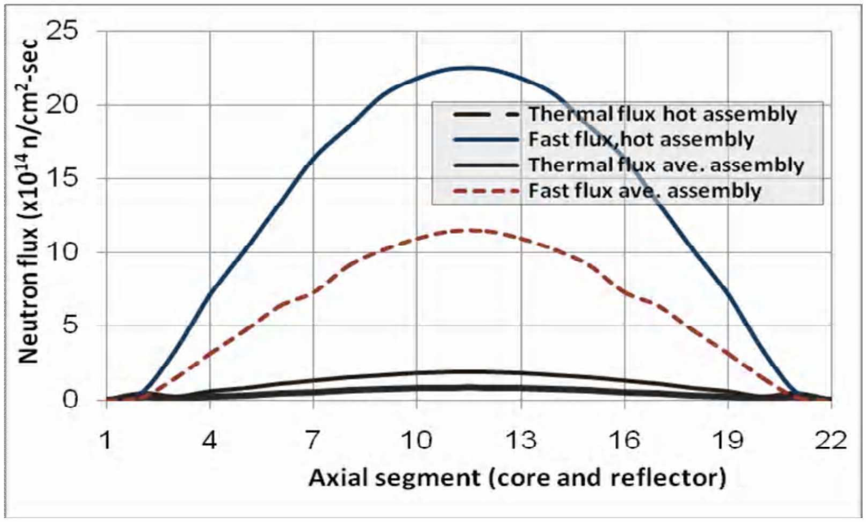

Figure 16: Hottest and average assembly average flux

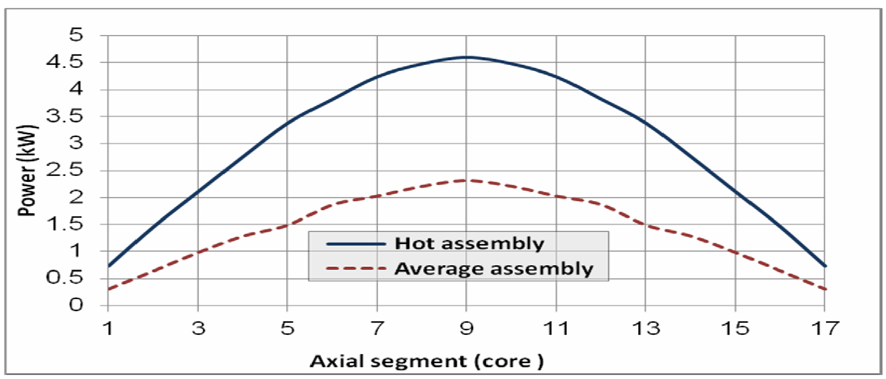

Figure 17: Hottest and average assembly axial flux

Fuel rod calculation

Referring to Figures (14) (the hottest assembly) and (15) (the average assembly) the power produced in the hottest channel is 416 kWt and the power produced in the average channel is 230 kWt. Axial analysis of the channels results in the flux and power distributions shown in Figures (18) and (19). The non smoothness of the profiles are attributed to statistical fluctuations due the small size of the axial segments of the fuel cell unlike the large size of the segments in the assembly and core.

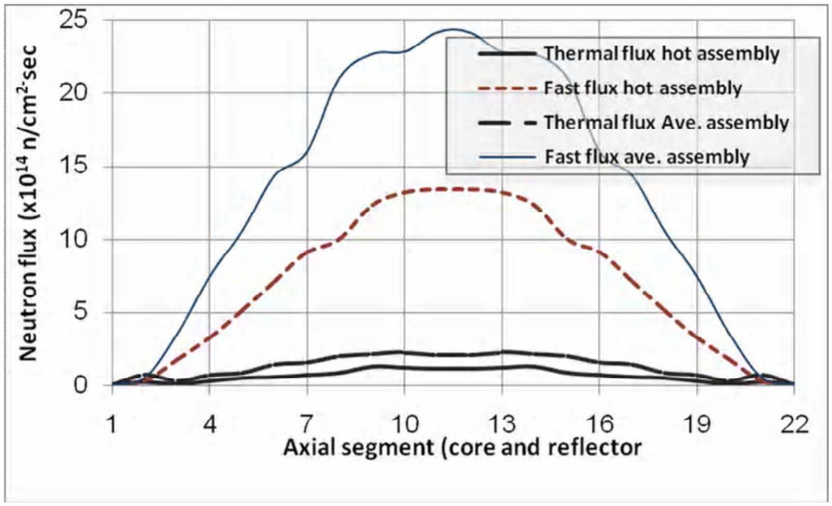

Figure 18: Hottest channel axial flux profile

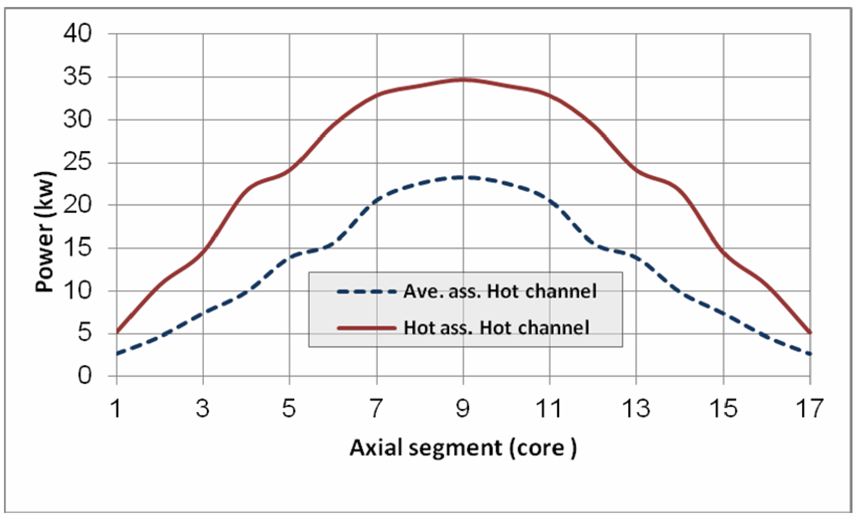

Figure 19: Hottest channel axial power profile

CONCLUSION

The subject of this study is a reactor core based on 13 by 13 annular fuel assemblies. This new design concept has been suggested first at the MIT where they have claimed a 150% power increase of the reactor output. The main task in this study is to utilize the capabilities and facilities of the Monte Carlo code MCNP for the neutronic analysis of annular fuel cells. The objective is to obtain the power profiles necessary for thermal hydraulic analysis.

In the first stage of this study the design parameters of the fuel cell are investigated and verified using MCNP. From the step-by-step approach, the annular design shows a comparable neutronic performance to the solid design. A confident basis for the next stage is established upon satisfactory comparison of the results to the published work. Detailed core and assembly description along with complete poison management made it possible to produce an X-Y flux and power map of the core and assembly in addition to axial power profiles in the core and hottest and average assembly and the hottest and average fuel cell. Such data is essential for thermal hydraulic analysis.

The results of this work should still be refined. Mainly, more work is needed in the area of poison management by increasing the amount of chemical shim and reducing the weight percent of burnable poison to cancel out the excess reactivity in the core. The effect of axial temperature variation on the axial power peaking should be included by performing successive iterations between MCNP and thermal hydraulic codes.

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