Can enhanced feedback effects and improved breeding coincide in a metal fueled, sodium cooled fast reactor?

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**Abstract**

The sensitivity of operational and safety parameters on different strategies for the improvement of in core breeding on fuel assembly level are investigated using the HELIOS 2.1 code. The operational characteristics is analyzed regarding criticality, breeding, pin power and burnup distribution. As additional key parameters, the conservation of the safety related feedback effects of the assembly are examined. It is demonstrated that the insertion of 1/3 of fertile fuel rods into the fuel assembly, while the overall Pu content of the assembly is kept constant, can improve the breeding of fresh Plutonium. A second proposal is the reduction of the initial Pu content of the assembly which is compensated by eliminating one ring of the fertile blanket around the core. This method proofs to be very efficient to improve the in-core breeding. The consequences on the fuel assembly multiplication factor, the fissile material content, and the pin wise power as well as burnup distribution is analyzed. Additionally, the effect of fine distributed material on breeding as well as on the safety related feedback effects is investigated for both proposals. A clear enhancement of the feedback effects is proven.

Keyword: nuclear reactors, fast reactors, plutonium breeding, enhanced feedback effects, moderating material

**Introduction**

Form technological point of view, fast reactors (FRs) are the future of nuclear reactor technology, since this kind of reactor is the backbone of future sustainable reactor development [1], as defined in the Generation IV International Forum [2]. In addition, nuclear energy can provide an excellent carbon free energy source. From reactor physics point of view, fast reactors can provide a wide range of operational modes between a fast breeder reactor and a fast burner reactor for waste transmutation. The essential point for this flexibility is given in the neutron balance which is the driver, not only for the breeding but also for the safety related sodium void effect. First attempts for the optimization of fast breeder core configurations using computational methods have already been performed in the begin of the 70ies [6]. Important works on the core optimization regarding breeding and sodium void effect have been provided in a special meeting of the British nuclear energy society on optimization of sodium cooled fast reactors regarding the core optimization on heterogeneous level [3], regarding the correlation between breeding and sodium void effect [5], and regarding inherently safe reactor designs [4]. Later on a broad study on the optimization of liquid metal fast breeder reactors (LMFBR) cores with a detailed investigation of the effect of different geometric arrangements on the level of the core design using fuel and blanket assemblies in 14 different arrangements [7] has been developed. In the 90ies new studies have been given on the heterogeneous core optimization with regard to reduced sodium void [10]. Another broad overview has been published on different methods and consequences of heterogeneous core arrangements for the reduction of the sodium void worth in small sodium cooled fast reactor (SFR) cores [9]. Besides the heterogeneous arrangement of the core the traditional methods for the reduction of the sodium void effect based on increasing the neutron leakage play an important role in the safety of fast reactors. The desired high neutron leakage is achieved by the so called ‘pancake’ core design – a big core diameter (~ 5 meters) in combination with a very small core height (≤ 1 meter) and by the replacement of the upper reflector of a fast breeder reactor core with a sodium plenum.

Complementary to these global acting methods which are simulated by full core calculations, the change of the material composition of core itself has been proposed early to manipulate the neutron spectrum. The first test has already published for the use of zirconium hydride (ZrH) in mixed oxide (MOX) fuelled, but steam cooled fast reactors [16]. Later on the use of moderating material to improve the fuel temperature coefficient and the sodium void reactivity has been discussed and investigated in detail for SFR with metallic fuel [12]. This publication is based on the earlier proposals of the insertion of zirconium-hydride pins in reactors with metallic fuel which has been investigated before in several publications [13, 14, 15]. The use of moderating materials has been developed in a next step to the use of fine distributed moderating material for enhancement of feedback effects [21, 20]. It has been demonstrated that this method has no significant impact on the power and burnup distribution [19] and that the use of fine distributed moderating material offers new possibilities for the optimization of core designs for transmutation [18]. Finally, a new and more promising material has been proposed to improve the thermal stability of the moderating component to temperatures above one to be expected in operational transients and accidents [17].

The experience of the investigation of fine distributed materials has shown that new possibilities can open new ways. These new possibilities have come up due to the rapid development of the spectral codes for light water reactor (LWR) analysis into the application in 2D in the 90ies. These codes solve the integral transport equation (collision probabilities or method of characteristics) in two dimensions on unstructured mesh. The codes have been developed in the late 90ies for light water reactor (LWR) technology, like HELIOS [22] or APOLLO [23]. The codes are used for multi-group fuel assembly calculations as basis for the cross section preparation for nodal full core calculations. These codes offer the chance to investigate fuel assemblies in full detail including multi group visualization of integral and resolved neutron spectrum including comparison with the used cross section set. Recently, the HELIOS code has got a major update including a significantly improved cross section master library with the release of HELIOS 2 [23]. The improved geometric possibilities of the lattice codes will be used now to investigate the effect of a heterogeneous arrangement of fissile and fertile material inside the fuel assemblies to answer the question:

Can enhanced feedback effects and improved breeding coincide in a metal fueled, sodium cooled fast reactor?

This question will be answered by a systematic study to compare homogeneous and heterogeneous arrangements of a reference fuel assembly to investigate the performance in criticality and breeding as well as the influence of the arrangement on the safety related feedback effects.

**Code and Modelling**

The study is based on a reference case derived from the fast breeder reactor core design with metal fuel, developed at the Indira Gandhi Centre for Atomic Research (IGCAR) [25, 26, 27]. The main data is given in Table 1. The fuel pins as well as the fertile pins are sodium bonded. The data for the required sodium properties is taken from Waltar, Reynolds: Fast Breeder Reactors [28]. The used plutonium vector for the fuel is given Table 2, no Pu-238 and Am-241 have been considered. The plutonium is mixed with 10.95% weight content in depleted uranium with 0.3% U-235 and 6% zirconium to metal alloy fuel. The fertile pins contain only depleted uranium with 6% zirconium.

Table 1: Fuel assembly data for the 1000 MWe metal fueled fast reactor design of the IGCAR

|  |  |  |
| --- | --- | --- |
| pin diameter | 8 | mm |
| fuel pellet diameter | 6.01 | mm |
| fertile pellet diameter | 6.398 | mm |
| cladding thickness | 0.53 | mm |
| pins per subassembly | 271 |  |
| pin pitch | 9.44 | mm |
| can wall thickness | 3.3 | mm |
| FA pitch | 168 | mm |
| av. fuel temperature | 1103 | K |
| clad temperature | 923 | K |
| sodium temperature | 773 | K |
| Zr content | 6 | % |
| Pu content | 10.95 | % |
| U-235 content | 0.30 | % |
| power density | 450 | w/cm |
| fuel density | 16.6 | g/cm³ |
| fuel assemblies in core | 271 |  |
| Fertile assemblies in core | 126 |  |

Table 2: Plutonium vector for the reference case

|  |  |
| --- | --- |
| Pu-239 | 68.79% |
| Pu-240 | 24.60% |
| Pu-241 | 5.26% |
| Pu-242 | 1.35% |

Sodium bond

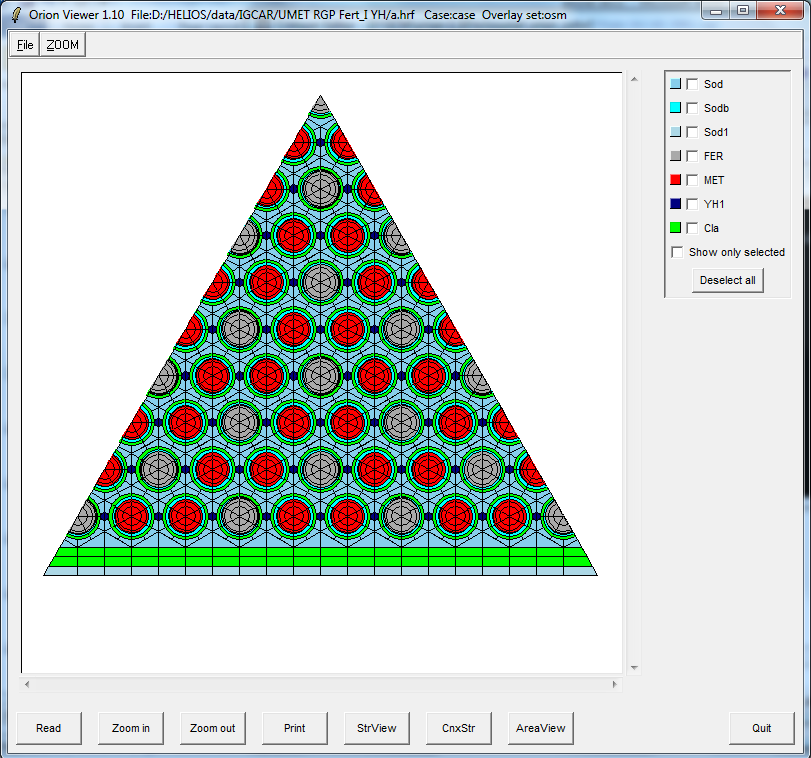
Sodium

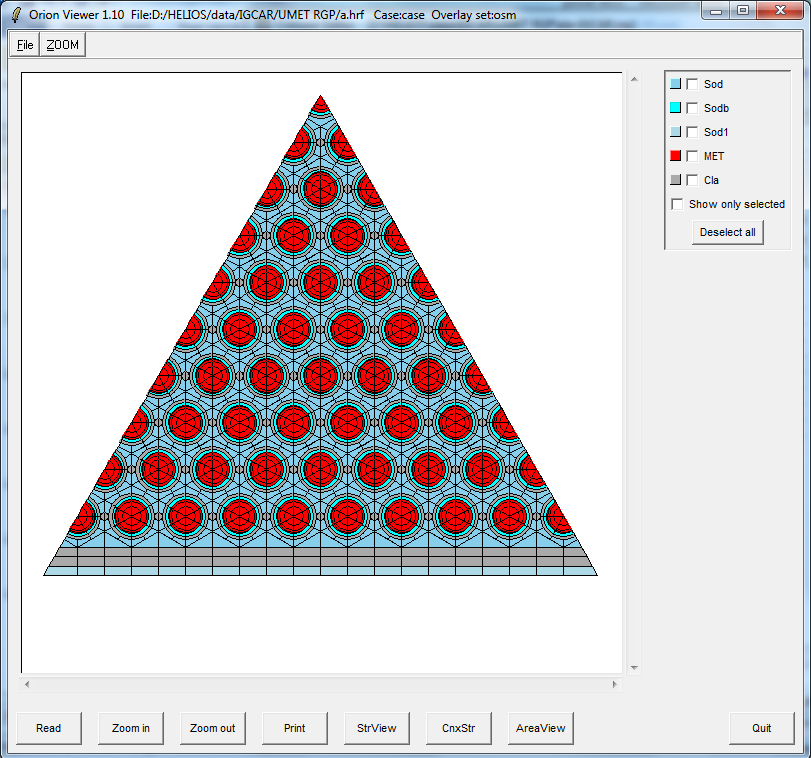
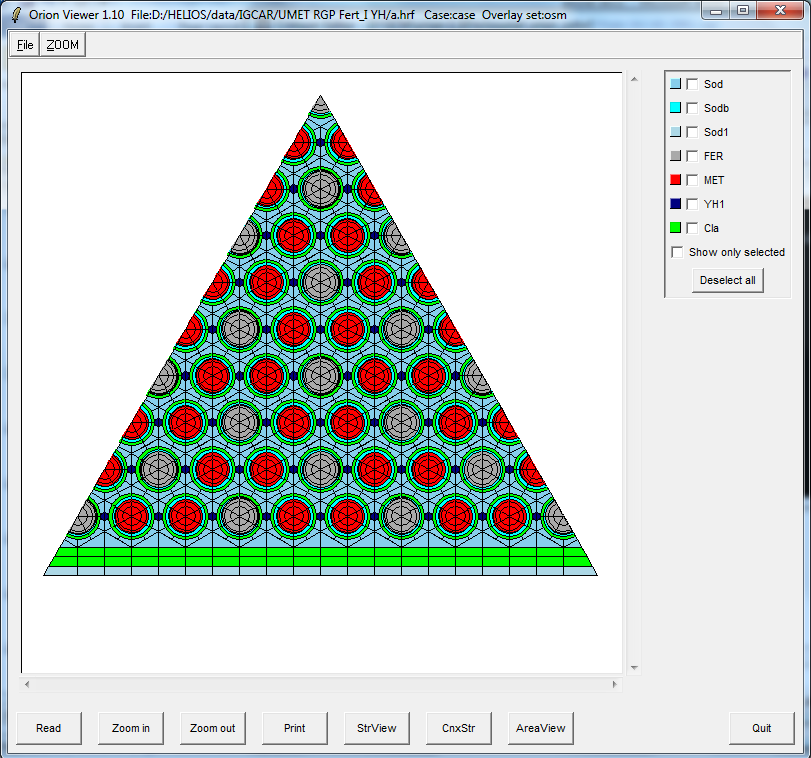
Fertile

Fuel

YH moderator

Cladding



Sodium bond

Sodium

Fuel

Cladding

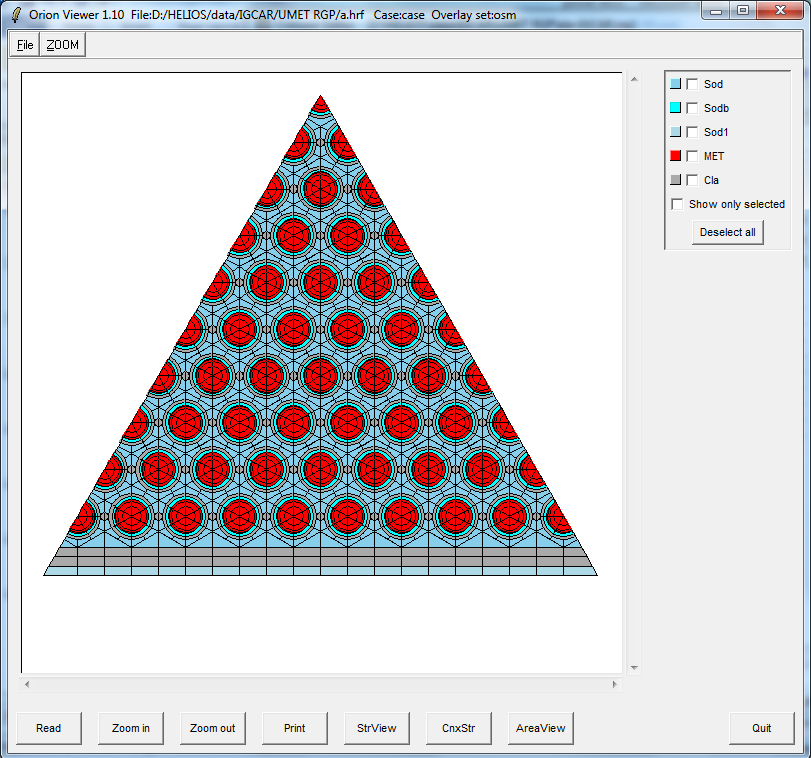


Figure 1: 1/6 of the reference fuel assembly geometry (left) and the heterogeneous fuel assembly geometry (right) corresponding to the IGCAR fuel assembly design for metal fuel

Cladding, wire wrapper and can wall are made from stainless steel 304 along the HELIOS definition. The Sodium density is calculated to 0.833 g/cm³ along the formula for liquid saturated sodium at 773 K given in Waltar, Reynolds [28].

The modeled geometric arrangement of the reference system with 10 rings, following the IGCAR design, is shown in Figure 1 for a 1/6 part of one fuel assembly. The specific power is set to 98.68 W/g corresponding to the maximum linear heat rate of 450 W/cm. The HELIOS 2.1 internal 177 group library is used for the calculations [29].

The applied HELIOS code package is mostly developed for light water reactor applications, but some features for fast reactor applications have already been implemented in earlier versions [30]. A cross comparison with MCNP for the initial value of fuel temperature and moderator effect on kinf was performed on a simplified basis at the beginning of the studies on fine distributed moderating material. The comparison has confirmed the very significant results caused by the insertion of moderating material on the feedback effects [21, 20]. In further comparisons with the SERPENT Monte-Carlo based lattice code with burnup capabilities good agreement was found for the burnup of actinides and minor actinides in fast reactor configurations for the HELIOS 2 code [31]. This good agreement in comparison with continuous energy methods gives confidence in the applicability of the code HELIOS, the methods applied inside the code as well as the by Studsvik Scandpower supplied master library, and thus the results for steady state as well as for the burnup calculations. Finally, it has to be kept in mind that the analysis is based on the changes caused by slight material changes and re-arrangements. The final absolute values are not the major information deduced, but the relative differences between the different calculated configurations based on identical modelling and input data.

The following cases have been calculated for the comparison:

* RGP – reference case 10.95 % Pu content geometry as given above data from doc 92 [26] see Figure 1, left, core with 271 fuel assemblies and 126 fertile assemblies
* RGP YH – reference case 10.95 % Pu content like RGP but with yttrium hydride (YH) in the wire wrapper
* Fert – heterogeneous geometry, see Figure 1, right, with corrected Pu content 16.5 % using 180 fuel pins instead of 271 and 91 fertile pins with fertile pin geometry, core with 271 fuel assemblies and 126 fertile assemblies
* Fert YH – like Fert but with YH in the wire wrapper
* Fert 1 – like Fert, but with corrected Pu content 13.5 % for one additional ring in the full core this leads to 331 instead of 271 fuel assemblies and 66 fertile assemblies
* Fert 1 YH – like Fert 1 but with YH in the wire wrapper
* Fert 2 – like Fert, but with corrected Pu content 11.25 % for one additional ring in the full core which leads to 397 instead of 331 fuel assemblies and no fertile assemblies
* Fert 2 YH – like Fert 2 but with YH in the wire wrapper
* RGP1011 – homogeneous case but with corrected Pu content 8.96 % for one additional ring in the full core this leads to 331 instead of 271 fuel assemblies and 66 fertile assemblies
* RGP1011YH – like RGP1011 but with YH in the wire wrapper

**Results and Discussion**

**Operational Parameters**

All cases described in the last paragraph have been calculated with the HELIOS 2.1 code version and evaluated until a postulated end of life (EOL) burnup of 200 000 MWd/tHM in 25 calculation steps. The feedback effects have been evaluated at begin of life (BOL), 50 000, 100 000, 150 000, and 200 000 MWd/tHM using the TREE option in HELIOS [30]. The insertion of the fine distributed moderating material has been accomplished by replacing the spacer wire made of stainless steel by a spacer wire consisting of Yttrium-mono-hydride (YH), see Figure 1.

In the first step of the study, the criticality of the fuel assembly in the infinite medium approximation is evaluated. Some important general trends can be identified in Figure 2. On the one hand, the insertion of the heterogeneity decreases the kinf at begin of life (BOL) slightly (compare the RGP case and the Fert case as well as the RGP1011 case and the Fert 1case). On the other hand, the heterogeneous arrangement reduces the reactivity swing. The lines Fert (marked with the blue triangles) as well as the one of Fert 1 (green diamonds) cross the lines for the homogeneous cases between 50 000 MWd/tHM and 100 000 MWd/tHM. The kinf at end of life (EOL) is in both heterogeneous cases higher. In the comparison of the RGP case to the Fert case the kinf at BOL is ~140 pcm lower and at EOL ~40 pcm higher. In the comparison of RGP 1011 with Fert 1 the values are ~-145 pcm and ~50pcm. Thus, the change in the Pu content in the assembly has no significant effect on the kinf change due to heterogeneity.

The reduced Pu content due to the postulated increase of the core size leads to a reduction of the kinf at BOL. The observation of the kinf over burnup shows a strong increase between BOL and 100 000 MWd/tHM followed by a moderate decrease in all cases with reduced Pu content. The kinf values at EOL come close for all cases with a comparably small difference of the EOL value dependent on the initial Pu content. The case with the adopted Pu content to a two rings lager core does not lead to a critical configuration at BOL, even in the infinite medium approximation. Thus this case will not be considered fur the further investigations. In the comparison of the RGP case with the RGP1011 case, the kinf is reduced by ~10000 pcm at BOL, and ~1000 pcm at EOL. The values for the heterogeneous case are in the same range.

The insertion of the moderating material decreases the kinf in all cases and the general trend is identical to the one of the corresponding case without moderating material. In the RGP reference case the kinf is decreased by ~2000 pcm at BOL and ~5000pcm at EOL. The introduction of heterogeneity decreases the penalty slightly. The reduction of the Pu content reduces the penalty at BOL and increases the penalty at EOL.

In general, the reduction of the burnup swing for the cases with reduced Pu content could offer a significantly longer cycle time for these kind of fuel assemblies.



Figure 2: kinf for the different cases over burnup

The fissile inventory given by the number density of the transuranium (TRU) isotopes and U-235 averaged over the whole fuel assembly increases for all investigated cases strongly in the first half of the observation period, see Figure 3. After 100 000 to 150 000 MWd/tHM appears a saturation effect. Some clear trends can be observed for the different cases. The application of the heterogeneous configuration leads to a clear increase in the fissile material inventory compare the RGP case (black squares), with the Fert case (blue triangles), as well as the RGP1011 case (dark red stars), with the Fert 1 case(green diamonds). This gain is mainly caused by the different geometry on the fertile rods. Due to the increased pellet diameter slightly more fertile material is inserted into the core. This increased amount of fertile material affects obviously the breeding performance of the fuel assembly.

The reduction of the Pu content due the postulated increase of the core size by one ring leads to a significantly higher production of fissile material in the fuel assembly as a function of burnup compare the RGP case (black squares) with the RGP1011 case (dark red stars), as well as the Fert case (blue triangles), with the Fert 1 case (green diamonds). The application of the stronger correction due to an increase of the core size by two rings would be even better with regard to the fissile material production, but a critical reactor core built with these fuel assemblies cannot be built due to the lack of criticality.

The insertion of the fine distributed material enhances the built up of fissile material in all cases. The gain at EOL is roughly half as much as the gain due to the application of the heterogeneous arrangement.

The clear increase of the content of fissile material compared to the initial amount indicates a strong improvement of the in-core breeding for the cases with decreased initial Pu content and an increased core size. On the one hand, this effect is highly desirable from the nuclear security point of view, since in fuel assemblies with strong in-core breeding the fresh plutonium is built in a matrix with the already existing Pu which leads to a Pu vector with a significant share of even Pu isotopes. This behavior is in strong contrast to the Pu breeding in pure fertile assemblies like it is used in heterogeneous core configurations. On the other hand, the improved in-core breeding gives hope for an increased core performance with increased cycle times, which has the potential to improve the economy of fast reactor operation significantly.



Figure 3: Fissile material inventory averaged over the fuel assembly as a function of burnup

The increase of the fissile inventory gives only a rough overview since all TRU isotopes are integrated. A more detailed view is given by the change of the Pu-239 content in the fuel assembly as a function of burnup (Figure 4). The content of the most important fissile isotope in fast reactors increases in the first operation phase up to 50 000 MWd/tHM and decreases at EOL in all observed cases. However, the maximum is shifted to higher burnups for the heterogeneous cases and even more pronounced for the cases with the lower absolute Pu content in the fuel assembly. The insertion of the fine distributed moderating material eliminates the gain of the heterogeneous arrangement in the Pu-239 breeding. These results indicate that the change in the neutron spectrum due to the insertion of the moderating material has a clear influence on the plutonium vector of the bred TRUs.



Figure 4: Absolute Pu-239 content averaged in the fuel assembly as a function of burnup

To discuss this effect of the plutonium vector an overview is given in Figure 5, which is split for the detailed analysis into Figure 6 to Figure 8. The general trend is for all different cases identical. The Pu vector contains ~70% Pu-239, ~25% Pu-240, ~5% Pu-241, ~1.5% Pu-242, and a very minor share of Pu-238. The Pu-239 content increases slightly in the begin of operation and decreases with increasing burnup after the initial phase until ~50 000 MWd/tHM while the Pu-240 increases with increasing burnup. The content of excellent fissile material (uneven Pu isotopes) in the Pu vector is lowest for the case with moderating material. However it should be kept in mind, that the absolute Pu-239 content is almost identical to the one in the reference case (see Figure 4). 

Figure 5: Overview of the Pu vector over burnup

The detailed view into the Pu-239 isotope in the Pu vector indicates that the separation of fissile and fertile material in the heterogeneous arrangement increases the Pu-239 share in the vector (see Figure 6). This effect can be observed in the comparison of the RGP case with the Fert case as well in the comparison of the RGP 1011 case with the Fert 1 case. Additionally, the reduction of the absolute initial Pu amount in the fuel assembly improves the plutonium quality as it has to be expected, see e. g. [28]. This can be observed in the comparison of the cases RGP with RGP 1011 and the cases Fert with Fert 1. The insertion of the fine distributed moderating material increases the amount of fissile material, but obviously on the cost of the Pu-239 share and thus on cost of the overall plutonium quality.



Figure 6: Comparison of the Pu-239 content in the Pu vector over burnup

The detailed view into the Pu-240 share in the plutonium vector confirms the perceptions gained in the analysis above. The reduced overall plutonium content in the assembly leads to a significant reduced Pu-240 production. The separation of the fissile and the fertile material in the heterogeneous arrangement leads to a comparable lower Pu-240 content, too, but the effect is smaller. Finally, the introduction of the moderating material leads to a clear increase of the Pu-240 production. This effect explains why the overall fissile material production is increased by the insertion of the moderating material while the Pu-239 content in the fuel assembly is almost not influenced by the insertion.



Figure 7: Comparison of the Pu-240 content in the Pu vector over burnup

The detailed view into the Pu-241share, Figure 8, in the plutonium vector is on the long term highest for the case with moderating material. The reduction of the initial Pu content in the assembly reduces the Pu-241 content. The separation of the fissile and the fertile material in the heterogeneous arrangement leads to a slightly lower Pu-241 content, too, but the effect is smaller. The Pu-242 production, Figure 9, is highest for the reference case. The production is slightly reduced in the heterogeneous cases and more reduced by the reduction of the overall Pu content in the fuel assembly. The Pu-242 content in the Fert 1 YH case is lower than in the RGP reference and in the Fert case. Thus, the breeding of higher plutonium in the case with moderating material is counteracted by the reduced Pu content in the assembly and the heterogeneity. There is a time delay for the breeding of higher isotopes in the case with moderating material. Thus, the main increase in the bred material is Pu-240 and Pu-241. In general, the insertion of the moderating material obviously leads to an increase mainly of the Pu-240 and Pu-241 production while the high initial Pu content tends to move the Pu vector to all higher isotopes. This observation coincides with the results in the literature, published for Mox fuel in the frame of the CAPRA program [32].

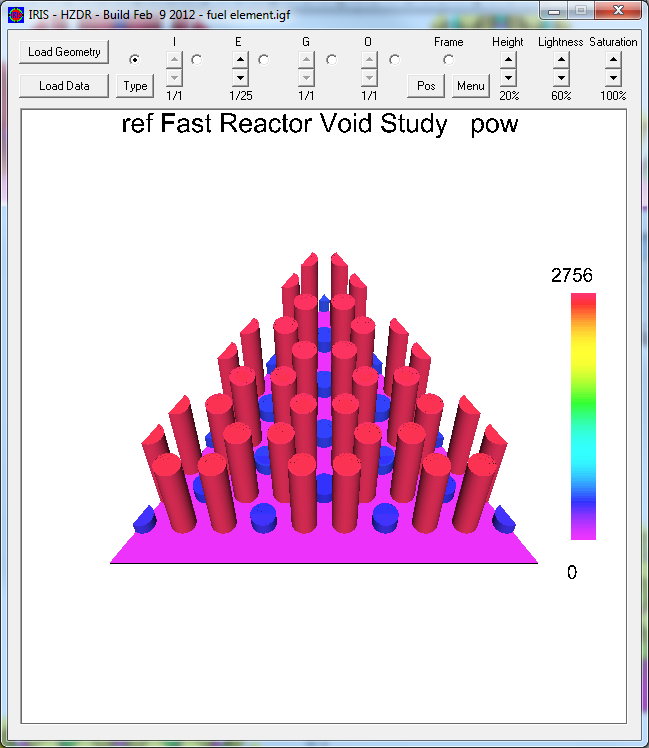
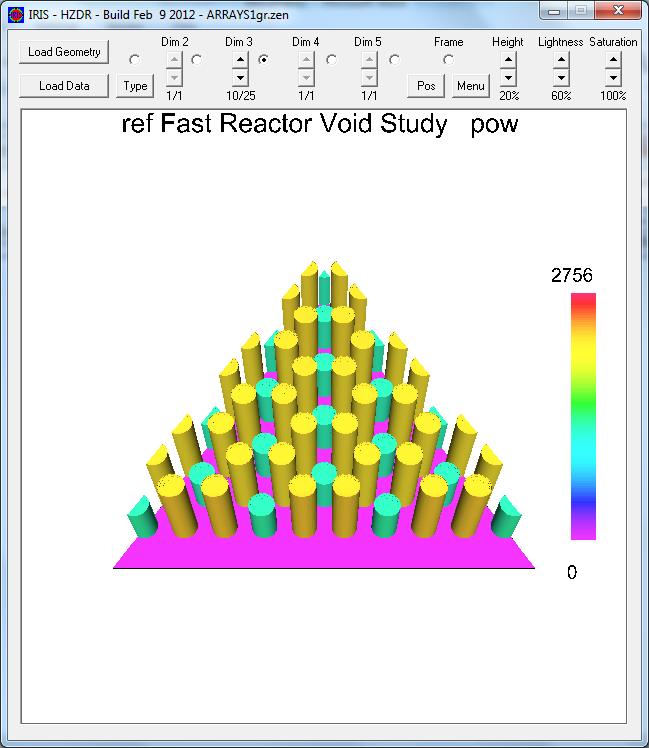
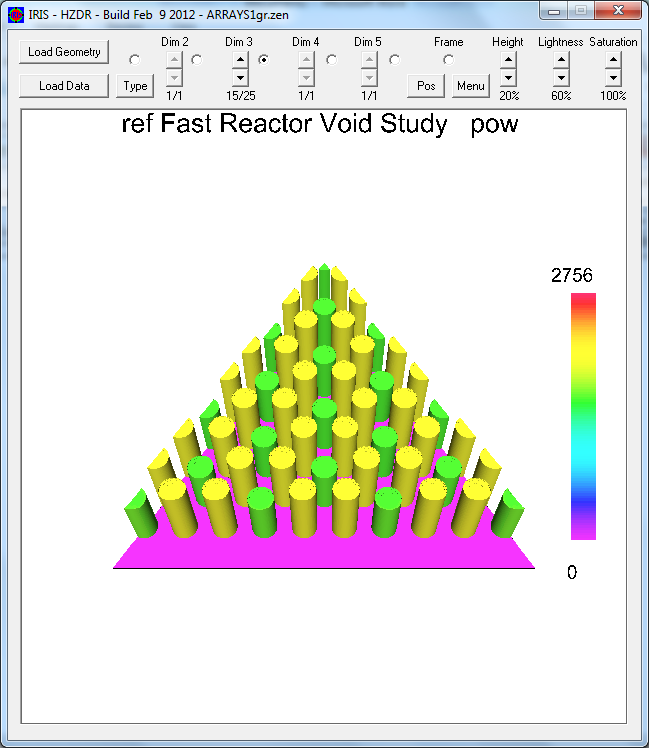


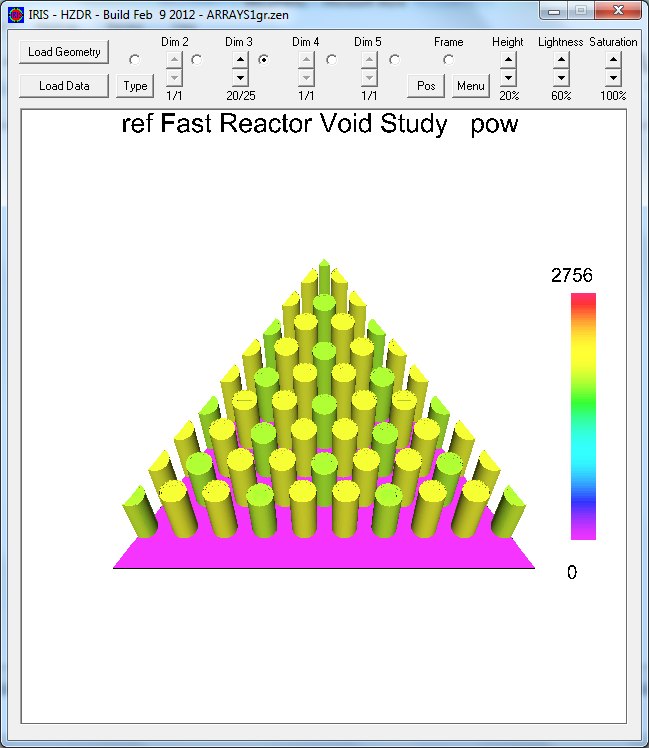
Figure 8: Comparison of the Pu-241 content in the Pu vector over burnup

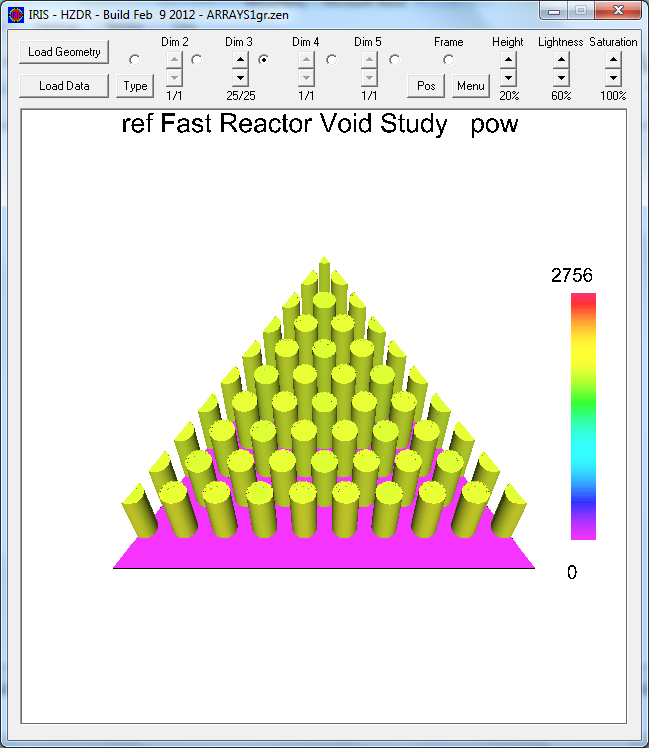


Figure 9: Comparison of the Pu-242 content in the Pu vector over burnup

The application of the heterogeneous arrangement causes a clear disturbance in the power distribution at BOL, especially. The power and burnup distribution is very flat in a typical homogeneous fast reactor assembly. The reason can be found in the very flat power distribution as well as in the comparably low interaction cross section for neutrons with energies characteristic for sodium cooled fast reactors. The pin power varies between the pin with the maximum and the pin with the minimum power by less than 2% over the whole burnup. The pin power distribution in the heterogeneous case Fert 1 YH shows strong differences in the power of the fuel and the fertile pins. To produce the target average power of 450 W/cm, given in the input, the maximal power at BOL has to be almost 650 W/cm. This value is strongly above the currently given limit of 450 W/cm which is almost kept in the homogeneous case. The power in the fertile pins is at BOL only ~90 W/cm. Due to the breeding process, the power distribution inside the fuel assembly becomes more and more flat. Already after 50000 MWd/tHM the maximum has reduced to 540 W/cm while the minimum has increased to more than 280 W/cm. At EOL the power difference between the fuel and the fertile pins is almost negligible. However, this power distribution at BOL requires a clear reduction of the fuel assembly power to keep the safety limits at BOL of the heterogeneous fuel assembly. Before thinking about the application of a heterogeneous assembly arrangement some important questions would have to be answered. Can the pin design of the fertile pins carried forward from the classical design taken from a fertile blanket arrangement to the heterogeneous design where the pins produce on the long term much more power compared to the traditional blanket design? Does the fission process induce too much swelling which would require a reduction of the fertile pin diameter which would eliminate the gain in breeding? Does the high pin power in the fuel rods cause increased swelling problems?



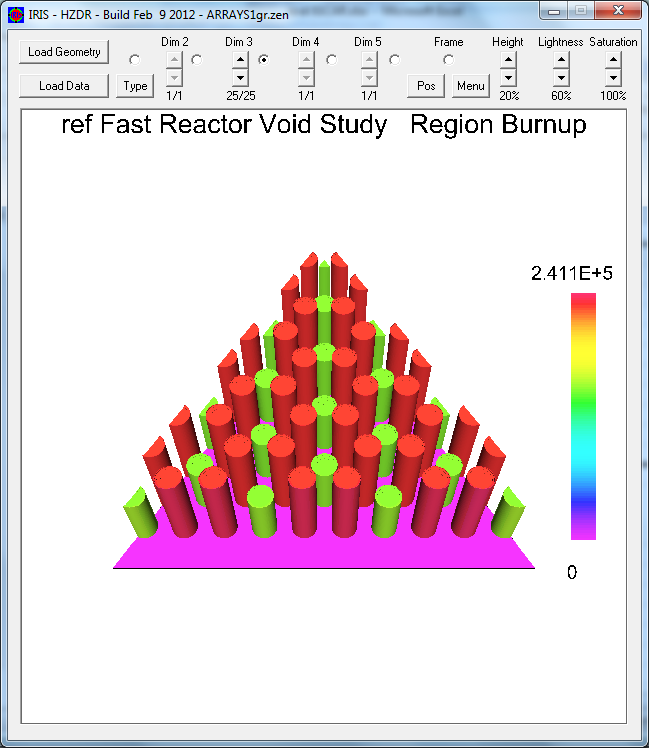
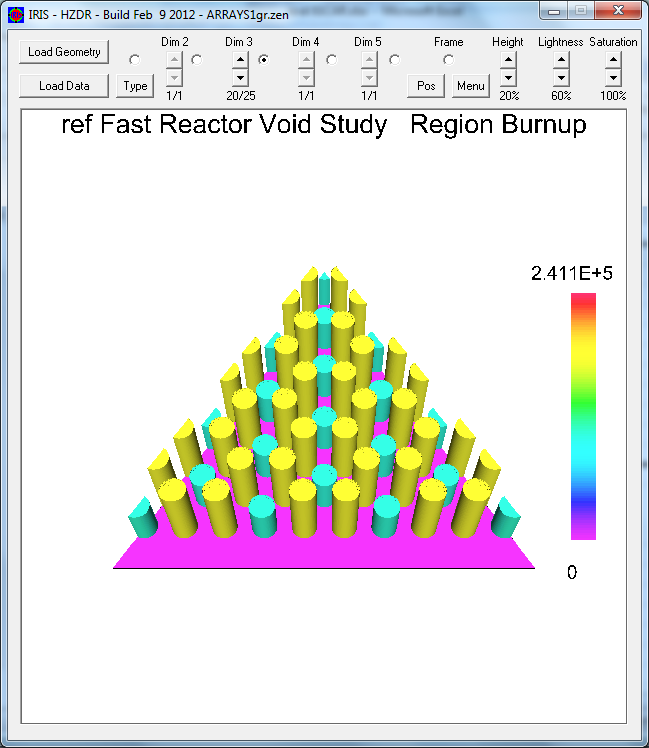
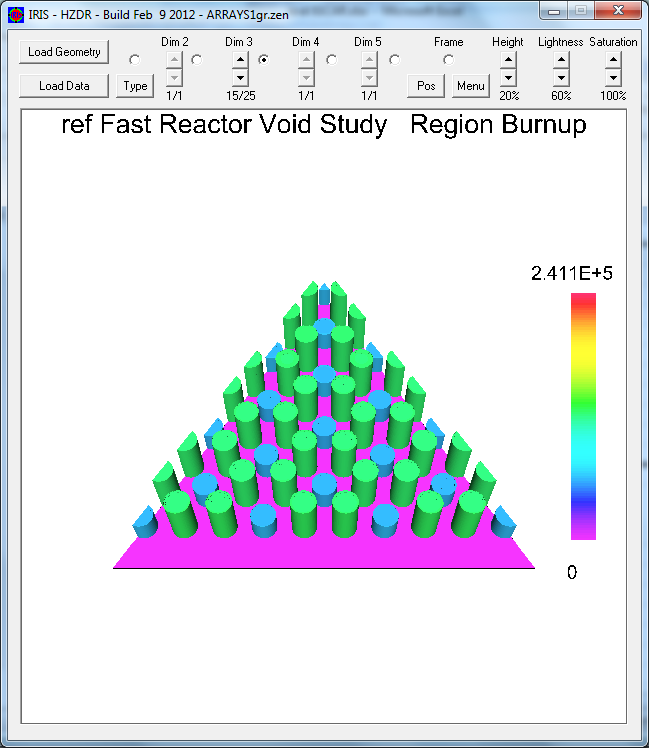
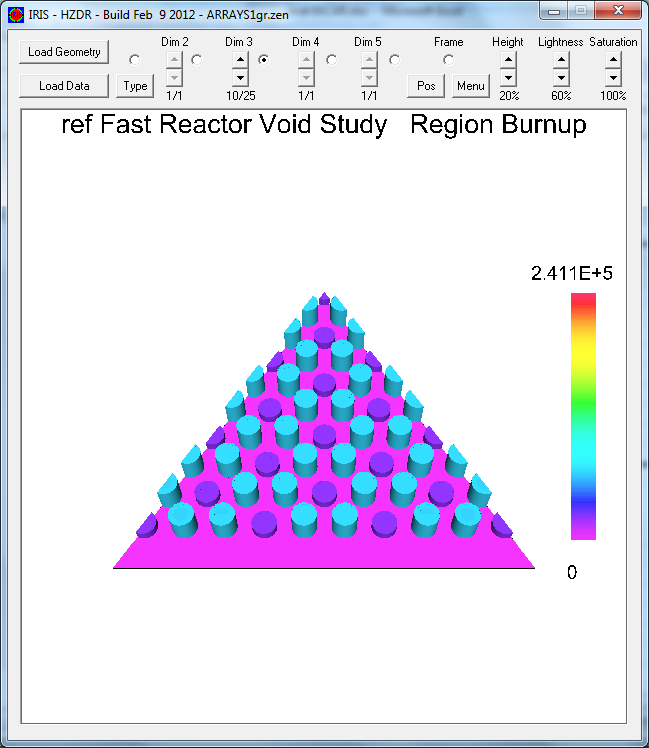


**650 W/cm**

**0 W/cm**

Figure 10: Power distribution in the fuel assembly at major burnup steps for the heterogeneous case Fert 1 YH (at average burnup of 0, 50000, 100000, 150000, 200000 MWd/tHM from top left to bottom right)

The burnup distribution (Figure 11) in the fuel assembly follows typically the power distribution. Consequently, the burnup distribution of the homogeneous case is as flat as the power distribution. The difference between the maximum and the minimum pin burnup is only 1.8% at maximum (after 1000 MWd/tHM) and reduces to 0.3% at EOL for the homogeneous references case RGP. The power imbalance in the heterogeneous case FERT 1 YH leads to a clear imbalance in the burnup distribution, too. At 50 000 MWd/tHM, the maximal burnt pin has a burnup more than 30% higher than the maximum burnt pin in the reference case. The minimal burnt fertile pin has at this point more than 50% less burnup than the minimal burnt pin in the reference case. The situation improves during the burnup, but there is still a clear imbalance visible at EOL. The maximum burnt pin faces a 15% higher burnup (229 200 MWd/tHM) than in the reference case and the minimum burnt fertile pin has 25% less burnup (149 200 MWd/tHM). The increased maximum burnup in the heterogeneous case requires definitely a reduction of the average target burnup of the heterogeneous fuel assembly or an increased target burnup which could probably have drawback on the pin design



**241 000 MWd/tHM**

**0 MWd/tHM**

Figure 11: Burnup distribution in the fuel assembly at major burnup steps for the heterogeneous case Fert 1 YH (at average burnup of 50000, 100000, 150000, 200000 MWd/tHM from top left to bottom right)

The comparison of the min. and the max. pin power (Figure 12) for the homogeneous and the heterogeneous fuel assemblies shows clearly the problem of the pin power distribution in the heterogeneous cases (Fert1) in contrast to the homogeneous cases (RGP). The difference between the min. and the max. pin power is almost not visible in the homogeneous RGP cases, the lines are almost identical for min. and max.. In contrast, in the heterogeneous Fert 1 cases appears a wide gap between the min. and the max. pin power. This gap closes during burnup slowly. At EOL both values have almost reached the same range. To investigate the effect of the requested reduced power, as described in the analysis of the power distribution, a new case red. Fert 1 is added. This case is operated at 67.4 W/g, or 307 W/cm to hold the limit of 450 W/cm at the pin with the maximum power. In this case the EOL power will be reduced to ~320 W/cm. It is important to note here that the burnup is applied for the x-axis. This is the reason why the curves for different power behave comparable. However, the changed fuel assembly power will significantly increase the cycle time of the fuel assembly which is required to achieve the identical target burnup. In contrast to the heterogeneity, the insertion of the moderating material has almost no effect on the min. and max. pin power.

Based on these results a discussion has to be taken up. Is the gain in breeding worth to face the challenges given due to the imbalanced pin-power and burnup distribution? Is it maybe more attractive to switch to the homogeneous arrangement with the lower initial Pu content and the increased core diameter?



Figure 12: Spread in the pin power for the reference case and the Fert 1 case

The more detailed analysis of the data is given based on tables. Following a general overview, and the comparison of the singular effects

* Homogeneous vs heterogeneous
* Without vs. with moderating material
* Compact core with high initial Pu content vs. larger core with lower initial Pu content

will be given.

The absolute and the relative fissile material amount in the fuel assembly increases compared to the initial amount in all investigated cases (Table 3). The reason is the relatively low plutonium content of only 10.95% in the reference fuel assembly which is used in the inner core of the IGCAR reference core design with metal fuel. All additionally investigated cases lead to improved breeding at all target burnup stages compared to the reference case. Already in the comparison of the raw numbers three different levels can be observed corresponding to the three different initial Pu loadings. For a more detailed analysis relative comparisons are given in the following tables. The relative evaluation of the increase of the fissile material amount shows the breeding behavior of each case with regard to different target burnups. The major gain in fissile material is achieved already after 100 000 MWd/tHM in the cases with comparably higher Pu content (RGP and Fert). After this time a part of the bred material is already eaten up to produce power in the reactor. In the cases with reduced initial Pu content in the fuel assembly (RGP 1011 and Fert 1) this break-even point is moved to 150 000 MWd/tHM. A further reduction of the initial Pu content (Fert 2) would shift the break-even point to even higher burnups. In the last row the required reduction of the operational power of the Fert 1 YH case is investigated. The power reduction by 31% (67.4 W/g instead of 98.7 W/g) leads to a small reduction of the gain in fissile material. However, the cycle time is increased by 44% to achieve the identical target burnup f 200 000 MWd/tHM. In addition in real operation, the target burnup would have to be lowered due to the imbalance of the burnup distribution by ~15 %, too. Combining these two changes leads to a real cycle time for the heterogeneous fuel assembly which will be ~30% longer with 31% lower power than for the reference fuel assembly.

Table 3: Absolute and relative increase in fissile material amount at different burnup stages compared to the initial fissile material amount of each case

|  |  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- | --- |
| Burnup [MWd/tHM] | absolute [particles/barn/cm] | | | relative [%] | | |
| 100 000 | 150 000 | 200 000 | 100 000 | 150 000 | 200 000 |
| RGP | 1.025E-04 | 9.474E-05 | 6.383E-05 | 7.0% | 6.4% | 4.3% |
| RGP YH | 1.128E-04 | 1.157E-04 | 9.469E-05 | 7.7% | 7.9% | 6.4% |
| Fert | 1.519E-04 | 1.513E-04 | 1.230E-04 | 10.3% | 10.3% | 8.4% |
| Fert YH | 1.646E-04 | 1.759E-04 | 1.583E-04 | 11.2% | 11.9% | 10.8% |
| RGP 1011 | 2.629E-04 | 2.846E-04 | 2.722E-04 | 21.7% | 23.5% | 22.5% |
| RGP 1011 YH | 2.7042E-04 | 3.0383E-04 | 3.0205E-04 | 22.30% | 25.10% | 25.00% |
| Fert 1 | 3.169E-04 | 3.446E-04 | 3.338E-04 | 26.1% | 28.4% | 27.5% |
| Fert 1 YH | 3.270E-04 | 3.676E-04 | 3.683E-04 | 27.0% | 30.3% | 30.4% |
| Fert 2 | 4.500E-04 | 4.961E-04 | 4.965E-04 | 44.3% | 48.8% | 48.9% |
| red. Fer 1 YH | 3.100E-04 | 3.488E-04 | 3.492E-04 | 25.6% | 28.8% | 28.8% |

For the detailed analysis of the changes in the built up of fissile material at different target burnups due to the different cases, the gain in fissile material of each case is compared to the reference case (Table 4). How much more breeding can be achieved when the fuel assembly and probably the core configuration is changed? As general result it can be observed that there is a clear gain compared to the reference case for all investigated cases in all target burnups. The strong relative increase with the achieved average target burnup is caused by the decrease of the breeding gain in the reference case already after 100 000 MWd/tHM target burnup. All other investigated cases achieve the maximum in the breeding gain later. The really strong relative increase, reflected by partly several hundred percent is a drawback of the very small growth of the fissile content in the reference fuel assembly which is foreseen to be operated in a heterogeneous core configuration with the core surrounded by breeding blankets. However, the results show that it is possible to enhance the in-core breeding significantly by different methods. The improved in-core breeding will result in a clear operational gain given by considerably longer cycle times and thus reduced out of operation times of the fast reactor due to fuel reloading. This is one of the major demands for increased availability and thus improved economy of fast reactor operation [1]. The increased cycle time in a fast reactor is of strong interest due to the complicated re-fuelling process requiring lengthy fuel assembly maneuvering which is much more complex than in light water reactors.

Table 4: Absolute and relative gain in fissile material amount compared to reference case at different burnup stages (absolute gain = case x – RGP, relative = case x/RGP-1)

|  |  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- | --- |
| Burnup [MWd/tHM] | absolute [particles/barn/cm] | | | relative [%] | | |
| 100 000 | 150 000 | 200 000 | 100 000 | 150 000 | 200 000 |
| RGP | 0 | 0 | 0 | 0% | 0% | 0% |
| RGP YH | 1.034E-05 | 2.099E-05 | 3.086E-05 | 9.6% | 23.0% | 49.8% |
| Fert | 4.943E-05 | 5.659E-05 | 5.914E-05 | 47.1% | 60.9% | 95.3% |
| Fert YH | 6.211E-05 | 8.116E-05 | 9.447E-05 | 60.0% | 85.9% | 151.2% |
| RGP 1011 | 1.604E-04 | 1.899E-04 | 2.084E-04 | 210.0% | 267.2% | 423.3% |
| RGP 1011 YH | 2.2780E-04 | 2.7636E-04 | 3.0599E-04 | 218.57% | 292.19% | 481.40% |
| Fert 1 | 2.144E-04 | 2.498E-04 | 2.700E-04 | 272.9% | 343.8% | 539.5% |
| Fert 1 YH | 2.245E-04 | 2.728E-04 | 3.044E-04 | 285.7% | 373.4% | 607.0% |
| Fert 2 | 3.475E-04 | 4.013E-04 | 4.327E-04 | 532.9% | 662.5% | 1037.2% |
| red. Fer 1 YH | 2.076E-04 | 2.541E-04 | 2.854E-04 | 265.7% | 350.0% | 569.8% |

In the following tables, the effect of the three invented changes (heterogeneity, moderating material, and reduced initial plutonium content) is separately analyzed in detail using the absolute and the relative gains in fissile material to get a deeper understanding how efficient each of the changes is. The analysis of the gain in fissile material due to the insertion of heterogeneity (Table 5) shows, that the absolute effect of heterogeneity is slightly stronger for the case with lower initial plutonium content in the fuel assembly (comparison of RGP 1011 to Fert 1). The higher initial Pu content reduces the absolute effect of heterogeneity slightly. However, in the relative evaluation the gain due to the heterogeneous arrangement is much lower for lower initial Pu content due to the improved breeding in the homogeneous RGP 1011 case with the lower initial Pu content. This demonstrates, that one separated change should never be observed isolated when a general insight is desired, but the isolated investigation supports the understanding of the influences. However, it has to be kept in mind, that the heterogeneous arrangement leads to a higher pin power and burnup peaking. Thus the fuel assembly power has to be reduced to keep a given linear heat rate limit which is dictated by the limits of fuel and cladding. Additionally, the question on the fertile pin behavior would have to be answered. Will the swelling due to the higher power production in the second half of the burnup period require a reduction of the pellet diameter? A reduced pellet diameter would immediately reduce the breeding gain compared to the homogeneous fuel assembly.

Table 5: Absolute and relative gain in fissile material amount due to the insertion of heterogeneity compared to the reference case at different fissile contents and at different burnup stages (absolute gain = case x – RGP, relative = case x/RGP-1)

|  |  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- | --- |
| Burnup [MWd/tHM] | absolute [particles/barn/cm] | | | relative [%] | | |
| 100 000 | 150 000 | 200 000 | 100 000 | 150 000 | 200 000 |
| RGP | 0 | 0 | 0 | 0.0% | 0.0% | 0.0% |
| Fert | 4.943E-05 | 5.659E-05 | 5.914E-05 | 47.1% | 60.9% | 95.3% |
| RGP 1011 | 0 | 0 | 0 | 0.0% | 0.0% | 0.0% |
| Fert 1 | 5.401E-05 | 5.995E-05 | 6.159E-05 | 29.9% | 28.7% | 27.5% |

The insertion of fine distributed moderating material leads to a gain in fissile material in all cases (see Table 6). The absolute gain increases clearly with burnup for all cases. The combination with the heterogeneous arrangement and the high initial Pu content seems to be most promising. The relative gain is highest in the RGP case due to the weakest over all breeding. The low relative values in the Fert 1 case are a result of the already enhanced breeding in this case due to the lower initial Pu content.

Table 6: Relative gain in fissile material amount due to the insertion of moderating material compared to the reference cases without moderating materials at different burnup stages (absolute gain = case x YH – case x, relative = case x YH/case x-1)

|  |  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- | --- |
| Burnup [MWd/tHM] | absolute [particles/barn/cm] | | | relative [%] | | |
| 100 000 | 150 000 | 200 000 | 100 000 | 150 000 | 200 000 |
| RGP | 0 | 0 | 0 | 0% | 0% | 0% |
| RGP YH | 1.034E-05 | 2.099E-05 | 3.086E-05 | 9.6% | 23.0% | 49.8% |
| Fert | 0 | 0 | 0 | 0% | 0% | 0% |
| Fert YH | 1.268E-05 | 2.457E-05 | 3.533E-05 | 8.7% | 15.5% | 28.6% |
| Fert 1 | 0 | 0 | 0 | 0% | 0% | 0% |
| Fert 1 YH | 1.010E-05 | 2.303E-05 | 3.448E-05 | 3.4% | 6.7% | 10.5% |
| RGP 1011 | 0 | 0 | 0 | 0% | 0% | 0% |
| RGP 1011 YH | 9.2338E-06 | 1.3347E-05 | 1.7294E-05 | 2.8% | 6.8% | 11.1% |

The reduction of the initial plutonium content in the fuel assembly is by far the most efficient method to improve the gain in the fissile material production. The method is even slightly more efficient in the combination with the heterogeneous arrangement. The Fert 2 case demonstrates already that there are some limits since the fuel arrangement has still to provide enough criticality to reach a critical core configuration at the begin of life. Even when this is achieved, it should be kept in mind that the reduction of the initial plutonium content has a high price. It leads to a decrease of the power density at constant core neutron flux, thus the neutron flux has to be increased to achieve an identical power level. However, the maximum core neutron flux is limited by the neutron fluence in the cladding and the related material damage over the lifetime of the fuel assembly, thus it is almost impossible to increase the core neutron flux above a certain limit. A possible compensation can only be given by an increase of the core size.

Table 7: Absolute and relative gain in fissile material amount due to reduction of the initial plutonium content in the assembly compared to the reference cases at different burnup stages (absolute gain = case Pured – case Puref, relative = case Pured/Puref-1)

|  |  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- | --- |
| Burnup [MWd/tHM] | absolute [particles/barn/cm] | | | relative [%] | | |
| 100 000 | 150 000 | 200 000 | 100 000 | 150 000 | 200 000 |
| RGP | 0 | 0 | 0 | 0.0% | 0.0% | 0.0% |
| RGP 1011 | 1.604E-04 | 1.899E-04 | 2.084E-04 | 210.0% | 267.2% | 423.3% |
| Fert | 0 | 0 | 0 | 0.0% | 0.0% | 0.0% |
| Fert 1 | 1.650E-04 | 1.932E-04 | 2.108E-04 | 153.4% | 175.7% | 227.4% |
| Fert 2 | 2.981E-04 | 3.447E-04 | 3.736E-04 | 330.1% | 373.8% | 482.1% |

**Safety Parameters**

Following the comparison of the operational parameters criticality and breeding of new fissile materials, the safety related feedback effects will be analyzed in comparison with the reference case in the next figures. The sodium void effect is calculated in a TREE calculation in the HELIOS code by a change of the sodium density to 1% compared to the reference state TREE calculation. The difference between both states is evaluated as sodium void effect in the infinite medium approximation. The real sodium void effect could only be evaluated by full core calculations. The infinite medium approximation is to be applied here since all evaluations are based on fuel assembly calculations with reflecting boundary conditions. However, for this study this the approximation seems to be adequate, since only the change of the sodium void effect compared to the reference solution is evaluated to investigate the effect of the changes made. The absolute sodium void effect in the infinite medium approximation in the reference case increases almost linearly with the burnup by 20%. The following general tendencies can be observed. The reduction of the Pu content increases the sodium void effect slightly; see the results for the RGP1011 case and the shift between the cases Fert, Fert 1, and Fert 2 (see Figure 13). The application of the heterogeneous arrangement reduces the sodium void effect by ~5%. The insertion of the fine distributed moderating material reduces the sodium void effect strongly by more than 20 to 25%. The reduction of the sodium void effect due to the heterogeneous arrangement and due to the introduction of moderating material is independent and combinable. Thus a reduction of the sodium void effect by 25 to 30% in the infinite system is achievable. From literature it is already confirmed that the gain in the infinite system can be combined with the traditional methods reducing the sodium void effect by increasing the neutron leakage from the core [20].



Figure 13: Changes of the sodium void effect in the infinite medium approximation for the different cases in comparison to the reference case

The fuel temperature or Doppler effect is calculated in a TREE calculation in the HELIOS code by a change of the fuel temperature by + 200 K compared to the reference state TREE calculation. The difference between both states is evaluated as Doppler effect. In general, the effect in the reference case is strongly reduced over the burnup of the fuel assembly (see Figure 14). In the reference case the negative Doppler effect decreases by more than a factor of two from BOL to EOL. Thus the processes induced by the irradiation of the fuel leading to fission product formation, breeding of Pu, and depletion of U-238 decreases the system stability. The Doppler effect is influenced by all three parameters, the absolute Pu content in the fuel assembly, the heterogeneous arrangement, and the insertion of the moderating material (see Figure 14). On the one hand, the absolute value of the negative Doppler effect increases with reduced initial Pu content in the fuel assembly, see RGP 1011 and compare Fert, Fert 1, and Fert2, as well as the corresponding cases with moderating material. The influence of the initial Pu content reduces with increasing burnup since in the cases with low initial Pu content significant breeding takes place, thus the Pu content increases. On the other hand, the effect is significantly reduced (by more than 40%) over the whole burnup cycle when the heterogeneous configuration is applied. This safety related drawback of the heterogeneous configuration could be a limit for the application, since the EOL Doppler effect is only -0.14 pcm/K anymore. The insertion of fine distributed moderating material is known to increase the Doppler effect significantly [21, 20]. The effect has been investigated and can be confirmed for the homogeneous case with an increase of ~120% over the whole burnup. The effect is even slightly stronger for cases with low initial Pu content, see RGP 1011 cases. The insertion of moderating material can completely compensate the reduction of the Doppler effect due to the heterogeneous arrangement, the influence is even over compensated by a factor of two. However, the efficiency of the inserted moderating material is slightly lower than in the homogeneous case, compare Fert to Fert YH. In general, the Doppler effect of the heterogeneous system with moderating material seems to be in an acceptable range for stable reactor operation.



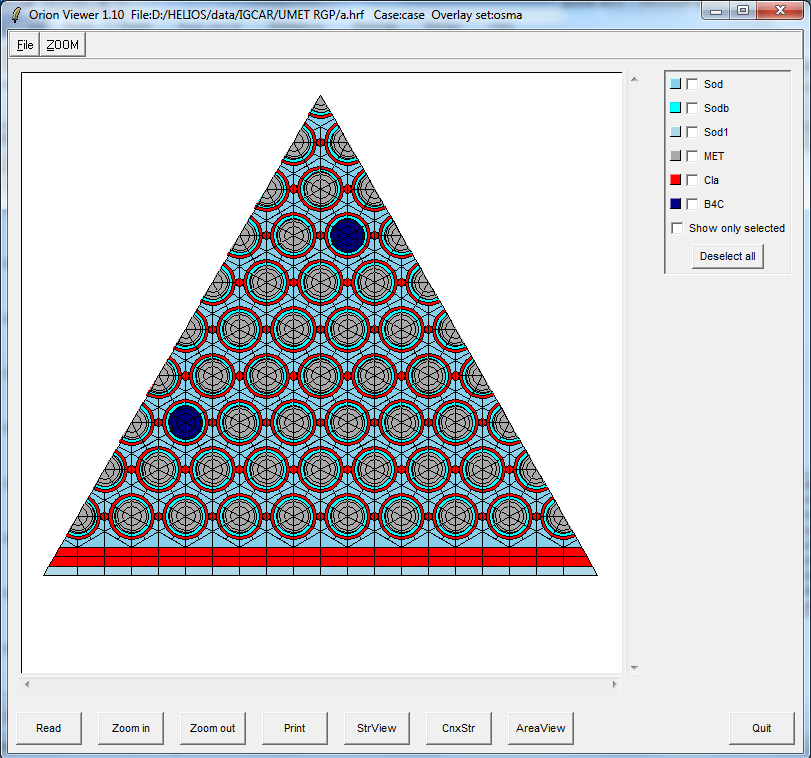
Figure 14: Changes of the Doppler effect for the different cases in comparison to the reference case

The coolant effect, which consists of the coolant temperature and the coolant density effect, is calculated in a TREE calculation in the HELIOS code by a change of the sodium temperature by + 50 K and the corresponding change of the sodium density compared to the reference state TREE calculation. The difference between both states is evaluated as coolant effect. The positive coolant effect increases over burnup by 25%. Thus, the behavior is very comparable with the sodium void effect and leads to a decrease of the stability of the reactor core. This is easy to explain since both effects are cause by identical physical processes. The reduction of the initial Pu content in the fuel assembly has almost no effect on the coolant effect, see RGP 1011 case (Figure 15). The application of heterogeneous configuration reduces the coolant effect like the sodium void, but the achieved reduction in the coolant effect is significantly stronger. The positive coolant effect is significantly reduced in all cases due to the insertion of moderating material. The efficiency of the insertion of the moderating material is slightly higher for the heterogeneous cases. This is most pronounced at BOL and with low initial plutonium content in the fuel assembly.



Figure 15: Changes of the coolant effect for the different cases in comparison to the reference case

The reactivity worth of the absorbing material (B4C with 80% B-10) is calculated in a TREE calculation in the HELIOS code by replacing two fuel rods by absorber rods compared to the reference state TREE calculation (see detail in Figure 16). The difference between both states is evaluated as worth of absorbing material. This way of approximating the insertion of control rods will give a first indication of the influence of the different changes in the fuel assembly. The absolute worth of the absorbing material decreases slowly from BOL to EOL by overall 7% in the reference case. The worth of the absorbing material is slightly increased in all cases at BOL but the worth decreases over burnup. On the longer term a slight decrease has to be expected. The insertion of moderating material, as well as the reduced initial Pu content and the application of the heterogeneous arrangement, leads to the increase of the absorber worth at BOL. Both changes are additive, but the reduction of the initial Pu content has a positive effect over the whole burnup, while the effect of the moderating material and the heterogeneous arrangement becomes negative at EOL. In general, the changes are in a small range and could even be in the accuracy range of the modelling (which does not reflect a SFR typical arrangement of the control assemblies).

**Figure 16: Changes in the absorber worth for the different cases in comparison to the reference case**

**Outlook to the Full Core and Discussion**

The lattice calculation is typically the first step in the neutron physical analysis of a fuel assembly. At this stage some important approximations have to be made to be able to do the calculations. The most important ones are the infinite medium approximation using reflective boundary conditions at the outer boundary of the calculated fuel assembly. This approximation is good for fuel assemblies which are surrounded by assemblies of the same type like it takes place in the center of a fast reactor core. However, even in the center of the core, the neighboring fuel assemblies of the equilibrium core will have a different burnup status. This cannot be considered in the lattice calculation as well as it is impossible to evaluate the effects of the neutron leakage from the core. Another important approximation is the burnup of the fuel assembly under constant power during the lattice calculation. This is in clear contrast to the real reactor operation. Here, the global power will be kept constant by the reactor operator by observing the core outlet temperature and keeping it within the foreseen operational margins. Thus the individual fuel assemblies will face the average neutron flux of all fuel assemblies at the different burnup stages within one operation cycle. To care for this effect an approximation for the correction of the lattice calculation to consider the fuel cycle management leading to a core configuration with differently burnt fuel assemblies has been developed. The approximation is based on the one group neutron flux curve over burnup calculated in the lattice code (see Figure 17). The neutron flux is varied over burnup to keep the power constant while the amount of fissile material changes due to breeding and burning processes.



Figure 17: Neutron flux over burnup as it is used in the lattice calculation to achieve a constant fuel assembly power

The one group neutron flux curve is used for the determination of the approximated average flux in the core. For the correction 4 cycles, each with an average burnup of 50 000 MWd/tHM are assumed with the average burnup stages to 50000MWd/tHM, 100000MWd/tHM, 150000MWd/tHM, and 200000MWd/tHM as approximation. The average flux curve for one operational cycle of the fast reactor core is now calculated for a core containing the same amount of fuel assemblies of each burnup stage. This average flux curve is put into relation to the one group neutron flux curve over burnup given in Figure 17. The result is the correction curve for the approximation of the change of the pin power over burnup given in Figure 18.



Figure 18: Correction curve for the pin power after using the described cycle correction for the lattice calculation

Applying the correction curve to the max. pin power of the fuel assembly of the Fert 1 YH case leads to an approximation of the cycle corrected max. pin power which is determined to be a good approximation for the equilibrium core. The corrected max. pin power curve is given in Figure 19. The max. pin power at the begin of live is reduced by ~7%. However this value is still significantly higher than the currently defined limit value of 450 W/cm. Nevertheless, this correction is very promising since the power is corrected exactly at the most demanding point for the heterogeneous fuel assembly design, at BOL. This is a drawback of the lowest content of fissile material which appears at BOL.



Figure 19: Approximated max. pin power in the Fert 1 YH fuel assembly after applying the correction for the fuel management

The request for the reduction of the average linear power in the fuel assembly which is required to keep the max. linear power limit give clear reason to think about possible compensation methods and requires an evaluation of the expected breeding gain versus the newly imposed challenges. As already mentioned, the compensation has to take place in the full core arrangement. The required data and possible compensation methods are collected in Table 8 and will be discussed in detail here. The max. pin power of the reference case and the Fert 1 YH case are given for the rough estimation. The power peaking factor for the Fert 1 YH case can be calculated to 1.444. One possible compensation step is the increase of the max. linear power from 450 W/cm to 500 W/cm. Both numbers are currently in the general discussion for the future core design. Additionally, it should be kept in mind, that each sub-channel is only exposed to two high power fuel rods in the heterogeneous case and not to three in the homogeneous case. The next possibility could be an increase of the core size by one ring. The idea is to incorporate one blanket ring into the core since the in-core breeding is significantly improved and thus fewer blanket assemblies will be required for the breeding. The step increases the number of fuel assemblies for power production by a factor of 1.22. However, it should be reminded that the power distribution in a real core is not completely flat, thus this value is only a rough estimation. The combination of both methods already ~36% of the 44% of the power peaking. Finally, the described fuel cycle correction approximation should be taken into account. All changes together lead to a rough estimation of a correction potential of ~45%. Thus the three methods together have the potential to roughly compensate the increase of the maximum power in the Fert 1 YH case. A further increase of the core size would increase the number of fuel assemblies once more by 20%. Additionally, the elimination of the fertile blanket would be a well appreciated change for the point of view of nuclear security.

Table 8: Estimation of a possible compensation of the reduced power density

|  |  |
| --- | --- |
| 453.9 | max. power reference case |
| 655.5 | max. power Fert 1 YH |
| 1.444 | power peaking factor Fert 1 YH |
| 1.111 | max. linear power from 450 W/cm to 500 W/cm |
| 1.221 | from 10 rings to 11 rings |
| 1.357 | combined 450 to 500 W/cm and 10 to 11 rings |
| 1.07 | cycle correction approx. of max. pin power |
| 1.445 | 450 to 500 W/cm, 10 to 11 rings, cycle corr. approx. |
| 1.199 | increase of core size by one ring from 11 to 12 |

Based on thes results a discussion has to be forced. Maybe it is more attractive and less challenging to apply a homogeneous configuration with reduced initial Pu content to achieve improved breeding on homogeneous basis without penalty in the pin-power and burnup distribution. The reduction of the initial Pu content compensated by a slightly increased core size seems to be the most efficient change. The increase of the core size on the cost of a reduced heterogeneous breeding blanket seems to be attractive from proliferation point of view reducing the proliferation risk in the homogeneous core. Additionally, the increase cycle length could be an economically really attractive opportunity since the short cycle length of the currently operating fact reactors is one of the major obstacles for the economic operation of this kind of reactors [1].

**Conclusions and Outlook**

An investigation of different options for the optimization of a sodium cooled fast reactor with metallic fuel has been performed. The sensitivity to three different parameters have been studied – homogeneous versus heterogeneous arrangement, different initial plutonium contents compensated by an increased core size, and the possible insertion of fine distributed moderating material. It is shown that breeding is most sensitive to the initial Pu content with an increase of breeding with decreasing initial Pu content. The feedback effects are most sensitive to the insertion of moderating materials which leads to a clear reduction of the positive feedback effects and an enhancement of the negative feedback effects.

According to Kord Smith a 100 pcm ‘jungle’ for LWR lattice calculation exists [33], when the calculations are performed following a good practice. In SFR the uncertainty is significantly higher due to the lower level of validation and real reactor operation experience and due to the significantly higher sensitivity of fast reactor cores on input data (e. g. geometry, libraries). Thus all observed absolute differences could be seen as not significant and dependent on the calculation tool and method. However, the relative differences between results for calculations based on the identical code and input with single changes are a meaningful tool to identify the sensitivity of operational and safety parameters on changes in the geometrical and the material arrangement.

The investigation of the different cases shows that the combination of all three parameters has the potential to significantly improve the breeding in a metal fueled fast reactor whereas the decrease of the initial Pu content compensated by and increase core size seems to be the most efficient change. However, the BOL criticality of this kind of fuel assemblies will be clearly decreased. This effect has to be compensated with an increased core size which has to be determined in a more advanced core design in a follow up step. The investigation of the safety relevant feedback effects demonstrates that the sodium void, the Doppler effect, and the coolant effect will be changed forming a more stable reactor core using these fuel assemblies. This combination of improved breeding and enhanced feedback effects like it is shown to be possible in heterogeneous fuel assemblies with fine distributed moderating material and possibly with reduced initial Pu content opens completely new possibilities in the design of metal fueled SFR cores. Important side effects are the increased in-core breeding which allows extended cycle times and the enhanced security of nuclear materials since the fresh bred plutonium is in the same fuel assembly as the fuel with the Pu vector of civil plutonium. The separation of fertile and the fuel rods will be in this case much more challenging than in the case of separated fuel assemblies and impossible in the case of a homogeneous arrangement with reduced Pu content. A premature unloading of the fertile assemblies like it would be attractive to produce weapon grade plutonium is almost impossible or at least economically much less attractive. However, it has to be accepted that the max. pin power in the heterogeneous assemblies is significantly higher than in the homogeneous assemblies. This has to be compensated by a reduced average assembly power. An approximation has been developed to correct the lattice code typical constant power over burnup with a more realistic approximation for a constant integral reactor power in a core with four cycles.

A first discussion and investigation of the possible compensation methods for the required reduced fuel assembly power is given. A slightly increased maximal linear heat rate, the replacement of one row of the radial fertile blanket by fuel, and the correction for the fuel cycle management have the potential to balance the reduced assembly power on full core level.

For a future application of the proposed ideas, more investigations are required. First of all the effects of the required decrease of the linear heat rate on the full core design have to be investigated in the case of the heterogeneous assembly. Ideally, this should be complemented with an investigation of methods to reduce the power imbalance at BOL on the fuel assembly level. Finally, a detailed evaluation has to take place to decide which of the proposed changes the most attractive one is. All proposed changes lead to drawbacks on the fuel assembly design, the coolability, and the core design. Thus thermal hydraulic simulations as well as full core calculations first on pure neutronic level and later on in coupled simulation will be required and possible effects on structural mechanics and the structural feedback should be investigated.

However, the investigation of the sensitivities should help to start a discussion to make up informed decisions on the optimization of breeding in the India fast reactor program. The next proposed step will be full core calculations based on ERANOS/ECCO a well validated fast reactor design tool. Unfortunately, this study hasn’t been possible with ERANOS/ECCO due to the geometric limitation s in ECCO. However, tests performed in HELIOS have shown that the detailed arrangement of the fertile pins in the fuel assembly does not influence the neutronic result significantly. The next steps will concentrate on the most promising finding of the work, designing a homogeneous core with reduced averaged Pu content and without fertile blankets. This could lead to a more proliferation resistant core design.

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