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Void Reactivity Coefficient for Hybrid Reactor Cooled by Liquid Metal

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Posted Date: 24 February 2025

doi: 10.20944/preprints202502.1839.v1

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Void Reactivity Coefficient for Hybrid Reactor Cooled by Liquid Metal

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Abstract: The negative value of void reactivity coefficient (α_V) is one of the most important passive safety properties for the operation of nuclear reactor. Herein, are presented calculated values of void reactivity coefficient for different geometries of reactors cooled by liquid lead (LFR) and sodium (SFR) with fuel U-238-Pu-239 and Th-232-U-233. The calculations were carried out for the reactors filled with either one or two types of fuel assemblies. The most interesting results are obtained for reactor filled with two different types of fuel assemblies (Hybrid reactor). The Hybrid reactor consists of central and peripheral types of fuel assemblies, using low enrichment fuel and high enrichment fuel, respectively. Both Hybrid reactors based on Uranium cycle (U-cycle) and Thorium cycle (Th-cycle) can maintain a negative value of void reactivity coefficient for wide range of reactor parameters. The calculation results of Hybrid reactor agree with experimental data from FBR-IME reactor.

Keywords: void reactivity coefficient; Lead Fast Reactor (LFR); Sodium Fast Reactor (SFR); Liquid Metal Cooled Reactor (LMCR); Thorium Breeder Reactor (TBR)

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1. Introduction

The liquid metal cooled reactors can be considered the future of nuclear plants due to of their potential for breeding efficiency and, consequently, economic justification. This type of reactor can achieve high coolant temperatures. Metal coolants remove heat from reactor core more rapidly, allowing for much higher power density. In other words, by using this type of reactor offers several advantages simultaneously, including economic viability, high power density and high coolant temperature.

The first experimental fast reactor was built in 1946 at Los Alamos. It was operated until 1953. The reactor was fueled with 2.5 liters of metallic plutonium, and the reactor core was cooled by mercury. [1]. In 1951, first experimental fast breeder reactor, EBR-I, which generated practical amount of useful power of 200 kWe, was constructed at the National Reactor Testing Station in Idaho [1]. Eutectic sodium-potassium served as the coolant of the reactor. Unfortunately, there are only about 20 experimental and commercial fast neutron reactors in operation worldwide [2,3]. One reason for this limited number is that fast reactors are typically cooled by liquid metal and have positive value of α_V coefficient, making it difficult to reduce this coefficient.

The negative value of α_V is a fundamental property of passive safety of all types of nuclear reactors. The second important is the passive removal of shutdown heat after the loss of coolant accident.

During recent years there has been stagnation in the development of liquid metal cooled fast reactors. However, the interest in fast reactors has been renewed due to necessity of limiting CO_2 emission and addressing global warming [1–27].

Fast reactors have nuclear characteristics that enable efficient use of uranium fuel and the capability to burn the long-lived actinides found in nuclear wastes [2]. The paper [2] presents the results of calculation of reactivity feedback for Sodium Fast Reactor (SFR) with U-cycle are presented. It provides a detailed a description of neutron physics and includes calculation results obtained by

 doi:10.20944/preprints202502.1839.v1

the perturbation method for voided condition in the SFR. Unfortunately the study does not provide estimated value of α_V.

To mitigate the overall reactivity increase during an accident, different concepts are used, including axial fuel expansion and radial core expansion. The core shape expansion is a passive method that utilizes thermal expansion and bowing of the ducts [4,5]. Additionally, there are other methods such as control rod driveline expansion or a special coolant cavity over core to reduce the α_V, as well as a passive shutdown system with hydraulically suspended rods [4,5].

The estimation of the value of α_V was studied in a water cooled reactor with a flexible fuel cycle which is a type of Boiling Water Reactor (BWR) [11].]. In the work exact perturbation calculations were applied, which quantitatively estimate α_V as a function of fuel rod diameter.

The effect of the distribution of voids inside the reactor on α_V was studied in the Na-cooled FBR-IME reactor, based on the Japanese JOYO experimental reactor. The central part of the reactor consist of 95 heterogeneous fuel assemblies, while peripheral part (outer/fertile) consist 295 assemblies of U-238 [12,13]. The peripheral part is used for conversion of U-238 to Pu-239. Unfortunately, it also plays a role as a reflector, which increases α_V. This work provides exact result of α_V for several distributions of no-sodium assemblies in central part of reactor.

The void reactivity was studied as affect of an unprotected loss of flow event (ULOF) in the MOX-fueled core based on the prototype reactor MONJU [14,15]. This reactor also utilized an inner core and an outer/fertile region.

Saturation concentration of U-233 in Th-cycle achieves 0.11 in fast reactor, while in in thermal reactor it is only 0.0137 [16,17]. Breeding and criticality in Th-cycle occur simultaneously in fast reactor, only [18]. In other words, to achieve the most effective and safe thorium reactor, a fast reactor with a negative value of α_V should be constructed [17–19].

Two large FSRs (3600 MWth): MOX-3600 and CAR-3600, and two medium FSRs (1000 MWth): MET-1000 and MOX-1000 and with different fuel types were compared in Ref.[20]. This excellent work presents various technical parameters of these reactors include numerical values of α_v for 100% void. All values of α_v are positive.

The reactors with a low value of α_V referred to as 'the low void worth core'(CFV) possess relatively complex core geometries. The cores are designed with radially or axially heterogeneous geometries, incorporating sodium plenums, fuel zones, fertile zones and absorbing zones. There are several reactors based on low void worth cores including the Russian reactor BN-800 [21,26], the Japanese concepts of Takeda [24–26] and Saito [23,26] and the French concepts CFV of Sciora [26] and ASTRID CFV of Beck [27]. For the CFV concept, the value of α_v is positive when a decrease in coolant density occurs in central part of the core (fissile and fertile zone) and negative when it occurs in the upper sodium plenum zone. This concept does not eliminate the risk of sodium boiling.

A characteristic feature of CFV cores is the leakage of neutrons primarily through the upper sodium plenum, during the voided conditions. This concept restricts the neutron leakage to the top of core base, which in turn limits the height of the active core.

This paper presents a concept of Hybrid reactor in which neutron leakage primarily occurs through the radial side area of the core. This approach allows for a higher value of neutron leakage rate that is proportional to the height of the core. The Hybrid concept consist two kind of fuel assemblies; assemblies with high enrichment fuel placed in the peripheral region and assemblies with low enrichment fuel located in central part of core. The study focuses on examination the influence of average fuel density and coolant volume fraction in the fuel cell on the decrease the value of α_v.

Additionally, the calculation results from Hybrid model are compared to experimental data from FBR-IME (Sec.8, [12,13]). The calculation results were obtained using Monte Carlo method, specifically employing the MCNP6.2 code [28].

2. Materials and Methods

The computer simulation model is based on the geometry of European Pressurized Reactor (EPR) [29,30]. This geometry was chosen because a large reactor core was necessary to study α_V in a

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general context. The modified model incorporates various type of coolants i.e. liquid sodium (Na) or lead (Pb). Additionally, it accounts for changeable volumes of fuel and coolant (Figures 1–5 and Tables 1–3) as well as different numbers of fuel assemblies [18,19]. The reactor core model consists 241, 137 or 101 assemblies (Figures 1–4). We use the A1 assembly in the reactor model, which employs a single type of fuel rod (Figure 6, [29]). The fuel is a mixture of U-238 and Pu-239 or Th-232 and U-233 to study the U-cycle and Th-cycle, respectively. Furthermore, three different values of fuel density are applied.

The reactor model has a changeable fraction of coolant in the volume of the fuel cell (VCR)

$$VCR = \frac{Coolant\_Volume}{Cell\_Volume}$$

The values of VCR parameters are in the width range of < 0.239, 0.976>.

Other parameter often used in literature is volume ratio of coolant and fuel (VCFR)

$$VCFR = \frac{Coolant\_Volume}{Fuel\_Volume}$$

It is assumed that, the thickness of cladding of fuel rods is equal to zero. For this reason VCFR=VCR/(1-VCR). This assumption is made to reduce the number of computations. It is a simplified model, intended solely for computational purposes, rather than for experimental research.

The wide range of values of the VCR or VCFR (Table 1) parameters enables studying a broad range of neutron flux spectrum (Sec.4). However in the Hybrid reactor was used only practical values of these parameters. This reactor model can achieve neutron flux characteristics typical of a fast reactor. For example, the average neutron energy (ANE) in fuel rods reaches 0.5 MeV, while the average neutron energy causing fission reaction (ANFE) for Hybrid reactor based on Na-cooled U-cycle is approximately 0.9 MeV in normal operating conditions and 1.1MeV for voided core state (Sec. 6).

Table 1. Dependence of VCR and VCFR parameter on radius of fuel rods.

Radius of fuel rod [cm]	VCR	$VCFR = \frac{VCR}{1 - VCR}$
0.11	0.976	40.7
0.21	0.913	10.5
0.31	0.810	4.3
0.41	0.667	2.0
0.51	0.485	0.94
0.62	0.239	0.32

The change of the VCR or VCFR parameter of the reactor model can be achieved by using assemblies with appropriate cell configurations, which entail the suitable fuel and coolant volumes (Figure 5, Table 1).

To reduce the density of fuel one can utilize porous materials or fill part of the fuel volume with air, for example. The simplest method for decreasing the average fuel density of fuel rods is to fill the central part of the rods with air.

The initial value of the effective multiplication factor (k_{eff}) is in the range of 1.05±0.02 for all cases. Uncertainty of α_V is discussed in Sec.7. In other words, the initial enrichment of fuel was determined for each case before the reactor was voided for all values of the VCR parameter, fuel type and geometry. Herein the reactor may contain one type of fuel assemblies (Figures 1–3) or two type fuel assemblies (Hybrid reactor, see Figure 4). The base parameters of the reactor cores are presented in

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Table 3. The core configuration presented in Figure 3 consists of 137 assemblies, referred to as the ‘small core’ or 101 assemblies, referred as the ‘very small core’.

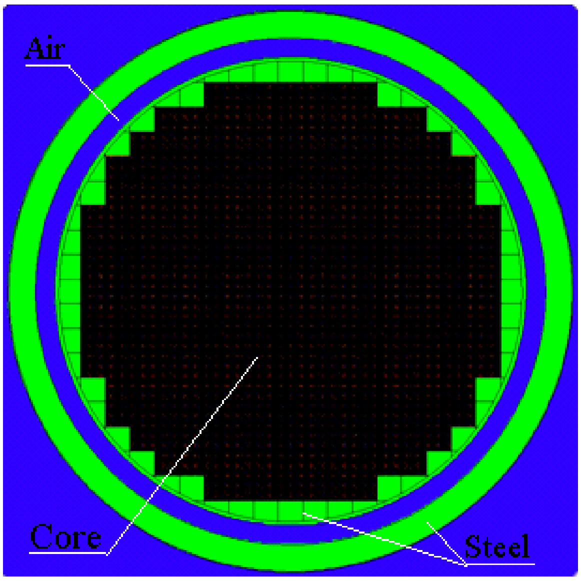

Figure 1. Large core, 241 assemblies. The colors mean: black - core, blue - air, light green – steel.

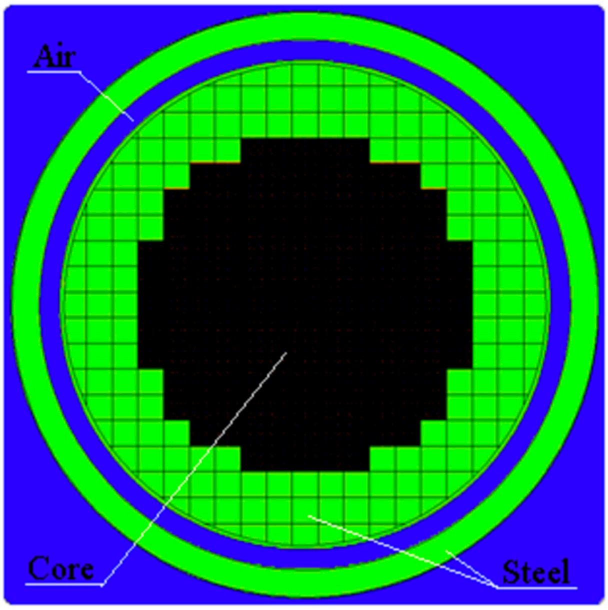

Figure 2. Small core + reflector, 137 assemblies. The colors mean the same as on Figure 1.

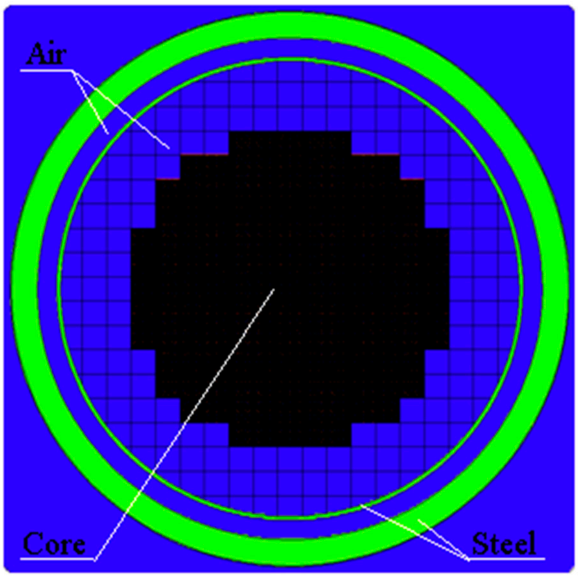

Figure 3. Small core of 137 assemblies or very small core of 101 assemblies. The colors mean the same as on Figure 1.

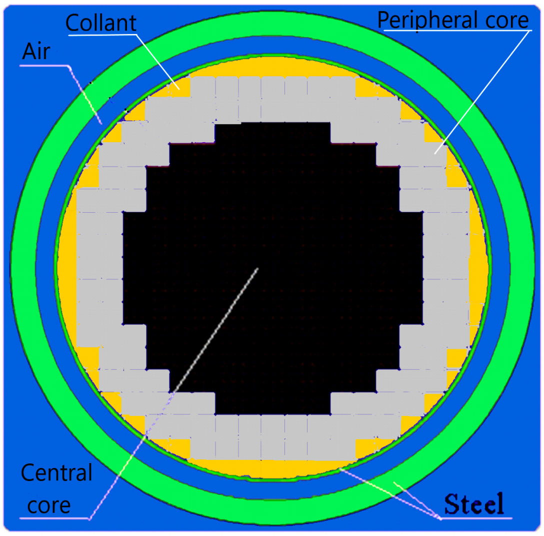

Figure 4. Hybrid reactor. Central core contains 137 assemblies. Peripheral core contains 104 assemblies. The colors mean the same as on Figure 1 but light grey and yellow color means a peripheral fuel assembly and coolant respectively.

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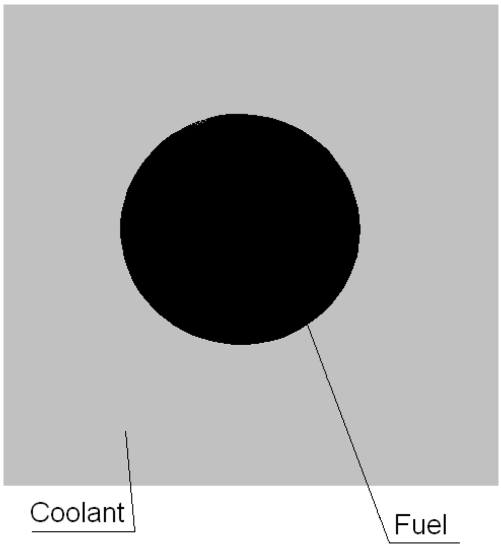

Figure 5. Fuel cell configuration. Dimension of the cell 1.26x1.26cm [29]. Radius of fuel rod and coolant and fuel volume are changeable.

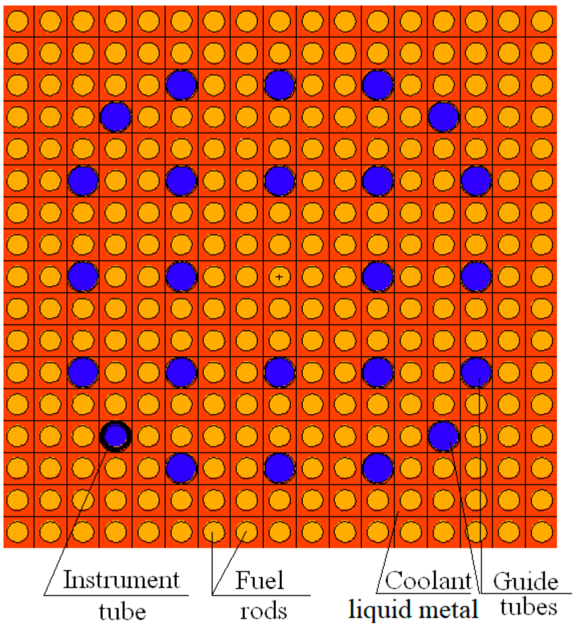

Figure 6. The modified A1 assembly based on Ref.[29]. .

Table 2. Base parameters of assembly A1 [29].

Name	Number
Cells	289
Fuel rod	265
Guide tubes	23
Instrument tube	1
Hight of fuel rod	420 cm

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Table 3. Basic parameters of reactor used for computation of α_V.

Geometry of core	Number of assemblies	Value of VCR	Fuel: U-cycle [g/cm³]	Fuel: Th-cycle [g/cm³]	Coolant
Figure 1	241	0.976, 0.913, 0.810, 0.667, 0.485, 0.239,	U-238+Pu-239 Density: 19.05, 11.0 (U-238+Pu-239)O_2 Density: 11.0, 8.0, 6.0	Th-232+U-233 Density 11.7	Pb, Na
Figure 2	137+steel reflector	as above	U-238+Pu-239 Density 19.05	as above	as above
Figure 3	137 Small core	as above	U-238+Pu-239, Density: 19.05 (U-238+Pu-239)O_2 Density: 11.0, 8.0, 6.0	Th-232+U-233, Density 11.7 (Th232+U-233)O_2 Density 10.0	as above
Figure 3	101 Very small core	as above	(U-238+Pu-239)O_2 Density 11.0, 8.0, 6.0	Not used	as above
Figure 5	137-central 104-peripheral	0.485, 0.271 -for central part 0.810, 0.667, 0.485 - for peripheral part	(U-238+Pu-239)O_2 Density: 11.0 - for central part Density: 11.0, 8.0, 6.0 – for peripheral part	Th-232+U-233 Density:10.0 for central part Density: 11.0, 8.0, 6.0 – for peripheral part	Na Pb

3. Definitions

There are different definitions of void reactivity coefficient [11]. For example, the "void reactivity coefficient" for BWR is defined as α_V at which the coolant flow rate reaches 90% of the nominal value [11]. In other case, the coolant is fully flow out, and the reactor vessel is filled with saturated steam. This case, is named as ''the 100% void reactivity coefficient''. The following definition of α_V is presented in ref.[11]

$$\alpha_V = \frac{\frac{k_1 - k_0}{k_1 k_0}}{V_1 - V_0}. \quad (1)$$

where,

α_V: void reactivity coefficient
k_1, k_0: effective multiplication factor at void fraction of V_1 and V_0, respectively:
V_1, V_0: void fraction at varied and nominal condition [%], respectively:

The definition in Eq.(1) suggests that α_V is a linear function of the void fraction.
However, this is not accurate, as α_V is not generally a linear function (Sec.4 and 5).
For this reason, we employed the following definition of α_V :

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$$\alpha_V(void) = \frac{k_{void} - k_{norm}}{k_{norm}}, \tag{2}$$

where,

void [%] is an average value of the void fraction in the fuel cells,

k_{norm}, k_{void} means effective multiplication factors at normally work and voided reactor core respectively.

The following definition often is used:

$$\Delta\rho(void) = \rho_{void} - \rho_{norm}, \tag{3}$$

where,

ρ_{norm}, ρ_{void} indicate the reactivity of reactor under normal operating condition and in a voided reactor core, respectively.

The definitions of α_V and $\Delta\rho$ are compared in Sec.7.

The definitions provided by Eq.(2 and 3) are more general than that of Eq.(1) because they do not assume a linear void dependence of $\alpha_V(void)$ on the void fraction. In other words, all values of $\alpha_V(void)$ are calculated directly for each specific value of void parameter.

$\alpha_V(100\%void)$ means 0% of coolant density. This means that the volume of coolant is filled by air at atmospheric pressure. This case simulates the Loss of Coolant Accident (LOCA).

$\alpha_V(n\%void)$ means that n% of part of coolant is evaporated and average density of coolant is equal to (100-n)% of its normal value. The normal value of coolant density is equal to the density of liquid metal.

The magnitude $\alpha_V(void)$ depends on a various parameters such as the VCR parameter or the type of fuel, and the size and shape of the reactor core (See Sec.4). If we are interested in α_V as a function of the VCR parameter and constant value of void, we can write it in the following form:

$$\alpha_V^{void}(VCR)$$

Whereas, if we are interested in α_V as a function of void parameter and with constant value of the VCR parameter we can express it in the following form:

$$\alpha_V^{VCR}(void)$$

Both function $\alpha_V^{void}(VCR)$ and $\alpha_V^{VCR}(void)$ are not linear functions of void and VCR parameters (See Sec.4 and 5).

4. Calculation Results for the Reactor Core Filled by Single Type of Assemblies

This section presents α_V for geometry of reactor cores depicted in the Figures 1–3, considering different parameter values as VCR, type of fuel, type of coolant, geometry and fuel assembly (Table 3). In this section we assume that the reactors consist of a single type of fuel assembly. The initial values of fuel enrichments are the same for both normal operation and during reactor void condition.

The Figures 7–10 present values of $\alpha_V^{100\%void}(VCR)$ for different types of core (Figures 1–3) and various combinations of fuel and coolant (from Table 3). These values can be both positive and negative. The values of $\alpha_V^{100\%void}$(VCR) are important because they indicate maximal positive value of α_V and minimal negative value for maximal value of the void parameter.

A very important is the value of $VCR_0^{100\%void}$ parameter for which $\alpha_V^{100\%void}(VCR_0^{100\%void})$=0. This is significant because for VCR<$VCR_0^{100\%void}$, it is not possible to obtain negative value of α_V. For these values of the VCR parameter reactor does not passively decrease the number of neutrons during the LOCA accident. In contrast, for reactors with a VCR greater than $VCR_0^{100\%void}$, the value of $\alpha_V^{100\%void}$ becomes negative, indicating that the reactor should be turned off during this accident. Please note that $VCR_0^{100\%void}$ is a characteristic parameter of a class (set) of reactors that differ only in the VCR parameter.

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All the $\alpha_V^{100\%void}(VCR)$ functions have negative values for sufficiently high values of the VCR parameter. For this reason, the VCR parameter is the most important factor to reduce the value of $\alpha_V^{100\%void}(VCR)$.

The greatest positive values of $\alpha_V^{100\%void}(VCR)$ one can be achieved with large cores or small cores that incorporate a reflector and for metallic fuel of U-238+Pu-239 (Figures 7–10). For this reason both a reflector and high value of a fuel density are unfavorable when the goal is to reduce the value of α_V.

The lower positive values of $\alpha_V^{100\%void}(VCR)$ are obtained for the thorium cycle (Figures 9 and 10). The primary reason for this result is the lower density of metallic thorium compared to that of metallic uranium (see Sec.6).

The negative value of $\alpha_V^{100\%void}(VCR)$ one can obtain especially easy for Pb-cooled Th-cycle (Figure 10). For this type of reactor, it is possible to obtain negative value of $\alpha_V^{100\%void}$ over a wide range of VCR parameter values.

The $\alpha_V^{100\%void}(VCR)$ function is partially increasing and decreasing function of *VCR* parameter for low and high value of *VCR* parameter, respectively. The absorption and leakage components of α_v are responsible for the increasing and decreasing part of the α_v function, respectively. To obtain negative value of α_v than the leakage component must be greater than the sum of absorption and spectrum components α_v [26].

The components mentioned above are closely interconnected. A high value of VCR indicates a high positive value of absorption component; however, this does not necessarily imply a higher value of αv. In fact, a high value of VCR parameter generally corresponds to lower value of fuel mass in the fuel cell which increases the leakage component. As the VCR value becomes sufficiently high, αv tends to decrease.

The small core makes it easier obtaining the negative value of $\alpha_V^{100\%void}$ for different type of reactors cores without reflectors (Figures 7–10). This finding extends the existing literature [2–9,20]. However, another significant parameter that decreases the value of $\alpha_V^{100\%void}$ is the density of the fuel (Figure 7). The most effective method of reducing of $\alpha_V^{100\%void}$ is applying the three methods simultaneously: reducing fuel density, decreasing the size of core and increasing the value of VCR parameter (Figures 7–11).

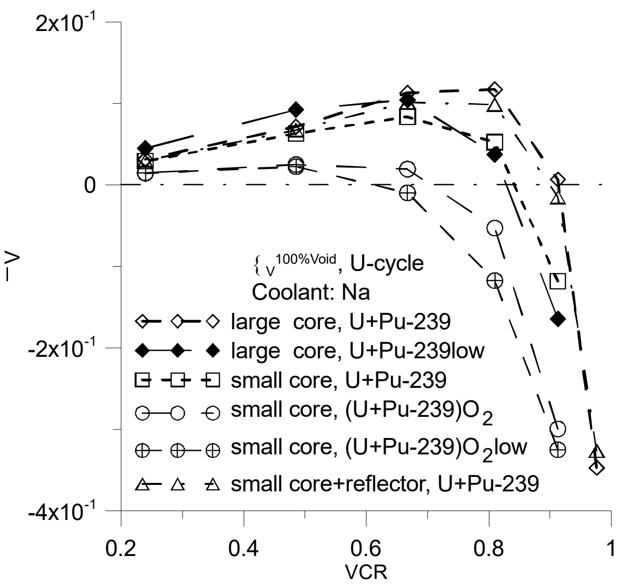

Figure 7. $\alpha_V^{100\%void}(VCR)$ for large core, small core and small core with reflector as a function of VCR parameter. Coolant: Na, Fuel: U+Pu means U-238+Pu-239 and (U+Pu-239)O$_2$ means U-238O$_2$+Pu-239O$_2$. 'low' means low density of fuel is equal to 8 g/cm3. Statistical error is equal to 0.002.

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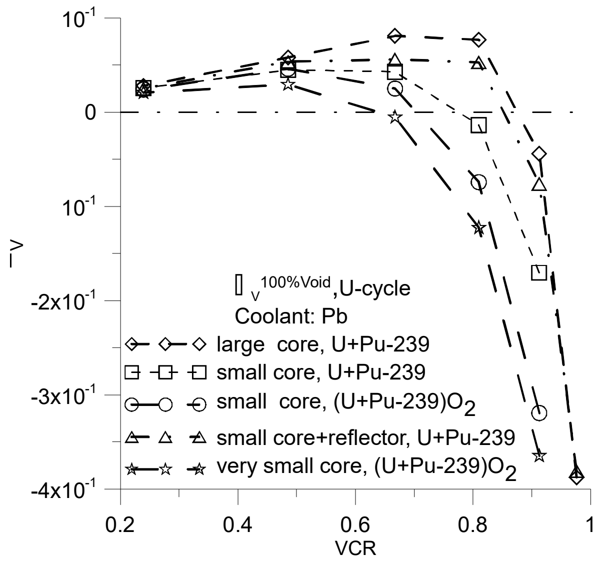

Figure 8. Same as on the Figure 7 but for coolant Pb.

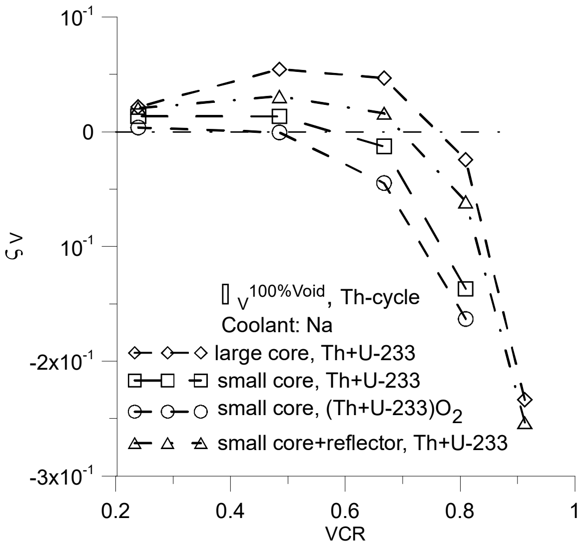

Figure 9. Same as on the Figure 7 but for Th-cycle. Fuel: Th+U-233 means Th-232+U-233 and (Th+U-233)O₂ means Th-232O₂+U-233O₂.

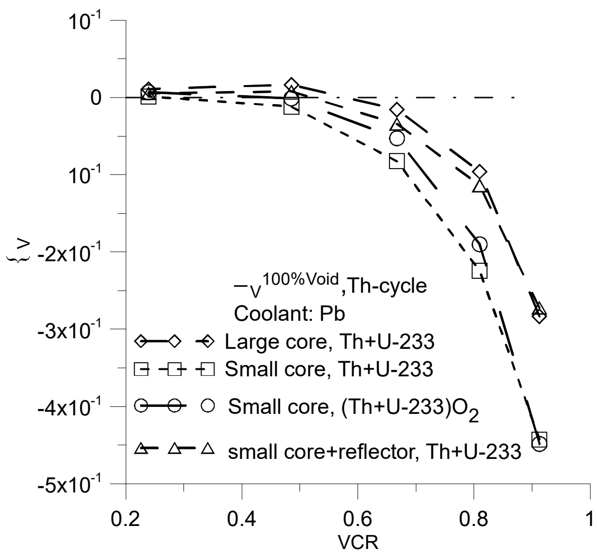

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Figure 10. Same as on a Figure 9 but for coolant Pb.

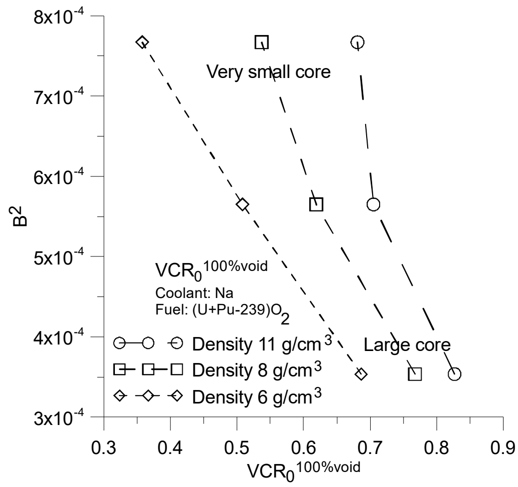

Figure 11. Geometry buckling B^2 parameter [31] versus the $VCR_0^{100\%void}$ for average fuel density equal to 11, 8 and 6 g/cm^3. Reactors geometry are from Figures 1 and 3.

The average fuel density plays important role in decreasing the value of $VCR_0^{100\%void}$. Unfortunately only a very small core with fuel density of 6 g/cm^3 achieves a $VCR_0^{100\%void}$ value of less than 0.5. The method of reducing the average density of the fuel does not practically allow for the construction of a large reactor filled solely with one type fuel assembly with a negative α_V (Figure 11).

Please note that the above conclusion applies to reactor core containing only one type of fuel assembly of U-cycle.

The large and medium reactors presented in Ref. [20] have reflectors and positive value of αv .

The results in this section predict that large reactors without the reflectors also exhibit a positive value of α_v. Why is it so difficult to obtain a low value of $VCR_0^{100\%void}$? This is because decreasing $VCR_0^{100\%void}$ simultaneously increases the fuel mass in the fuel cell, which in turn decreases neutron flux leakage.

Unfortunately, the results presented in this section exhibit a high value of $VCR_0^{100\%void}$. However, they provide guidance on how to significantly reduce this quantity (see Sec. 6).

5. Loss of Neutrons

This section presents relation between α_v and neutron loss coefficient *Loss* Eq.(4), and β_v Eq.(6) as a functions of VCR parameter. The fraction of lost neutrons can be defined in the following form:

$$oss(VCR) = Absorption(VCR) + Escape(VCR), \tag{4}$$

$$Fission(VCR) = 1 - Loss(VCR), \tag{5}$$

where:

Absorption - means absorption fraction without absorption neutron induce fissions i.e. (n,γ)
Escape - means total escaped fraction of neutrons,
Fission - means fraction of all actinides fission reaction.
Loss is a function of VCR and void parameter.

The β_v function can be defined as relative change in the *Loss(VCR)* function

$$\beta_V^{n\%void} = \frac{Loss_{void} - Loss_{norm}}{Loss_{norm}} \tag{6}$$

where

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$Loss_{norm}$, $Loss_{void}$ – means loss neutrons fraction at 0% void (normal work) and n% void respectively.

The *Incore(VCR)* function can be defined as the neutron fraction inside the core in the following form:

$$Incore(VCR) = 1 - Escape(VCR) = Absorption(VCR) + Fission(VCR). \quad (7)$$

Comparisons between $\alpha_V^{n\%void}$ and $\beta_V^{n\%void}$ for the Pb-cooled U-cycle in a small core reactor as a function of the VCR parameter are presented in Figure 11. Similarly, we obtained relationships between $\alpha_V^{n\%void}$ and $\beta_V^{n\%void}$ for all cases discussed in this work.

It is important to note, that $\beta_V^{100\%void}$ has negative and positive values. The positive values means positive values of $\Delta Loss = Loss_{void} - Loss_{norm}$ and decreasing number of neutrons in the core. Conversely, negative value of $\Delta Loss$ suggests an increases in the number of neutrons in the reactor core. In other words, the presence of void can either increase or decrease the neutron populations in the core, which, in turn, changes the sign of α_V.

Please note, that the *ΔLoss* fraction is equal to the negative value of the *Δfission* fraction (Figure 12). In other words, the positive value of the *ΔLoss* fraction indicates a decrease in the fission fraction and a negative value of α_V. The primary reason for this is the high value of *ΔEscape* fraction during evaporative cooling. The calculations results take into account changes of neutron flux energy distribution and cross sections of corresponding reactions during reactor voiding. Specifically, the results consider the increase in fission cross sections induced by an increase in the average neutron energy. However, this increase in the fission cross section can be mitigated by decrease in the neutron fraction within the reactor core, which is caused by in neutron escape. The *Incore(VCR)* function can be useful for illustrating this effect (Figure 12).

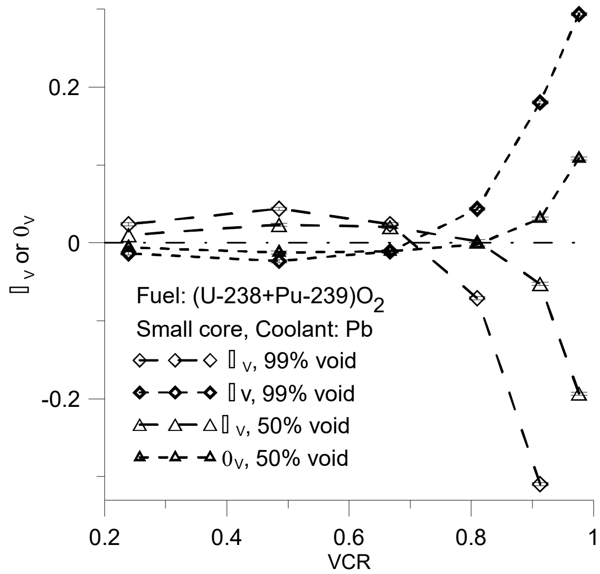

Figure 12. The $\alpha_V^{99\%void}$, $\beta_V^{99\%void}$ $\alpha_V^{50\%void}$ and $\beta_V^{50\%void}$ for U-cycle and coolant of Pb, Small core.

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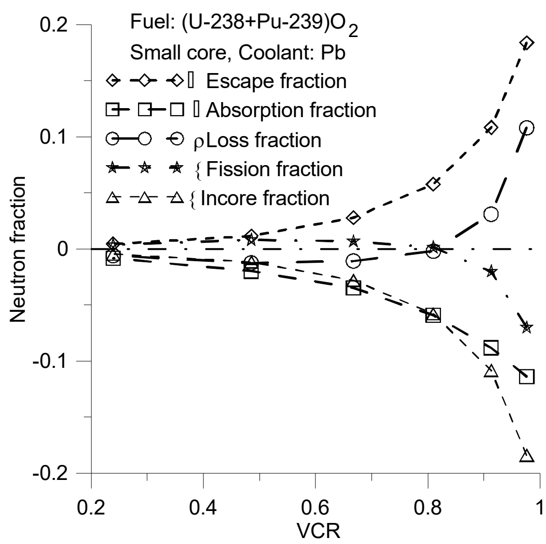

Figure 13. Change of the Escape, Absorption, Loss, Fission and Incore fraction during 50% void. Small core.

The results obtained in this section help us to get negative value of α_V for Hybrid reactor (see Sec.6.).

6. Calculation Results for Hybrid Reactor.

Using results from the previous sections we can derive intriguing outcomes for the Hybrid reactor (Figure 4, Table 3). Hybrid reactor contains two part of core: central core and peripheral core. The central core contains fuel assembly with an optimal value for the VCR=0.485 (VCFR=0.94) parameter, standard fuel density and a low value of fuel enrichment. In contrast, the peripheral core is composed of fuel assembly with a high level of fuel enrichment and three values of both the VCR parameter and fuel density. By simultaneously utilizing these three above parameter characteristics for the peripheral assemblies, we achieve the smallest values of α_V. The calculations were conducted with of central fuel enrichments of 1 and 5 % for U-cycle and Th-cycle. The enrichment of peripheral fuel was adjusted to ensure k_{eff}=1.05±0.02.

Herein two examples of Hybrid reactors are presented: Na-cooled U-cycle and Pb-cooled Th-cycle. . It is important to note that the Na-cooled U-cycle predicts maximal values of α_V whereas the Pb-cooled Th-cycle exhibits minimal values of α_V when reactor core is filled with a single type of assemblies (compare Figures 7–10).

The Pb-cooled Th-cycle Hybrid reactor has significantly lower value of α_V compared to the corresponding Na-cooled U-cycle (Figures 14 and 15). This indicates that Th-cycle reactors are characterized by substantially greater passive safety than U-cycle reactor.

The values of $VCR_0^{100\%void}$ are significantly less than 0.485 for all presented cases. While exact values of $VCR_0^{100\%void}$ were not calculated but one can be easily estimated using the $\alpha_V(VCR)$ function and extrapolation method (Figures 14, 16 and 17). It is important to note, that the values of VCR for central and peripheral part are significantly greater than $VCR_0^{100\%void}$. This can be expressed in the following inequalities: $VCR^{central} > VCR_0^{100\%void}$ and $VCR^{peripheral} > VCR_0^{100\%void}$. These results indicate that α_V has negative value (see Sec.5).

The negative value of α_V can be explained in another way. Please note, that the value of β_V is positive for all the cases presented in this section (Figure 16). A positive value of β_V always corresponds to negative value of α_V (Sec.5). The primary reason of positive value of β_V is a high value of *ΔEscape* quantity which indicates an increase of escaping neutrons in a voided reactor core, an increase in *ΔEscape* leads to decrease in *ΔIncore*, that is, it reduce the fraction of neutrons in the core. The relative increase in *ΔEscape* exceeds 60% and the average energy of the escaping neutrons increases by more than 20%. This results in reduction the average value of neutron flux density in the

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peripheral part of the core (Figure 17.). Consequently this leads to a decrease in fission reaction within the core, as evidenced by the negative value of *Δfission* Figure 18).The *Δfission* fraction is negative for all samples presented in this section.

Please note on the cases of VCR=0.485 and normal value of fuel density for both central and peripheral assemblies. In these scenarios, a significant difference in fuel enrichment between peripheral and center is sufficient to obtain negative value of α_V. This phenomenon occurs because the Hybrid reactor consist high fuel enrichment in peripheral region and allows a high value of leakage neutrons during voided reactor. Please note, that analogical large reactor Na-cooled U-cycle based on single type of assembly has $VCR_0^{100\%void}$=0.827 and positive value of $\alpha_V = 0.03$.

One can observe that α_V is a decreasing function of peripheral VCR parameter (Figure 14) and an increasing function of peripheral fuel density (Figure 15). In other words, the negative value of α_V is decreasing function of the both difference of fuel enrichment, fuel density, as well as the VCR parameter between central and peripheral parts (Tables 4 and 5, Figure 15).

The ANE and ANFE have a weak dependence on the peripheral fuel density (Figure 16). However, α_V is an increasing function of peripheral fuel density. For these reasons, it is advantageous to reduce the peripheral fuel density.

The type of coolant and fuel has a significant impact on the α_V value. The higher value of cross section on neutron absorption of coolant increases of α_V value. An important factor contributing to an increase in α_V is the increase in the number of fission reaction induced by fast neutrons (Table 6) and ANFE for the voided reactor core (Figures 19 and 20). These magnitudes have significantly greater values for the U-cycle compared to the Th-cycle. The reason for this is that the fission cross section (FCS) for U-233 is higher than that of Pu-239 in the intermediate neutron energy range (Table 6). Additionally, the FCS for Th-232 is lower than that of U-238 for neutron energies higher than 0.5 MeV. Number of directly fission events of U-238 is significantly greater than Th-232.

The geometry of Hybrid reactor presented in this section is not yet optimized. Furthermore, reducing of the value of α_V can be achieved by increasing the outer surface area of peripheral part at the top and bottom of the core, for example.

Herein is $\alpha_V^{100\%void}$, is primarily calculated. However neutron flux density, power density and evaporation rate of the coolant in peripheral region are significantly greater than in central region. For these reasons, the rate of decrease in the coolant density in peripheral region will be greater than in the central region. Why is which α_V(void) functions were calculated for both total void and peripheral void. The α_v for the peripheral void was calculated for two cases: Partial Void-A and Partial Void-B.

Partial void–A refers to the case in which the average coolant density was changed only in the peripheral assemblies. In this scenario, the thin layer of coolant between the fuel assemblies and core barrel has a constant normal density. This situation does not reflect a real case.

Partial void–B, on the other hand, describes the case in which the average coolant density was modified in the peripheral region of reactor specifically in the peripheral assemblies and the thin layer of coolant between the fuel assemblies and the core barrel. This represents a more realistic scenario.

The difference between Partial Void-A and Partial Void-B predicts a significantly impact of the peripheral thin layer of coolant on the α_v.

The results presented in this section indicate that the value of α_V is negative for all cases of the Hybrid reactor (Tables 4 and 5 and Figures 14–21).

The concept of Hybrid reactor allows for the construction a large and efficient reactor with a negative value of α_V. A low value of central fuel enrichment allows the use of natural uranium or thorium, as the criticality of Hybrid reactor is determined by peripheral part of reactor. In other words the central region should be consists blanket/fertile assemblies for conversion fuel. The fuel conversion ratio (CR) is decreasing function of fuel enrichment. For this reason the CR for peripheral region will be less than 1 [17,18].

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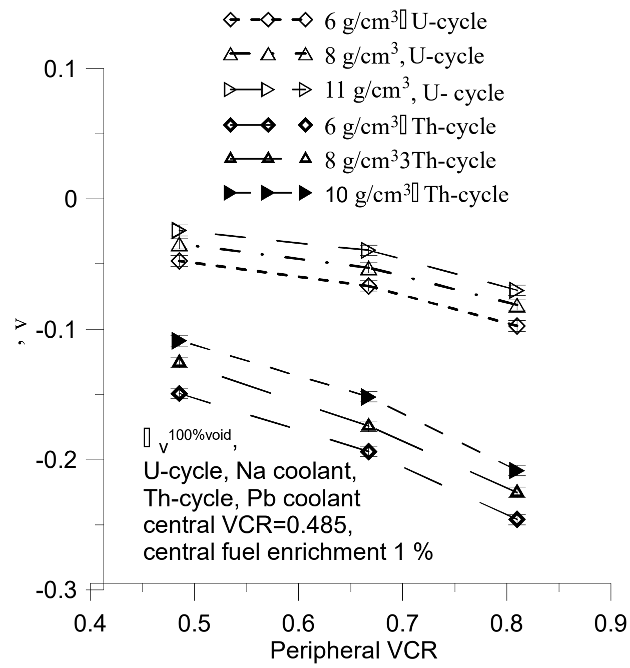

Figure 14. The $\alpha_V^{100\%void}$ for U-cycle and Th-cycle as a function of peripheral VCR for fuel density equal to 6, 8 and 11g/cm.

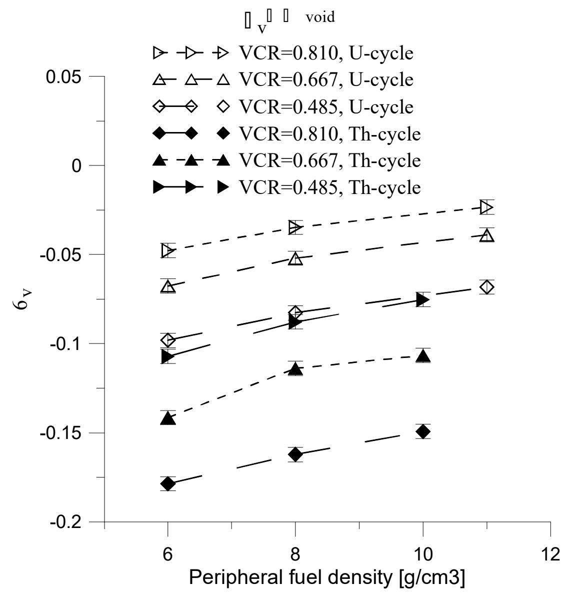

Figure 15. The $\alpha_V^{100\%void}$ for Na-cooled U-cycle and Pb-cooled Th-cycle as a function of peripheral fuel density for fixed peripheral VCR=0.810, 0.667 and 0.485. Fuel central enrichment is equal to 1% and 5% for U-cycle and Th-cycle respectively.

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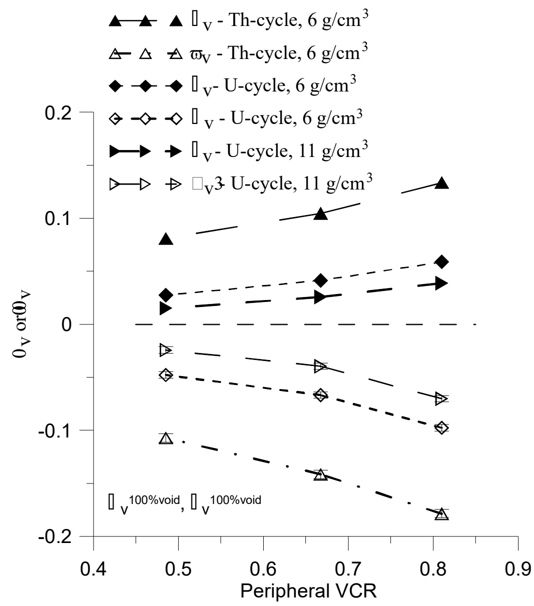

Figure 16. The $\alpha_V^{100\%void}$ and $\beta_V^{100\%void}$ for Na-cooled U-cycle and Pb cooled Th-cycle as a function of peripheral VCR. Central enrichment is equal to 1 and 5% for U-cycle and Th-cycle respectively.

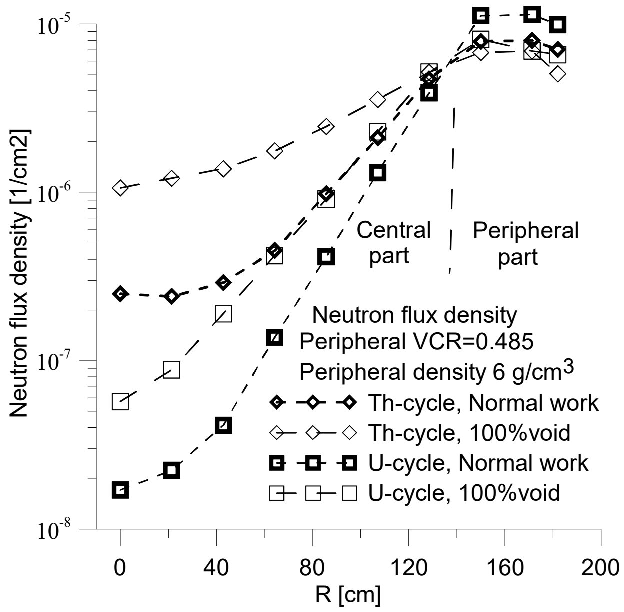

Figure 17. Average neutron flux density in fuel rods as a function of distance from the reactor axis. The value of average neutron flux density was calculated on the central part of fuel rod in the range (-179.5,174.4) cm and normalized per one initial neutron. Central enrichment is equal to 1 and 5% for U-cycle and Th-cycle respectively.

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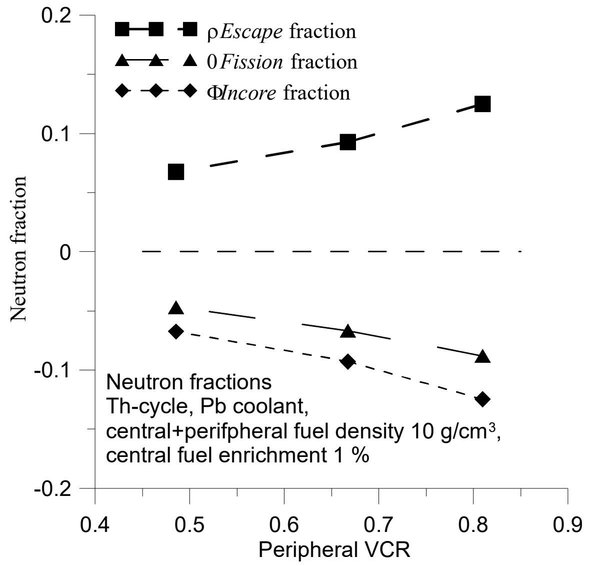

Figure 18. *ΔEscape, ΔFission, ΔIncore* fraction for Pb-cooled Th-cycle reactor.

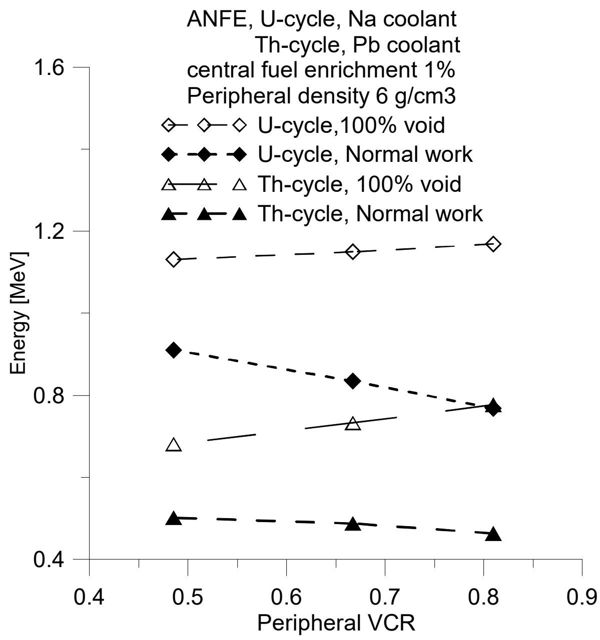

Figure 19. The ANFE for Na-cooled U-cycle and Pb cooled Th-cycle as a function of peripheral VCR.

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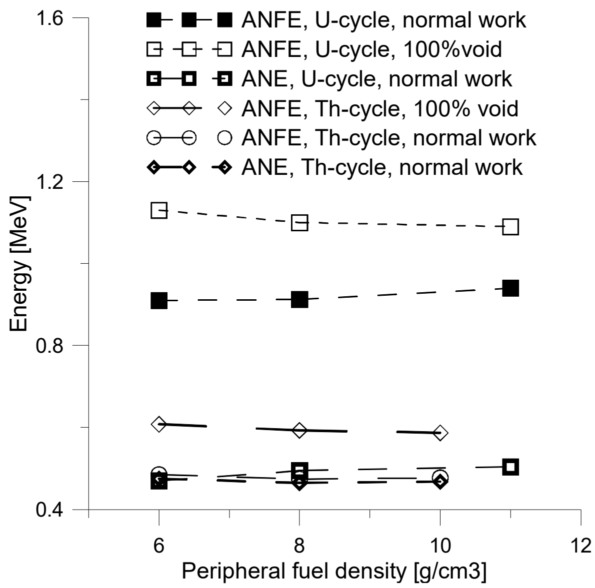

Figure 20. The ANFE and ANE for Na-cooled U-cycle Pb cooled Th-cycle as a function of peripheral fuel density for peripheral VCR=0.485. Central fuel enrichment is equal to 1 and 5 % for U-cycle and Th cycle respectively.

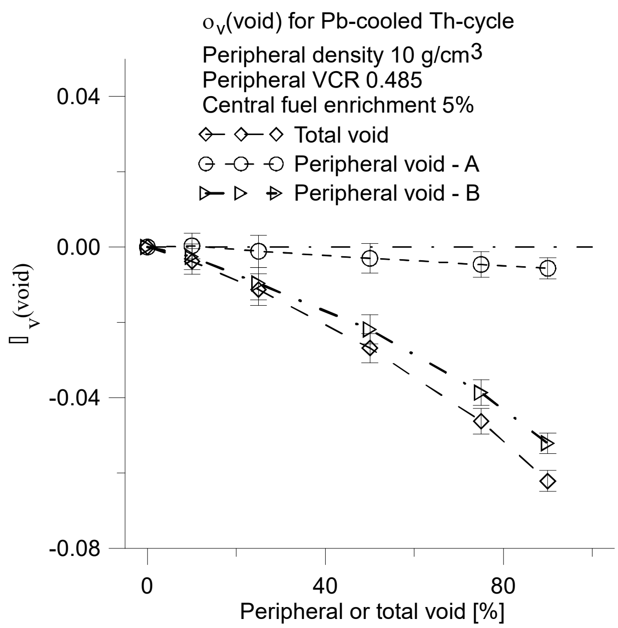

Figure 21. The α_v(void) function for Pb-cooled Th-cycle as a function of peripheral or total void. The errors represent total uncertainty including perturbation error for the changing of average density of the coolant of 10%.

Table 4. Base parameter of Hybrid reactor for central part: VCR=0.485, fuel (U-Pu-239)O_2 , fuel density 11 g/cm3, coolant Na.

Peripheral VCR	Peripheral fuel density	$\alpha_V^{100\%void}$ Error <0.004	Peripheral enrichment	ANFE for normal work [MeV]	ANFE for 100% void core [MeV]
Central fuel enrichment 1% of Pu-239					
0.810	11	-0.07018	27	0.81	1.15
0.667	11	-0.03958	19	0.88	1.11

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0.485	11	-0.02428	14	0.94	1.09
0.810	8	-0.08147	40	0.78	1.15
0.667	8	-0.05296	25	0.85	1.12
0.485	8	-0.0346	17	0.91	1.10
0.810	6	-0.0976	50	0.77	1.17
0.667	6	-0.06691	30	0.83	1.15
0.485	6	-0.04775	20	0.91	1.13
Central fuel enrichment 5% of PU-239					
0.810	11	-0.038	26	0.81	1.08
0.667	11	-0.015	17	0.87	1.07
0.485	11	-0.005	13	0.93	1.06
0.810	8	-0.052	33	0.79	1.09
0.667	8	-0.024	21	0.85	1.08
0.485	8	-0.011	15	0.91	1.08
0.810	6	-0.063	42	0.78	1.10
0.667	6	-0.033	24	0.84	1.09
0.485	6	-0.021	19	0.89	1.07

Table 5. Base parameter of Hybrid reactor for central part: VCR=0.485 and fuel (Th-U233)O_2 , fuel density 10 g/cm3, coolant Pb.

Peripheral VCR	Peripheral fuel (Th-U-233)O_2 density	$\alpha_V^{100\%void}$ Error <0.004	Peripheral enrichment	ANFE for normal work [MeV]	ANFE for 100% void core [MeV]
Central fuel enrichment 1% of U-233					
0.810	10	-0.208	26	4.58E-01	7.26E-01
0.667	10	-0.152	19	4.81E-01	6.78E-01
0.485	10	-0.109	14	4.95E-01	6.29E-01
0.810	8	-0.225	30	4.55E-01	7.46E-01
0.667	8	-0.174	20	4.71E-01	6.94E-01
0.485	8	-0.126	16	5.02E-01	6.52E-01
0.810	6	-0.246	39	4.63E-01	7.77E-01
0.667	6	-0.194	28	4.87E-01	7.32E-01
0.485	6	-0.149	19	5.01E-01	6.81E-01
Central fuel enrichment 5% of U233					

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0.810	10	-0.149	24	4.25E-01	6.27E-01
0.667	10	-0.107	17	4.48E-01	6.03E-01
0.485	10	-0.070	14	4.75E-01	5.87E-01
0.810	8	-0.162	29	4.26E-01	6.37E-01
0.667	8	-0.114	17	4.48E-01	6.16E-01
0.485	8	-0.088	15	4.65E-01	5.93E-01
0.810	6	-0.179	37	4.27E-01	6.50E-01
0.667	6	-0.141	24	4.45E-01	6.30E-01
0.485	6	-0.107	18	0.46842	6.08E-01

Table 6. Intermediate and fast fission fraction Na-cooled U-cycle and Pb-cooled Th-cycle and peripheral VCR=0.485.

Peripheral fuel density [g/cm3]	Intermediate/fast fission fraction, U-cycle, normal work, Central enrichment 1% [625eV-0.1MeV]/ [>0.1MeV]	Intermediate/fast fission fraction, U-cycle, 100%void Central enrichment 1% [625eV-0.1MeV]/ [>0.1MeV]	Intermediate/fast fission fraction, Th-cycle, normal work, Central enrichment 5% [625eV-0.1MeV]/ [>0.1MeV]	Intermediate/fast fission fraction, Th-cycle, 100%void, Central enrichment 5% [625eV-0.1MeV]/ [>0.1MeV]
6	0.40/0.60	0.30/0.70	0.44/0.56	0.41/0.59
8	0.40/0,60	0.31/0.69	0.45/0.55	0.42/0.58
10 or 11	0.41/0.59	0.33/0.67	0.45/0.55	0.42/0.58

Influence of the Central Region Parameters on α_v.

In this section, we examine the influence of the central region parameters: the average fuel density and the central VCR parameter on α_v, while the central fuel enrichment maintained at a stable level of 1%.

The peripheral fuel density is set at 11 g/cm3, and the peripheral VCR values are 0.485, 0.667, 0.810.

We compare the $\alpha_V^{100\%void}(VCR)$ functions for central fuel densities of 11 g/cm3 and 8g /cm3, as well as central VCR values of 0.271 and 0.485 (Figure 22).

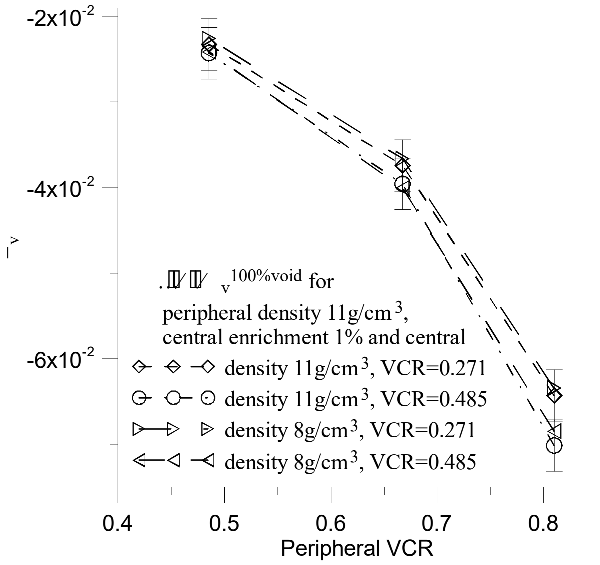

Figure 22. Comparison α_v(VCR) functions for different values of fuel density and VCR parameters in the central core region.

The fuel density of central core region does not influence on the α_v (Figure 22). However, the central VCR has a weak influence on α_v. Strictly speaking, a decrease in the central VCR parameter of about 44 % induce an increase in the relative value of α_v of approximately 4-9%. The decrease value of the central VCR parameter decreases absorption component of α_v, which should lead to a decrease in α_v. The slight increase in α_v suggests that the absolute value of leakage component of α_v simultaneously decrease. In other words, from point of view of α_v , there is no need to employ a very low values of central VCR in the Hybrid reactor. One can employ them to increase average neutron energy.

7. Calculation Uncertainties

This section presents the values of the uncertainties for both the α_v (Eq.(2)) and the difference in reactivity defined by Δρ (Eq.(3)).

The Δρ and α_v are functions of initial value of k_{eff} (Figure 23). Calculated uncertainties of α_v and Δρ include both the systematic error due to the uncertainty of the initial value of k_{eff} and standard deviation resulting from the statistical errors of initial values of k_{eff} (Figure 23).

Maximal value of the systematic error (MSE) in the range Δk_{eff}=0.02 is calculated using corresponding fitting function of $\alpha v(k_{eff})$ and $\Delta\rho(k_{eff})$ for Na-coolled U-cycle and Pb-cooled Th-cycle respectively.

The standard deviation of reactivity σ(ρ) was determined using standard deviation σ_{keff} of k_{eff} and the following formula:

$$\sigma(\rho) = \frac{\partial \rho}{\partial k_{eff}} \sigma_{keff} = \sigma_{keff} / k_{eff}^2 \tag{7}$$

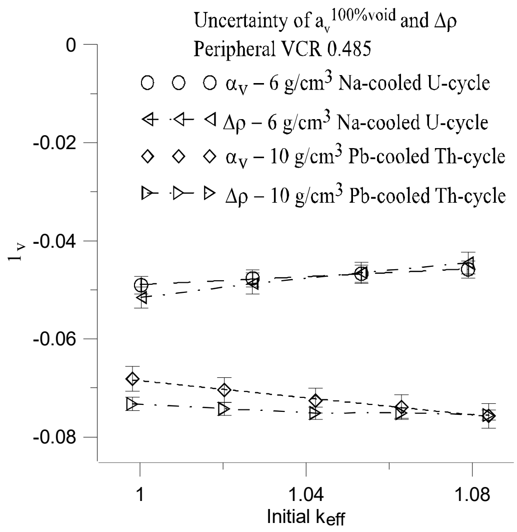

Figure 23. The total uncertainties of α_v and Δρ as a function of initial value of k_{eff} for U-cycle and Th cycle.

The standard deviation of Δρ is calculated using the following formula:

$$\sigma(\Delta\rho) = \sqrt{\sigma(\Delta\rho)^2_{norm} + \sigma(\Delta\rho)^2_{void}} \tag{8}$$

The standard deviation of α_v is calculated from the following equation:

$$\sigma(\alpha_v) = \sqrt{(\frac{\partial\alpha_v}{\partial k_{norm}}\sigma(k_{norm}))^2 + (\frac{\partial\alpha_v}{\partial k_{void}}\sigma(k_{void}))^2}. \tag{9}$$

Total uncertainties of α_v and Δρ for 100% void are equal to sum of standard deviation and MSE. These are denoted as $\Delta^{Tot}(\alpha_v)$ and $\Delta^{Tot}(\Delta\rho)$, respectively (Table 7).

The total uncertainties are significantly less than absolute value of α_v and Δρ (Figures 14–16 and 21).

Table 7. The uncertainties of α_v and Δρ for U-cycle and Th-cycle.

	$\sigma(\alpha_v)$	MSE(α_v)	$\Delta^{Tot}(\alpha_v)$	σ(Δρ)	MSE(Δρ)	Δ^{Tot}(Δρ)
Na-cooled U-cycle	0.0008	0.001	0.0018	0.0002	0.0018	0.002
Pb-cooled Th-cycle	0.0008	0.0017	0.0025	0.0008	0.001	0.0018

Total uncertainty Δ^{Tot}_{pert} of $\alpha_v(void)$ function (Figure 21) is determined by perturbation effective multiplicity factor k^{pert}_{eff} and its standard deviation $\sigma(k^{pert}_{eff})$ and MSE. The perturbation error was calculated for 10% of average coolant density using PERT command of MCP6.2 code. Maximal value of Δ^{Tot}_{pert} achieves 0.004 for void=20%.

8. FBR- IME Reactor Versus Hybrid Model

The main aim of the section is compare the difference in reactivity Δρ of FBR-IME [12,13] and Hybrid reactors for low value of average void. The Hybrid reactor is based on VCR parameter of 0.485 and fuel density of 11g/cm3 and central fuel enrichment of 1%. In addition, the Hybrid-Inverse reactor is presented. The geometry of this model is identical to that of the Hybrid reactor, but assemblies with fuel enrichment of 1% are placed in peripheral region, while assemblies with high fuel enrichment are positioned in central part of reactor (likely as in FBR_IME, (see Table 8).

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The Δρ for Hybrid reactors are calculated for two different values of ρ. The difference between them are less then correspondent value of standard deviation (Figure 24 and Table 8).

The low value of void results low value of difference in reactivity Δρ and requires high value of statistics. For this reason measurement of Δρ is a difficult task. The standard deviations of Δρ for voids of 2.93% and 5.87% for FBR_IME are significantly greater than corresponding values of Δρ [Table 8, Figure 24] and also exceed the corresponding values of Hybrid-Inverse reactor. However, the results of Δρ are likely applicable to both FBR-IME and Hybrid-Inverse. Despite the many differences between FBR-IME and Hybrid-Inverse reactors the results are practically the same in the range of experimental and calculation errors. Strictly speaking, the differences between Δρ for void of 2.93% and 5.87 % are approximately to 0.0005 and 0.0014, respectively. These differences are both less than corresponding standard deviation of FBR-IME. Whereas, this difference for void of 0.98% is approximately 0.0024 which is less than sum of corresponding deviation of FBR-IME and Hybrid-Inverse.

These reactors have a low value of fuel enrichment in the peripheral region (blanket/fertile) (Table 8, [12,13]).

The Δρ(void) is a decreasing function of void for both the FBR-IME and Hybrid reactors within the range of (0.98%, 5.87%) void. However, the FBR-IME exhibits a positive, whereas Hybrid reactor a negative value of Δρ. Special arrangement of high enrichment MOX fuel assemblies in the central part of the FBR-IME is insufficient to achieve a negative value of Δϱ.

The Hybrid model results a negative value of Δρ for all cases. In this scenario the average coolant density was modified in total peripheral region of reactor, which includes the peripheral assemblies and thin layer of coolant between the assemblies and the reactor barrel.

It is important to note, that blanket/fertile assemblies placed in the center part of the reactor significantly decrease Δρ (Figure 24, Table 8), while those placed in peripheral part acts as reflector and increase Δϱ (see Sec.4).

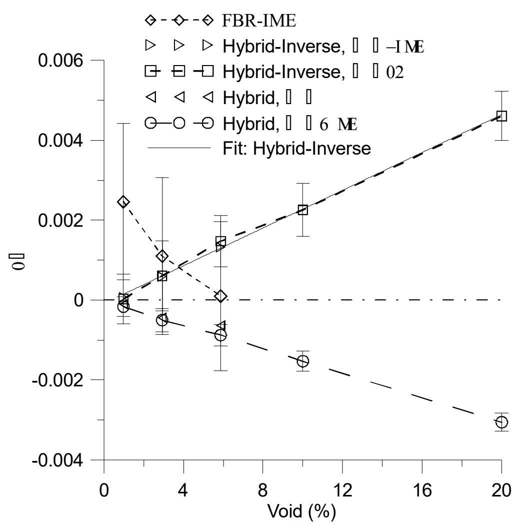

Figure 24. The difference Δϱ for FBR-IME, Hybrid and Hybrid-Inverse reactors.

Table 8. Comparison base parameters of FBR-IME, Hybrid and Hybrid-Inverse reactors.

Reactor/Void [%]	ρ	σ(ρ)	Δρ	σ(Δρ)	Central enrich.[%]	Peripheral enrich.[%]
FBR-IME [12,13]	0.12209	0.001464			25-42	0.71
FBR IME / 0.98	0.12455	0.001303	0.00246	0.00196	25-42	0.71
FBR IME / 2.93	0.12319	0.001307	0.00110	0.001963	25-42	0.71
FBR IME / 5.87	0.12219	0.001156	0.00010	0.001866	25-42	0.71

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Hybrid	0.068285	0.000182			1	14
	0.180811	0.000087			1	19
Hybrid / 0.98	0.06812	0.000165	-0.00016	0.000246	1	14
	0.18070	0.000094	-0.00011	0.000128	1	19
Hybrid /2.93	0.06778	0.0002	-0.0005	0.000289	1	14
	0.18037	0.000087	-0.00044	0.000123	1	19
Hybrid / 5.87	0.06741	0.0002	-0.00088	0.000271	1	14
	0.18018	0.000094	-0.00063	0.000128	1	19
Hybrid / 10.0	0.06676	0.000183	-0.00153	0.000258	1	14
Hybrid /20.0	0.05412	0.000143	-0.00327	0.000232	1	14
Hybrid-Inverse	0.03978	0.000175			8	1
	0.09558	0.000204			9	1
Hybrid -Inverse/0.98	0.04010	0.000599	0.000030	0.000624	8	1
	0.09565	0.000204	0.000074	0.000289	9	1
Hybrid Inverse/2.93	0.04126	0.000626	0.000608	0.000880	8	1
	0.09614	0.000204	0.000589	0.000289	9	1
Hybrid Inverse/5.87	0.04204	0.000616	0.00147	0.000646	8	1
	0.09690	0.000204	0.00132	0.000289	9	1
Hybrid- Inverse/10.0	0.04439	0.000633	0.002254	0.000662	8	1
Hybrid-Inverse/20.0	0.03983	0.000584	0.004606	0.000616	8	1

The $\sigma(\Delta\rho)$ is calulated using Eq.(8).

The standard deviation $\sigma(\alpha_v)$ of $\alpha_v = \frac{\Delta\rho}{\Delta v}$ should be determined from the following formula:

$$\sigma(\alpha_v) = \frac{\partial\alpha_v}{\partial\Delta\rho}\sigma(\Delta\rho) + \frac{\partial\alpha_v}{\partial\Delta v}\sigma(\Delta v) \tag{10}$$

If we assume that $\sigma(\Delta v)$=0 that $\sigma(\alpha_v) = \sigma(\Delta\rho)/\Delta v$

Using above equations one can easy show that $\sigma(\alpha_v)/\alpha_v = \sigma(\Delta\rho)/\Delta\rho$. It means that relative values $\sigma(\alpha_v)$ of α_v and $\sigma(\Delta\rho)$ of $\Delta\rho$ are the same. In real reactor the uncertainty of Δv can be significantly greater than 0. This implies, that the total relative uncertainty of α_v can be considerably greater than total uncertainty of $\Delta\rho$. For this reasons we compare $\Delta\rho$ in this section.

9. Discussion

The choice of Hybrid reactor geometry was made due to it has large core without both a reflector and an upper sodium plenum (compare Ref.[27]). The Hybrid reactor has all features of fast reactor with negative αv (Sec.6 and 8). Its core has relatively simple geometry and special arrangement of fuel assemblies optimized for leakage neutron flux during void conditions.

The purpose of this section is to present the main differences between a fast reactor core and a PWR core, as well as to explain their influence on the value of α_v.

-On top of that, the active fuel zone in SFRs and LFRs should be only about 1 meter due to significant concerns about the core void reactivity, which is much shorter than in the conventional PWRs.

The high value of active zone is intentional and is chosen to exhibit that a negative value can still be achieved even with an elevated core height.

Reducing the height of the active fuel zone to about 1 m is advantageous from the perspective of α_v and can be applied in Hybrid concept, as it increases the neutron leakage fraction and decreases α_v .

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-In the case of sodium-cooled fast reactor (SFR), the fuel lattice should be very tight to minimize the void reactivity and all SFR designs are based on the hexagonal lattice.

The neutron flux spectrum and average value of neutron energy do not depend on the type of lattice. Instead, they are influenced mainly by the VCR or VCFR parameter. The Hybrid reactor is designed based on the practical value of these parameters (i.e. central region VCR=0.485 or 0.271 (Figure 22) and average value of neutron energy is being approximately 0.5MeV. The square and hexagonal fuel lattices are described as options of fast reactor in Reference [7]. In other words, the hexagonal lattice can be used in Hybrid reactor without changing α_v. The minimal value of VCR parameters for FBR-IME and Hybrid reactor are equal to 0.374 and 0.271, respectively.

-In addition, there should be thick steel reflector in both radial and bottom sides of the core.

Excessive core height affects α_v in a manner similar to that of a thick bottom and top reflector. This means that the thick bottom reflector is included in the calculation results of Hybrid reactor. However, the thick radial side reflector presents challenge that needs to be addressed. This reflector should be removed. Additionally, the blanket/fertile should be positioned at the center of the core.

-Actually, there are significant differences in the core configurations between fast reactors and PWRs.

While there are many technical distinctions between the core of fast reactors and PWRs, the neutron flux spectrum and neutron physics in the Hybrid reactor are similar to those in fast reactors. Many technical concepts from fast reactors such as bending fuel assembly or upper gas plenum can be applied in concept of Hybrid reactor in construction future fast reactors with negative value of α_v.

10. Conclusions

The Hybrid reactor concept offers the very effective passive and promising method for achieving a negative α_v value for a wide range of basic reactor parameters (Sec.6).

The fuel density and VCR parameter are important reactor parameters, reduces value of α_v (Sec.6). Decrease average fuel density and increase VCR parameter in peripheral fuel assemblies induce decrease α_v. (Sec.6, Figures 14 and 15).

The $VCR_0^{100\%void}$ parameter is a very convenient parameter for determining the reduction value of α_v. Reducing this parameter is crucial for decreasing the value of α_v. Additionally, this parameter can be used to compare different method of reducing α_v. To achieve a negative α_v value, the reactor must have a VCR value exceeds $VCR_0^{100\%void}$ (Sec.4).

Particularly noteworthy is low value of α_v for Pb-cooled Th-cycle Hybrid reactor (Sec.6).

Herein are presented calculated results for liquid Pb and Na cooled reactor, only. However, the Hybrid concept can be useful for obtaining the negative value of α_v in reactors cooled by other liquids.

Abbreviations

The following abbreviations are used in this manuscript:

VCR	Volume fraction of coolant in the cell
VCFR	Volume ratio of coolant and fuel in the cell
FBR-IME	Brazilian Fast Breeder Reactor
α_V	Void reactivity coefficient
β_V	Loss of neutrons function
ANE	Average neutron energy in the fuel rods
ANFE	Average neutron energy causing fission reaction
ρ	Reactivity
$\Delta\rho$	Change in reactivity
CVF	Low void worth core
k_{norm}	effective multiplication factor at normally work
k_{void}	effective multiplication factor at voided reactor core
MSE	Maximal value of the systematic error
σ	Standard deviation

Δ^{Tot}	Total uncertainties
SFR	Sodium Fast Reactor
FBR	Fast Breeder Reactor
LFR	Lead Fast Reactor

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