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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.

1. 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 (Merk et al., 2015), as defined in the Generation IV International Forum (Generation IV international forum, 2014). 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 (Heusener, 1970). 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 (Barthold et al., 1977; Tzanos and Barthold, 1977), regarding the correlation between breeding and sodium void effect (Bruna et al., 1977), and regarding inherently safe reactor designs (Lancet, 1977). 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 (Barthold et al., 1979) has been developed. In the 90ies new studies have been given on the heterogeneous core optimization with regard to reduced sodium void (Chang et al., 1991). 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 (Hill and Khalil, 1990). 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 ($\sim$5 m) in combination with a very small core height ($\leq$ 1 m) 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

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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 (Jevremovi et al., 1993). 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 (Tsujimoto et al., 2001). 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 (MacDonald, 1966; Tsujimoto et al., 1994; Hamid and Ott, 1993). 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 (Merk et al., 2011, 2012). It has been demonstrated that this method has no significant impact on the power and burnup distribution (Merk and Weiß, 2011) and that the use of fine distributed moderating material offers new possibilities for the optimization of core designs for transmutation (Merk, 2013). 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 (Merk, 2013).

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 (Villarino et al., 1992) or APOLLO (Sanchez et al., 1988). 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 (Wemple et al., 2008). 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:

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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.

2. 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) (Riyas, 2009, 2010, 2014). 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 (Waltar and Reynolds, 1981). 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.

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/cm^3 along the formula for liquid saturated sodium at 773 K given in Waltar, Reynolds (Waltar and Reynolds, 1981).

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/cm^3
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%

The modeled geometric arrangement of the reference system with 10 rings, following the IGCAR design, is shown in Fig. 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 (HELIOS-2, 2011).

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 (HELIOS, 2003). A cross comparison with MCNP for the initial value of fuel temperature and moderator effect on k_{inf} 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 (Merk et al., 2011, 2012). 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 (Rachamin et al., 2013). 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 rearrangements. 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 (Riyas, 2010) see Fig. 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.