COMPARISON OF SQUARE AND HEXAGONAL FUEL LATTICES FOR HIGH CONVERSION PWRs

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ABSTRACT

This paper reports on an investigation into fuel design choices of a PWR operating in a self-sustainable Th-^{233}U fuel cycle. Achieving such self-sustainable with respect to fissile material fuel cycle would practically eliminate concerns over nuclear fuel supply hundreds of years into the future. Moreover, utilization of Light Water Reactor technology and its associated vast experience would allow faster deployment of such fuel cycle without immediate need for development of fast reactor technology, which tends to be more complex and costly. In order to evaluate feasibility of this concept, two types of fuel assembly lattices were considered: square and hexagonal. The hexagonal lattice may offer some advantages over the square one. For example, the fertile blanket fuel can be packed more tightly reducing the blanket volume fraction in the core and potentially allowing to achieve higher core average power density. Furthermore, hexagonal lattice may allow more uniform leakage of neutrons from fissile to fertile regions and therefore more uniform neutron captures in Thorium blanket. The calculations were carried out with Monte-Carlo (MC) based BGCore system, which includes neutronic, fuel depletion and thermo-hydraulic modules. The results were compared to those obtained from Serpent MC code and deterministic fuel assembly transport code BOXER.

One of the major design challenges associated with the SB concept is high power peaking due to the high concentration of fissile material in the seed region. In order to explore feasibility of the studied designs, the calculations were extended to include 3D fuel assembly analysis with Thermal-Hydraulic (TH) feedback. The coupled neutronic - TH calculations were performed with BGCore code system. The analysis showed that both hexagonal and square seed-blanket (SB) fuel assembly designs have a potential of achieving net breeding. While no major neutronic advantages were observed for either fuel lattice arrangement.

1. INTRODUCTION

Possibility of achieving self-sustainable Th-^{233}U fuel cycle was investigated in previous reports [1][2] and proven to be possible in principle. These studies indicate that replacing the UO_2 fuel by homogenously mixed ThO_2-$^{233}UO_2$ fuel does not provide sufficient rate of fertile to fissile conversion necessary for self-sustainable reactor operation. In order to achieve high conversion ratio, a careful design optimization should be performed employing heterogeneous fuel geometry and advanced materials. Therefore, a PWR assembly with uniform distribution of fissile nuclides was modified to heterogeneous two fuel zone configuration: "seed" region containing most of the fissile ^{233}U and a fertile ^{232}Th "blanket" region. The main idea behind the "seed-blanket" (SB) configuration is to maximize the neutron leakage from the seed zone to the blanket zone and thus achieving maximum neutron capture rate in Th by avoiding competition for neutron absorption with ^{233}U.

Previous studies of the high conversion PWRs were conducted for a square fuel pin lattice using fuel assembly transport and burnup code - BOXER [3].

The reported optimization of heterogeneous seed-blanket (SB) fuel assembly was performed by adjustment of seed and blanket region dimensions and geometry of the fuel unit cell in each region. In addition, heterogeneous duplex pellet for the seed fuel pin was used to improve conversion ratio.

The calculations reported in references [1] and [2] were performed with BOXER code for 2-dimensional infinite fuel assembly lattice. BOXER code was developed for the analysis of conventional Light Water Reactors, in particular, for generation of few group homogenized cross sections for full core nodal diffusion theory simulators. The code uses cross section library based on JEF-1[5] evaluated data, which is somewhat outdated, although multiple recent studies indicate that it produces fairly accurate results for conventional LWR fuel designs. Other studies however, report relatively high sensitivity of the results to the cross section library used [7]. In addition, for fuel assembly configurations with more than one unit cell type, spectrum calculations in BOXER code require declaration of the "principal cell type" for which the calculations are performed with zero net current boundary conditions. The outgoing current spectrum is then used as a boundary condition for all other cell types in the assembly. In the case of highly heterogeneous high conversion ratio LWRs however, the power may shift from one assembly zone to another during fuel depletion indicating that the net current of neutrons may reverse its direction, which, in turn, means that the assumption of having a single "principal" cell type is no longer valid. In other words, the "principal" fuel cell may need to be different at BOL and EOL.

Due to the listed above assumptions, which are built into the BOXER code computational model, for the highly heterogeneous assembly designs, the results of the studies reported in [1] and [2] require additional verification. Therefore, the first objective of this work is to compare and verify the previously obtained results by repeating the analysis with Monte Carlo (MC) based computer codes which are free from many approximations used in the solution of neutron transport problems by the BOXER code. Two MC codes were used for this purpose – BGCore [8] and Serpent code [10] in combination with the most recent cross section library derived from JEFF-3.1 evaluated data file.

BGCore [8]reactor analysis system was developed for calculation of in-core fuel composition and spent fuel emissions following discharge. It is based on MCNP code to perform continuous energy neutron transport calculations coupled with detailed fuel depletion module. BGCore also includes integrated THERMO [9] module, which calculates coolant flow and temperature distribution in the core components to provide thermal hydraulic feedback to the neutron transport calculations.

Only the most promising case was analyzed, including simplified thermal hydraulic analysis, which, based on the previous studies, suggested that sustainable cycle is achievable but will require a reduction of the core average power density from about 104 to 70 W/cm^3, in order to meet thermal limits of a typical PWR.

The second objective of this work was to examine potential advantages of triangular fuel lattice with respect to achieving net breeding of ^{233}U and compare the main performance parameters with the square lattice case. In a hexagonal lattice, the neutron leakage from fissile seed to fertile blanket region is more uniform resulting in more uniform distribution of neutron captures in Thorium as compared with the square lattice. Better utilization of Th would allow to reduce the amount (and volume) of fertile material needed to achieve break-even breeding ultimately leading to potentially higher achievable core average power density. In addition, hexagonal lattice allows for higher fuel volume fraction in the blanket region, again, potentially allowing to reduce the volume of the fertile zone with low power and, therefore, to increase the average power density.

BOXER code modeling capabilities are restricted to square fuel lattices only. Therefore, in this study, the analysis of hexagonal fuel assemblies was performed only with BGCore and SERPENT codes.

The major design challenge associated with SB concept is the high power peaking in the seed zone, which contains high concentration of fissile material. In order to determine feasibility of thermal hydraulic design, the calculations were extended to include 3D fuel assembly model with Thermal-Hydraulic (TH) feedback. The TH analysis was performed with BGCore code to assure that all major thermal margin requirements are met. The limiting parameters that were evaluated include: axial location of onset of nucleate boiling, peak fuel central line temperature, and minimum departure from nucleate boiling ratio (MDNBR). These limiting parameters are obtained by assuming the same pumping power as in a typical PWR core.

2. METHODOLOGY OF NEUTRONIC AND THERMAL HYDRAULIC ANALYSIS

This section presents a number of topics each devoted to a specific methodology aspect related to modeling of high conversion LWRs. First, a brief description of the codes used in this work is presented, including data libraries and main modeling methodology assumptions.

Then, we briefly introduce neutronic analysis performed in previous studies in order to optimize the SB concept and verify its main conclusions.

The last part of this section presents results of coupled neutronic-TH analysis, which was carried out with BGCore system. This part describes the neutronic-TH coupling methodology for evaluation of basic thermal hydraulic parameters, such as: coolant temperature and density, clad temperature and central line fuel temperature profiles. In this part of the analysis, the coolant

mass flow rate was adjusted such that the pumping power requirements are equal to those of a typical PWR. Finally, we show via coupled neutronic – TH analysis that maximum achievable power density was estimated correctly in the previous studies.

2.1. Codes Description

BOXER

BOXER code is a part of ELCOS [4] system, which is designed to perform a complete LWR core analysis. BOXER performs unit cell and two-dimensional transport and depletion calculations. The results of BOXER calculations are: k-inf, neutron flux, power distribution and reaction rates. The goal of this code is to generate few group homogenized parameters for use by 3D nodal code. The cross section library is based on JEF-1 nuclear data.

SERPENT

SERPENT is a continuous energy Monte Carlo neutron transport code with burnup capabilities developed at VTT research center in Finland [10]. This code allows modeling of arbitrary three-dimensional geometries. The code was developed as an alternative to deterministic lattice physics codes for generation of homogenized multi-group constants for reactor analyses using nodal codes. The code has a number of features that dramatically reduce CPU time required for its execution, among them unified energy grid for storing cross section data and the use of Woodcock delta-tracking of particles.

The current version of SERPENT contains libraries based on JEF-2.2, JEFF-3.1, ENDF/B-VI.8 and ENDF/B-VII evaluated data files.

BGCore

BGCore is a system of codes recently developed at Ben-Gurion University, in which Monte-Carlo code MCNP [11] is coupled with fuel burnup and depletion module SARAF. Data library for SARAF is based on JEFF-3.1 files [12]. BGCore utilizes multi-group (MG) methodology for calculation of one group transmutation cross-sections ([13][14]) which significantly improves the efficiency of burnup calculations. In addition to the depletion module, BGCore system includes a TH feedbacks module - THERMO. The modules are executed iteratively so that the coupled system is capable of predicting fuel composition, power, coolant density and temperature distributions in various types of reactor cores [9].

2.2. Neutronic analysis

This section describes 3 different assembly designs considered in this study. Three main design parameters have been varied, namely: lattice type (triangular or square), seed enrichment and blanket fuel pin radius. The list of considered cases is summarized in 'HEX-2' *Case: 1/6-th of hexagonal assembly layout.*

Table 1.

The parameters were varied in order to observe their effect on reactivity versus Fissile Inventory Ratio (FIR) trade-off for each of the assembly types. In general, lowering fuel enrichment results in lower k-inf and thus generally shorter cycle length. At the same time however, it also results in higher conversion ratio and therefore longer period during which FIR is higher than unity.

The first case in 'HEX-2' *Case: 1/6-th of hexagonal assembly layout.*

Table 1 (REF) is identical to previously reported best performing design for a square lattice. It is designated as the reference case. FIR in this reference case was optimized using a number of design strategies.

- Neutron captures were maximized by employing heterogeneous Seed-Blanket fuel assembly geometry.
- Seed and Blanket zone dimensions were chosen to be on the order of neutron migration length in each region, so that the leakage of neutrons from seed to blanket zone is maximized.
- Large blanket pin dimensions were chosen to lower the H/HM ratio and increase resonance absorption in thorium,
- Th in hydride (ThH_2) rather than in oxide (ThO_2) form was used in order to create more homogeneous fuel and moderator mixture and further increase resonance captures in Th. This allows to reduce the total volume of the blanket zone without reducing FIR, which may therefore also allow to increase the core average power density.
- Heterogeneous ("duplex") pellet was used as a seed fuel. Such duplex pellet consists of two concentric radial zones: inner zone, containing most of the fissile material and thin outer ring of Th. Such an arrangement helps to improve conversion ratio even more.

Square fuel assembly configuration with the most favorable performance was found to be 11x11 fuel pins with standard PWR unit cell size. This fuel assembly design is shown in Figure 1. At this stage of the analysis, only the basic physics of the design was explored. Therefore, the design does not include control rod guide tubes. More practical arrangements of pins and guide tubes within the fuel assembly will be addressed in future analyses.

As stated earlier, hexagonal lattice allows closer packing of fuel pins and correspondingly higher fuel volume fraction. This is a very favorable feature for the blanket zone since heat removal is not an issue, while it may allow reducing the total volume of the blanket and increasing the core average power density.

In this work, we used a hexagonal lattice and followed the same design strategies to maximize FIR, while observing a number of constraints.

- Fuel in-core residence time in 3-batch fuel management should be at least 1000 Effective Full Power Days (EFPD).
- At fuel discharge, FIR should be close to or above 1.0.
- Assembly average power density in all cases was kept constant and equal to 70 W/cm^3 as suggested by earlier results.

Two hexagonal lattice cases were considered ('HEX-2' *Case: 1/6-th of hexagonal assembly layout.*

Table 1). In both cases, the blanket fuel pin radius was increased (relative to the reference case), while the relative volume of the blanket region was decreased from about 74% in the reference case to about 66% in HEX-2 case. Moderator to fuel volume ratio in the seed unit cell (V_m/V_f) in hexagonal lattice cases was kept identical to that of seed unit cell of the square lattice. Percent of U^{233} in the blanket was kept constant, while fissile enrichment of the seed was adjusted to satisfy the above constraints.

The main parameters for comparison of different fuel assembly design options were k-inf and Fissile Inventory Ratio. The FIR ultimately defines the feasibility of self-sustainable fuel cycle. It is formally defined in Equation I. The equation includes the mass of ^{233}Pa in the summation of fissile inventory at any time t. This is because, in the case of a core shut down, all ^{233}Pa will relatively quickly decay into fissile ^{233}U ($T_{1/2} \sim 27$d).

$$FIR \equiv \frac{Fissile\ Mass\ Inventory\ at\ time\ t}{Initial\ Fissile\ Mass\ Inventory} = \frac{M_{^{233}U}(t) + M_{^{233}Pa \rightarrow ^{233}U}(t) + M_{^{235}U}(t)}{M_{^{233}U}(t=0)} \qquad (I)$$

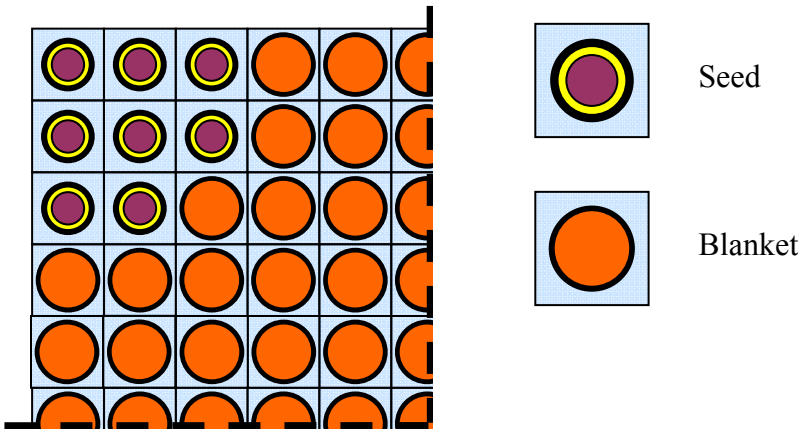

Figure 1: 'REF' Case - Quarter of square assembly layout.

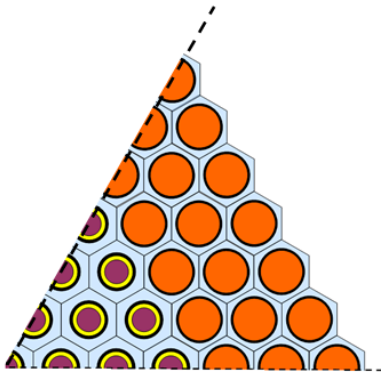

Figure 2: 'HEX-1' Case: 1/6-th of hexagonal assembly layout.

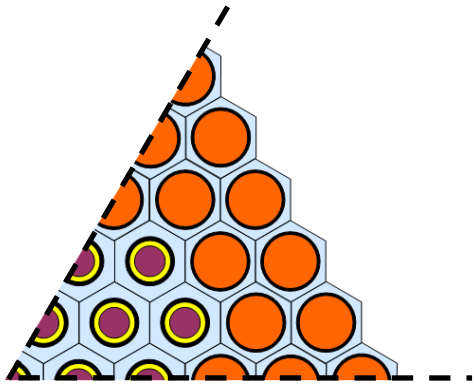

Figure 3: *'HEX-2' Case: 1/6-th of hexagonal assembly layout.*

Table 1: List of cases considered in the neutronic analysis

Case	Lattice Type	Pin radius, cm	Vol. fraction	Enrichment ($^{w}/_{0}$)		Shown in
		Seed / Blanket	*Seed / Blanket*	*Seed*	*Blanket*	
REF	Square	0.4095 / 0.53	0.26 / 0.74	21.00	0.21	Figure 1
HEX-1	Hex	0.4095 / 0.56	0.29 / 0.71	20.76		Figure 2
HEX-2	Hex	0.4095/ 0.55	0.34 / 0.66	15.00		Figure 3

2.3. Thermal Hydraulic analysis with the BGCore system

This section presents modelling approach used to evaluate feasibility of the SB concept from thermal-hydraulic point of view. High concentration of fissile material in the seed region causes a large power peak. Preliminary thermal-hydraulic (TH) analysis was performed [2] for the most promising SB fuel assembly design in order to estimate the maximum core power density, at which typical PWR thermal margins would be preserved. The previously selected most promising case [1, 2] according to BOXER calculations was 11x11 fuel pins arranged in a square lattice. Fuel types and dimensions of the basic parameters are summarized in Table 2. The enrichments in the inner seed and blanket zones were chosen to be 24.3 ($^{w}/_{0}$) and 0.21 ($^{w}/_{0}$) respectively.

The previously reported results included simplified thermal hydraulic calculations, which suggested that typical PWR power density, 104 W/cm^3, should be reduced to about 70 W/cm^3. This power density was obtained based on the following list of constraints:

- Onset of nucleate boiling in the hot channel should occur at the same height as in a typical PWR core.
- The peak fuel Central Line temperature (T_{CL}) was set to be below 2000 ºC (below melting point for oxide fuel).
- Minimum Departure from Nucleate Boiling Ratio (MDNBR) in the hot channel should be comparable to that of conventional PWR.

Preliminary calculations however relied on radial power distribution obtained from the BOXER code with an assumed cosine shape axial power profile. In this work, two sub-channel SB thermal hydraulic analysis was performed as an iterative feedback to the neutronic calculations in BGCore system.

The objective of this analysis was to confirm the findings of previous studies with regards to the maximum achievable power density in a self-sustainable Th-^{233}U PWR core.

Thermal hydraulic performance of the SB assembly was compared with that of a conventional PWR assembly by assuming the same pumping power requirements as in a standard PWR core. The pumping power is roughly proportional to the product of the pressure drop across the core (Δp) and coolant mass flow rate ($\dot{m}$). This is the most logical approach since it would not require replacement of the core primary coolant pumps. The value of the mass flow rate for each considered power density value was calculated iteratively until the convergence on the pumping power value was achieved. Initially, the assembly is sub-critical but its reactivity increases rapidly with burn-up. It is reasonable to assume that sub-critical fuel assemblies would have power peaking lower than one, while super-critical assemblies would have power peaking higher than one. In this study, we performed our analysis for a state at which SB fuel assembly has k-inf value of 1.0 and operates at core average power density (i.e. power peak of 1.0). Conventional PWR fuel assembly however typically has the highest reactivity value at BOL and, as a result, will likely to operate at the highest power also at BOL. Therefore, a typical radial power peaking factor of 1.4 was applied for the analysis of the reference PWR case. Logical flow of the coupled neutronic-TH analysis is described in Figure 4.

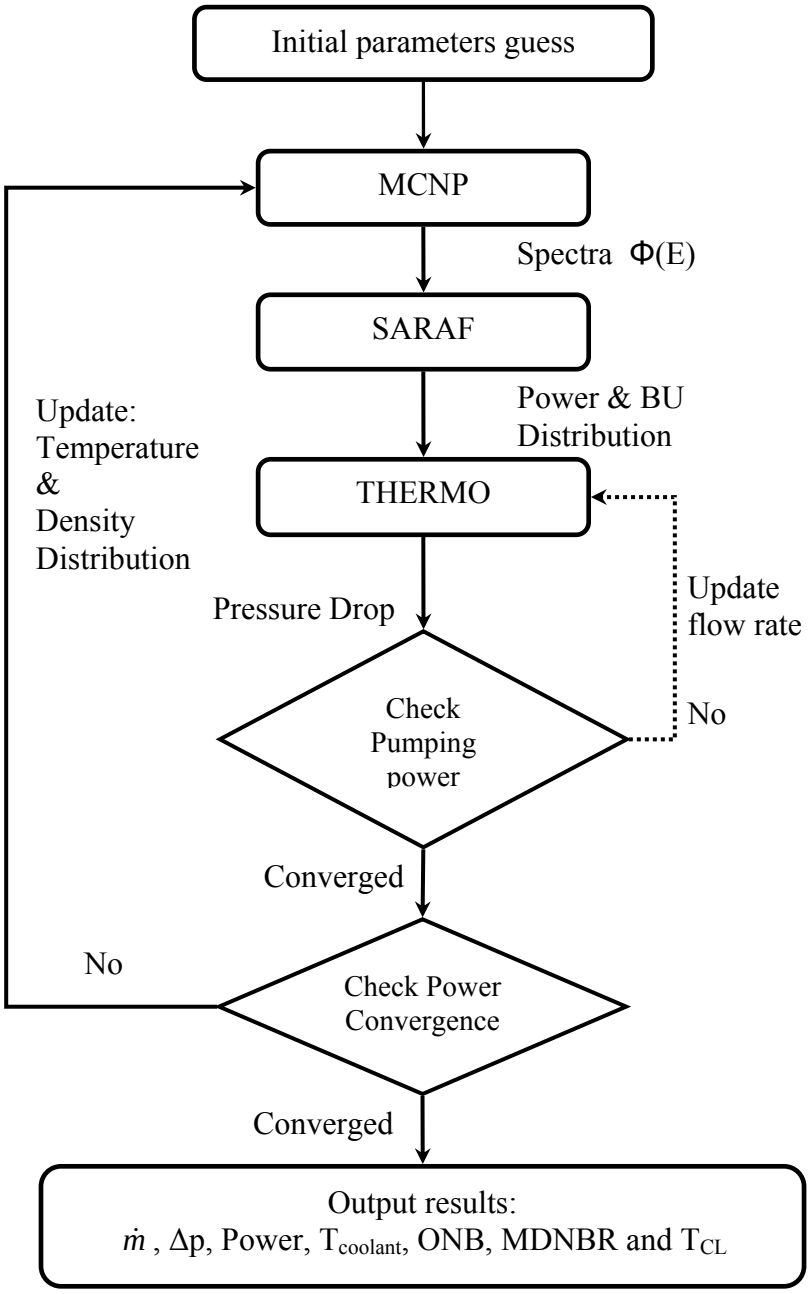

Figure 4: Iterative pumping power scheme for limiting parameter calculation

3. RESULTS OF THE NEUTRONIC ANALYSIS

3.1. Verification of BOXER Results

The main design parameters of the REF case are summarized in Table 2. As mentioned in the introduction section, all preliminary calculations were performed with the BOXER code. This section presents the comparison of BOXER results with MC based SERPENT and BGCore codes.

The second part of this section presents the results of two modified hexagonal lattices and compares them to the square lattice results. These calculations were performed with SERPENT code.

Table 2: Design parameters for the square SB assembly [1,2].

Parameter	**Value**
Pin Pitch [cm]	1.26
Cladding material	Zircaloy
Blanket fuel pellet radius [cm]	0.5500
Blanket cladding outer radius [cm]	0.6155
Blanket fuel material	ThH_2-$^{233}UH_3$
Inner seed fuel pellet radius [cm]	0.3595
Outer seed fuel pellet radius [cm]	0.4095
Seed cladding outer radius [cm]	0.4750
Inner seed fuel material	ThO_2-$^{233}UO_2$
Outer seed fuel material	ThO_2
Fuel rod length [m]	3.66
Inlet pressure [bar]	155.5
Average power density [W/cm^3]	70

The analysis of each case was performed for a range of seed enrichments and blanket pin sizes, however the results of only one case per design with the most favorable performance are presented in Table 3.

Figure 5 and Figure 6 present the results of k-inf and FIR as a function of irradiation time calculated with different codes. BOXER results for k-inf differ from those obtained by the BGCore and Serpent codes by 3%Δk at BOL and 0.2% Δk at EOL. Serpent and BGCore results are in very good agreement. The FIR results predicted by BOXER (Figure 6) are higher compared to those of BGCore and Serpent. This difference results in about 100 days longer period during which FIR remains above unity. Under-prediction of k-inf by BOXER code is consistent with over-prediction of FIR and results from higher Th capture rate. We attribute most of the discrepancy to different cross section libraries used by the codes.

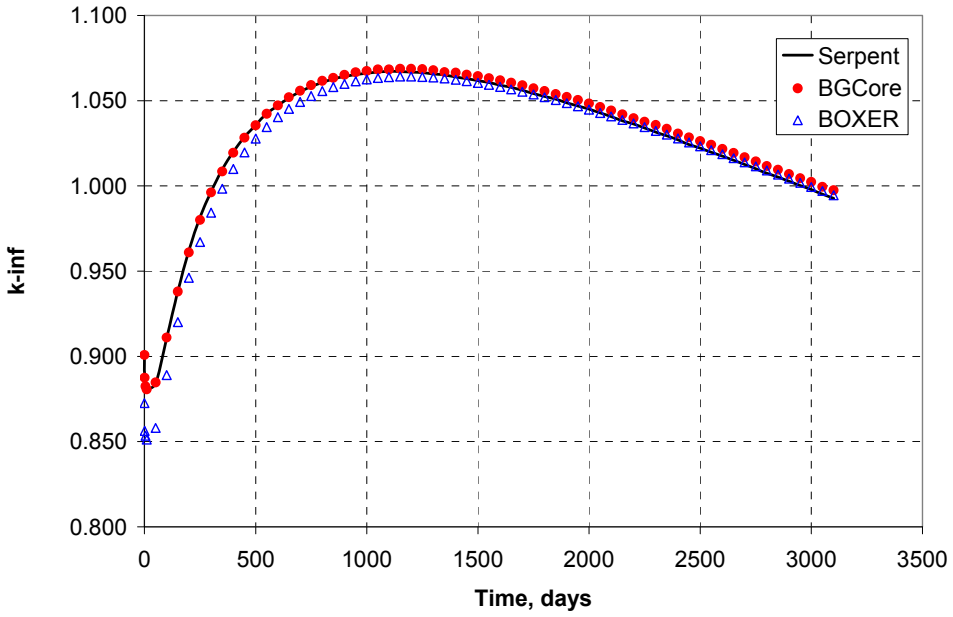

Figure 5: Comparison of k-inf for the 'REF' case with 20 $^{w}/_{0}$ seed enrichment

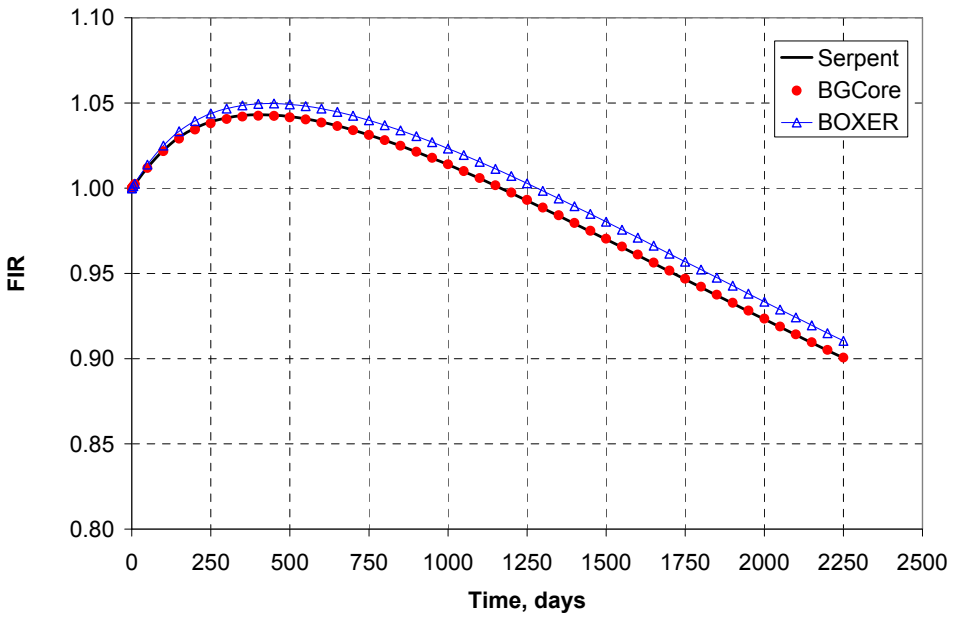

Figure 6: Comparison of FIR for the 'REF' case with 20 $^{w}/_{0}$ seed enrichment

3.2. Evaluation of Hexagonal Lattice Cases

Figure 7 and Figure 8 present the results of k-inf and FIR obtained from Serpent for the cases described in 'HEX-2' *Case: 1/6-th of hexagonal assembly layout.*

Table 1. The results indicate that HEX-1 design has more favorable neutronic performances. It is expressed in slightly longer cycle length and longer period of net breeding (FIR>1) than in the reference case. As for the HEX-2 design, no final conclusions can be made because both the cycle length and the period of net breeding are shorter than in the REF case. This is a result of the lower blanket volume fraction and smaller blanket zone radial dimensions, which results in less efficient breeding. However, the initial k-inf for HEX-2 case is somewhat higher than in the REF case, which may allow more flexible fuel management and lower power peaking. Table 3 quantitatively compares the results of the studied cases.

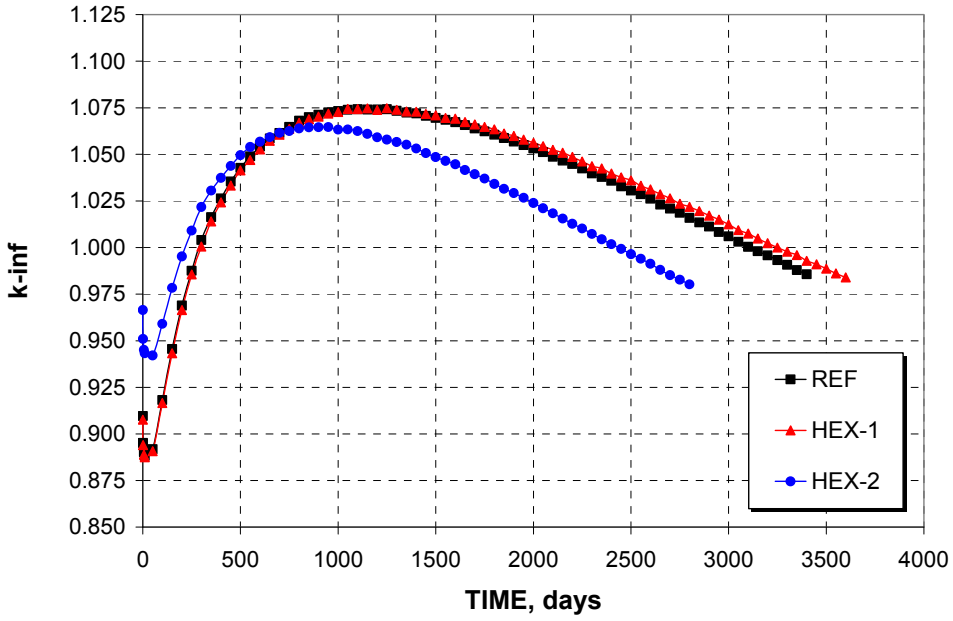

Figure 7: k-inf results for the different designs obtained by Serpent

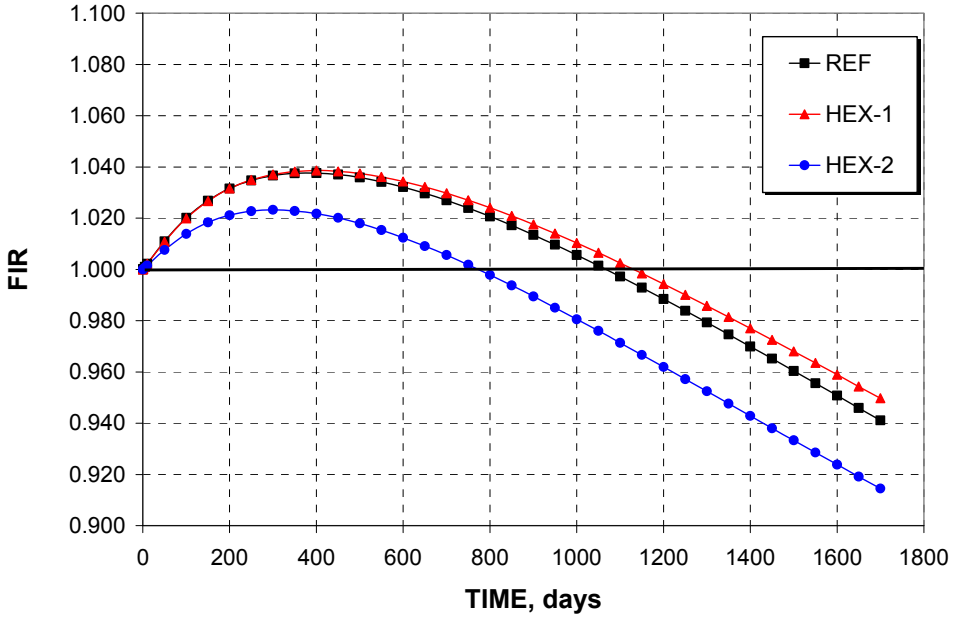

Figure 8: FIR results for the different designs obtained by Serpent

Table 3: Summary of performance parameters for studied designs

Design Parameters				Performance Parameters		
Case	Seed /Assembly Enrichment, $^{w}/_{0}$	Blanket pin radius, cm	Specific power, W/gHM	Initial k-inf	Cycle Length, days	T, days FIR>1
REF	21.0 / 2.92	0.53	16.07	0.909	950	1067
HEX-1	20.76 / 2.94	0.56	15.62	0.908	1002	1131
HEX-2	15.0 / 2.63	0.55	16.69	0.966	706	773

Since reactivity as a function of time curves for the studied cases exhibit highly non-linear behavior, the cycle length presented in Table 3 was evaluated from Eq. II

$$2\times\left[\int_{o}^{T_1}\left(\rho(t')-\rho_L\right)dt'+\int_{T_2}^{t}\left(\rho(t')-\rho_L\right)dt'\right]=\int_{T_1}^{T_2}\left(\rho(t')-\rho_L\right)dt' \qquad (II)$$

where, $\rho(t)$ is the reactivity curve obtained from the assembly burnup calculations and ρ_L is the assumed leakage reactivity worth (taken as 0.03), T_1 and T_2 are the time points at which the studied assembly becomes super-critical and becomes sub-critical again correspondingly. These points are shown in Figure 9 and they have different values in each particular case. The parameter *t* in Eq.II is the fuel residence time during which a 3-batch core can in principle sustain criticality. Equation II effectively represents integral neutron balance over the fuel residence time, where twice as many excess neutrons are produced by the fuel than required to sustain criticality. It should be noted that this balance does not account for loss of neutrons in reactivity control materials and effects of assemblies loading pattern. Furthermore, the time at which the excess neutrons are produced may not perfectly match the time at which these neutrons are needed to maintain criticality of the core. Therefore, in order to obtain realistic results, 3D full core analysis will be required.

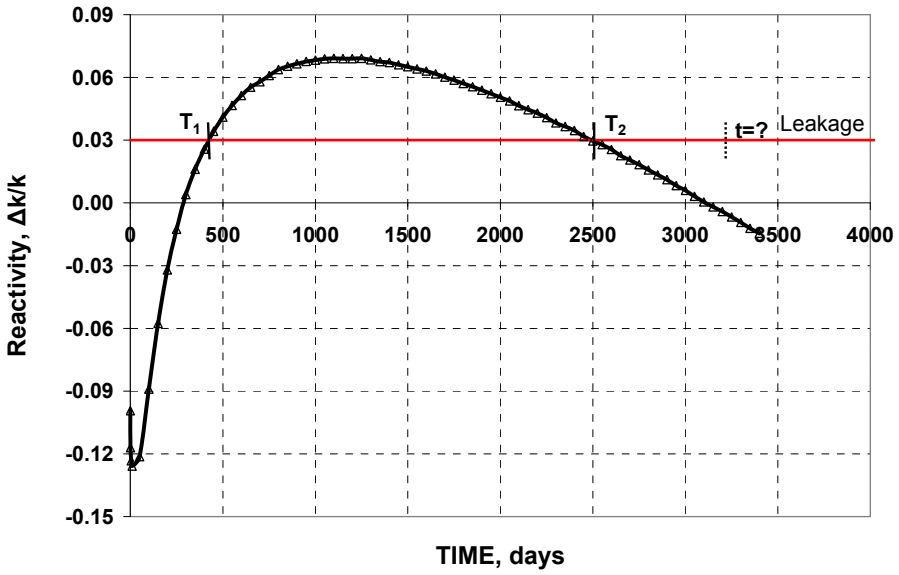

Figure 9: Representative reactivity vs. time curve for a typical SB assembly

3.3. Thermal Hydraulic Analysis Results

This section presents the results of thermal-hydraulic analysis for the considered cases ('HEX-2' *Case: 1/6-th of hexagonal assembly layout.*

Table 1) obtained with BGCore system. First, the reference (REF) square lattice case was considered with power density of 70 W/cm^3 to confirm previous findings. Then, we performed similar coupled neutronic – thermal hydraulic calculations to estimate the maximum core power density within the adopted design constraints (see Methodology section).

The limiting parameters used in this assessment include fuel central line (CL) temperature, axial position of the onset of nucleate boiling (ONB) and departure from nucleate boiling ratio (DNBR). These parameters were obtained assuming that the total core pumping power requirements remain equal to those of a typical PWR.

TH calculations were performed for all three designs for power density values ranging from 70 to 100 W/cm^3. As mentioned earlier, the greatest power imbalance between seed and blanket regions of individual fuel assemblies occurs in the seed region at BOL, when the assembly is subcritical. Consequently, one would expect that overall assembly power peaking factor (relative to the core average power density) will be lower than unity. Therefore, all the analyses for SB assemblies were performed at a burnup point, when the assembly becomes critical. It was also assumed that at that point the assembly has power peaking factor of 1.0, i.e. operates at core average power density.

The analysis results for the REF case with power density of 70 W/cc include coolant temperature distribution, average and Central Line fuel temperatures (shown in Figure 10 and Figure 11 respectively). The coolant outlet temperature is below the saturation point for both seed and blanket regions as shown in Figure 10. In addition, the central line fuel temperature is below the melting point as presented in Figure 11. In general, the obtained results indicate that the conclusions of previous studies are still valid and that the reference design satisfies the necessary thermal limits.

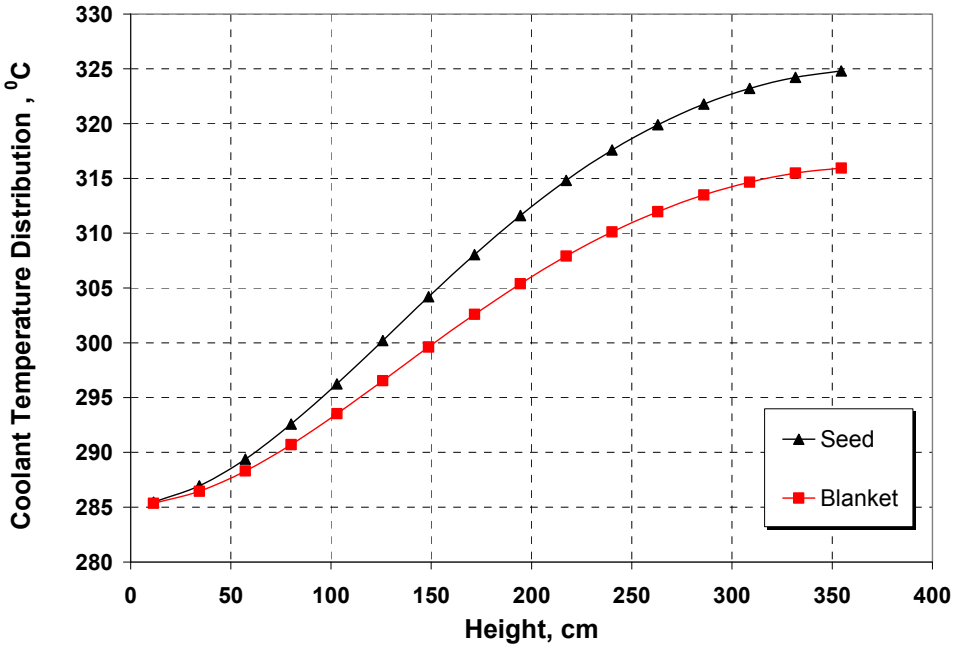

Figure 10: Coolant temperature in seed and blanket regions at 70 W/cm^3, REF case.

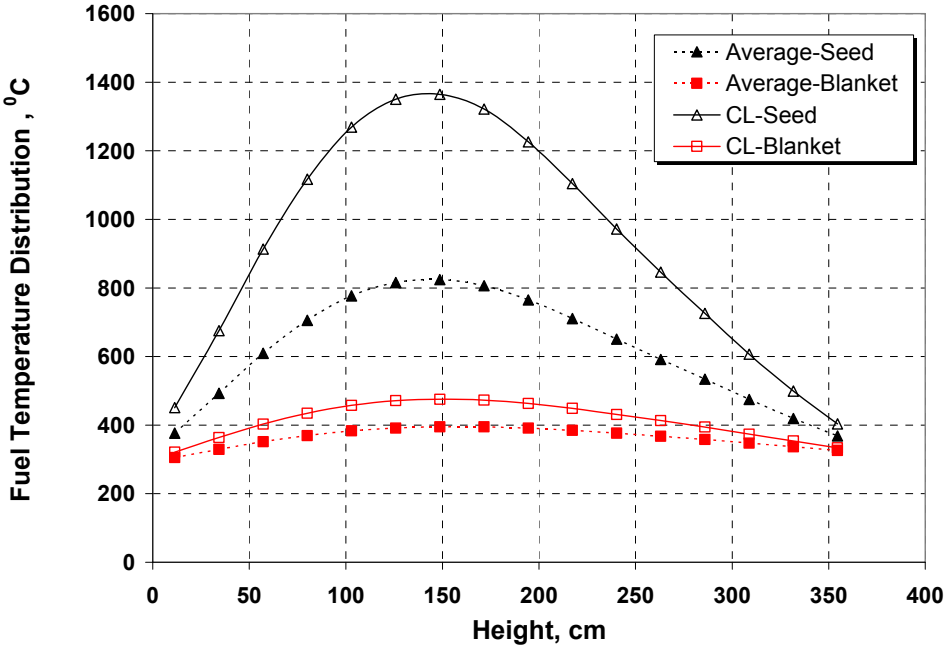

Figure 11: Fuel temperature in seed and blanket regions at 70 W/cm^3, REF case.

The results of the analyses for axial ONB location, maximum T_{CL}, and minimum DNBR in the seed region for hexagonal and square lattice cases are compared in Figure 12 through Figure 14. As expected, all TH limits are more restrictive in the seed region due to the higher peaking factor.

Both hexagonal lattice cases perform slightly better than the REF square lattice case. Figure 12 demonstrates that for a typical PWR pumping power, sub-cooled boiling does not occur until 85 W/cm^3 for the hexagonal lattices as opposed to the 80 W/cc in the square lattice. The bulk outlet coolant temperature is below saturation in all calculated cases for both seed and blanket sub-channels. The maximum T_{CL} in the seed is higher in the square design than in hexagonal configurations. However, its value is still far from the melting point with considerable margin as shown in Figure 13. The calculated minimum DNBR in the seed for all cases was much higher than the limiting value 1.3, as shown in Figure 14 and, again, slightly more favorable for the HEX cases.

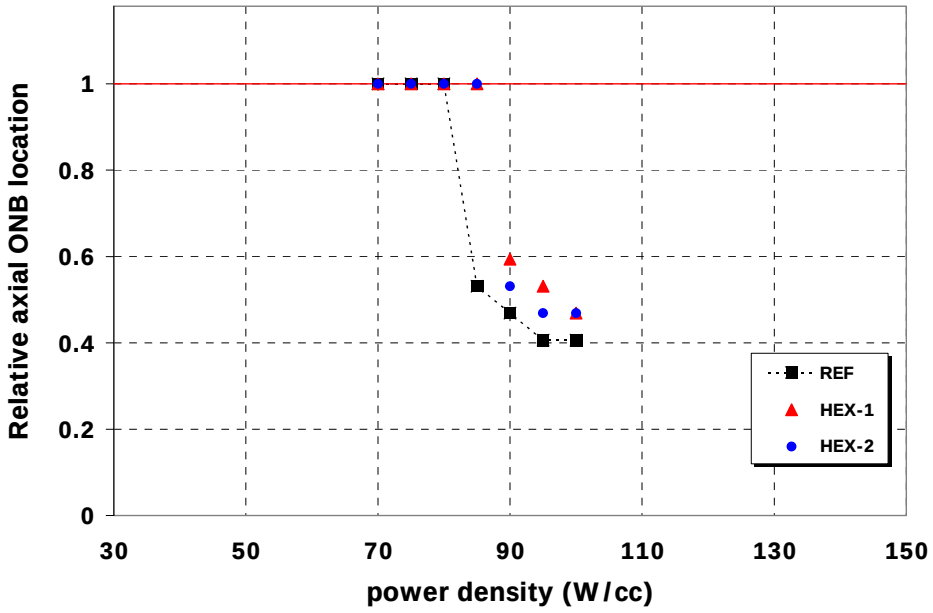

Figure 12: Axial location of ONB vs. power density in the seed

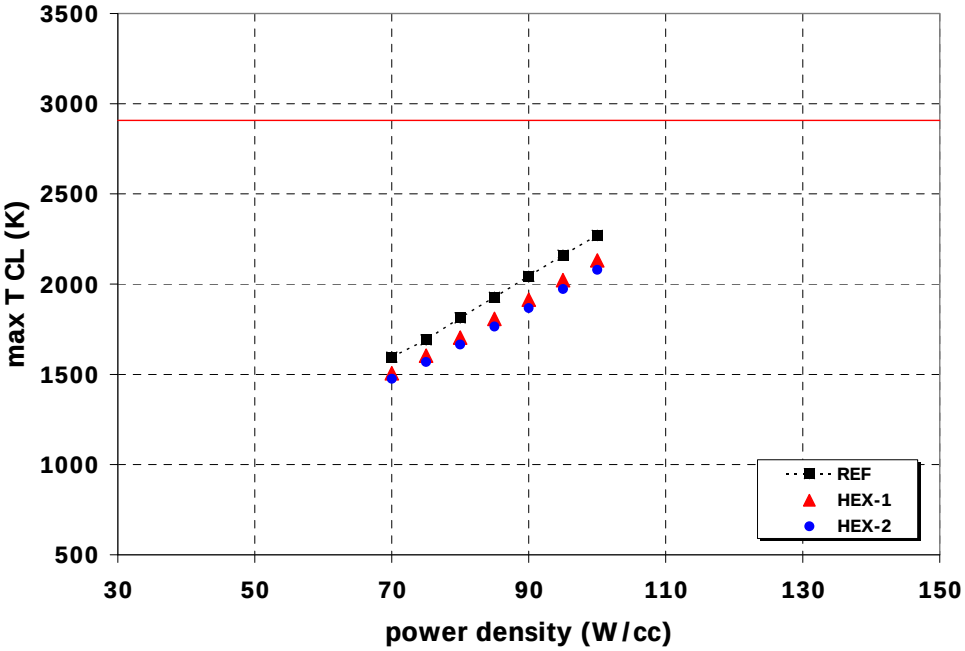

Figure 13: Maximum T_{CL} vs. power density in the seed

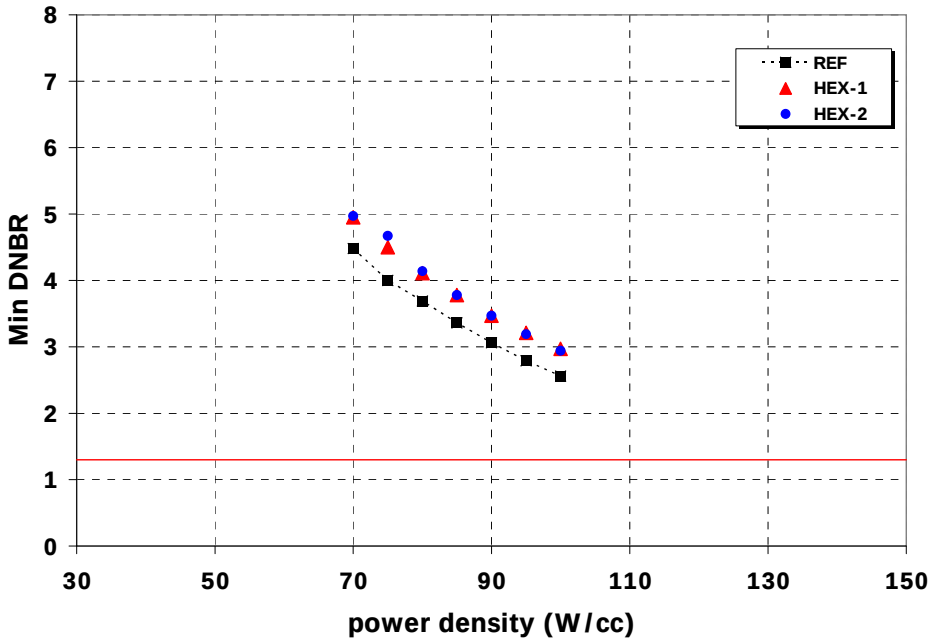

Figure 14: Minimum DNBR vs. power density in the seed at MOL

The presented results show that the most restricting parameter for all configurations is the ONB. The ONB occurrence requires significant reduction in power density, to 80 W/cm^3 in the square lattice and to about 85 W/cc in the hexagonal lattice designs for the core pumping power constrained case.

4. COCNLUSIONS AND FUTURE WORK

In this work, we compared hexagonal and square fuel lattice for high conversion PWR with U^{233}-Th fuel. The designs included 3 types of assemblies of different lattice configurations and pin layouts, seed enrichment and blanket pin dimensions. For each of the designs, a set of calculations was conducted in order to maximize the conversion ratio, while maintaining reasonably long fuel cycle length. The general design optimization criterion was to maintain fuel cycle length of at least one year, while achieving Fissile Inventory Ratio of one or higher at the end of the fuel in-core residence time.

The results indicate that these criteria can be met with both square and hexagonal fuel lattices. It was also observed that hexagonal lattices show slightly more preferable neutronic performance. The benefits of hexagonal lattice are expressed in longer period of time, during which both the assembly criticality and net breeding are possible.

In some hexagonal lattice cases, the period of net breeding was shorter. However, these cases had higher initial reactivity, which may offer more flexibility with respect to in-core fuel management, power peaking and reactivity control.

In order to verify feasibility of the studied designs from a TH point of view, the calculations were extended to include coupled neutronic – TH analyses. The main goal of this task was to

estimate maximum achievable power density, while maintaining thermal margins comparable to a typical PWR values.

From the heat removal point of view, hexagonal lattice cases also exhibit more favorable performance primarily because of the higher relative volume fraction (larger dimensions) of the seed region. This reduces both linear power and heat flux in the seed and therefore allows for some increase in the achievable power density.

This work is considered as preliminary feasibility study since all calculations were performed on an assembly lattice level. More detailed investigation of Th-^{233}U fuel SB concept must include detailed 3D core simulations and more realistic assumptions such as consideration of power distribution within seed and blanket zones, sub-channel analysis which includes possibility of lateral flow and consideration of guide tubes.

We also performed verification of the results obtained in previous studies with BOXER code by repeating the calculations with Monte Carlo codes SERPENT and BGCore. Although results obtained with Monte Carlo codes do not perfectly match those obtained with BOXER code, the main conclusion of previous studies remains valid i.e. the net breeding is achievable in seed-blanket PWR fuel assembly for both square and hexagonal lattices.

NOMENCLATURE

BOL	Beginning of Life
BU	Burnup
CL	Central Line
EFPD	Effective Full Power Days
EOL	End of Life
FIR	Fissile Inventory Ratio
HEX	Hexagonal
LWR	Light Water Reactors
MC	Monte-Carlo
MDNBR	Minimum Departure from Nucleate Boiling
MG	Multi-Group
ONB	Onset of Nucleate Boiling
PWR	Pressurized Water Reactor
REF	Reference
SB	Seed-Blanket
TH	Thermal Hydraulic
3D	Three-Dimensional

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