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Neutron shielding studies on an advanced molten salt fast reactor design

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ABSTRACT

The molten salt reactor technology has gained some new interest. In contrast to the historic molten salt reactors, the current projects are based on designing a molten salt fast reactor. Thus the shielding becomes significantly more challenging than in historic concepts. One very interesting and innovative result of the most recent EURATOM project on molten salt reactors – EVOL – is the fluid flow optimized design of the inner reactor vessel using curved blanket walls. The developed structure leads to a very uniform flow distribution. The design avoids all internal structures. Based on this new geometry a model for neutron physics calculation is presented. The major steps are: the modeling of the curved geometry in the unstructured mesh neutron transport code HELIOS and the determination of the real neutron flux and power distribution for this new geometry. The developed model is then used for the determination of the neutron fluence distribution in the inner and outer wall of the system. Based on these results an optimized shielding strategy is developed for the molten salt fast reactor to keep the fluence in the safety related outer vessel below expected limit values. A lifetime of 80 years can be assured, but the size of the core/blanket system will be comparable to a sodium cooled fast reactor. The HELIOS results are verified against Monte-Carlo calculations with very satisfactory agreement for a deep penetration problem.

1. Introduction

In the today's view molten salt reactors have a long history, based on the somewhat curious idea of the early phase of nuclear development in the late 40ies and early 50ies, the nuclear aircraft. "The idea of using molten fluoride salts and thus liquid nuclear fuel in a reactor is rather old. Molten salt reactors were already proposed during the post-World War II attempt to design the nuclear powered aircraft. The Aircraft Reactor Experiment, a small thermal reactor (2.5 MW) using circulating molten salt, operated for several days in 1953" (MacPherson, 1985). This first experiment has been followed by a larger scale experiment with 8 MW thermal, the Molten Salt Reactor Experiment (MSRE). "Design of the MSRE started in the summer of 1960 and construction started 18 month later, at the beginning of 1962. The reactor went critical in June 1965, and was briefly at full power a year later "(MacPherson, 1985). A major step in the MSRE was the demonstration of the use of thorium as fertile material and U-233 as fissile material. The reactor was operated until December 1969 and the U-235 fuel salt was successively replaced with U-233. Finally, the reactor was operated based on U-233 fuel for several months. It was the first time U-233 has been used as reactor fuel. (MacPherson, 1985).

The molten salt reactor technology has gained some new interest nowadays (Waldrop et al., 2012). This interest has been focused in the EURATIOM project MOST – Review on Molten Salt reactor Technology (Renault and Delpech, 2005; Mathieuet et al., 2005). Following the MOST project, two recent important projects have been launched EVOL (EVOL – Evaluation and Viability of Liquid Fuel Fast Reactor, 2010; Renault and Delpech, 2010) and MOSART (Ingatiev et al., 2012, 2007). This renewed interest can be explained by some really interesting features of molten fluoride salts. "Molten fluoride salts have some beneficial characteristics, like the wide range of solubility of uranium and thorium, the thermodynamic stability and the resistance against radiologic decomposition, the low vapor pressure at operation temperature and the compatibility with nickel based alloys which are traditionally used as construction material (MacPherson, 1985). In contrast to the MSRE launched in the 60ies, both projects focus on the development of a molten salt reactor with fast neutron spectrum.

In "In addition molten salt fast reactors (MSFRs) exhibit large negative temperature and void reactivity coefficients. This is a unique safety characteristic not found in solid-fuel fast reactors (Mathieu et al., 2009)" (Generation IV International Forum, 2009). This unique safety characteristic and the specific features of the fluoride salts described above lead to superior inherent safety characteristics. MSFR systems have been recognized as a long term alternative to solid-fuelled fast-neutron systems with unique favorable features (negative feedback coefficients, smaller fissile inventory, easy in-service inspection, simplified fuel cycle, etc.) (Generation IV International Forum, 2009). In the frame of the development of future energy resources and an improved nuclear waste

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management, the molten salt reactor concept offers a large capability of different operational regimes. Molten salt reactors are one of the six concepts selected by the Generation IV International Forum (GIF) Generation IV International Forum, 2009 for further study. In contrast to molten salt reactors previously studied, the specificity of the MSFR is the removal of any solid moderator from the core. This choice is motivated by the study of parameters such as feedback coefficient, breeding ratio, graphite lifespan, and U-233 initial inventory (Mathieu et al., 2009). This change results in a fast neutron spectrum molten salt reactor (Merle-Lucotte et al., 2011).

The EVOL MSFR is proposed to be operated in the Th/U-233 fuel cycle with fluoride salts. Since U-233 does not exist in nature, it is foreseen to start the reactor with plutonium and minor actinides (TRUs or transuranium isotopes) as fissile material which is produced in currently operating light water reactors (Merle-Lucotte et al., 2009).

One very interesting and innovative result of the EVOL project is the fluid flow optimized design of the inner reactor vessel using curved blanket walls (Rouch, 2012a,b; Rouch et al., submitted for publication). The developed structure which leads to a very uniform flow distribution without using internal flow guiding structures is an important step forward in the design of molten salt fast reactors. The absence of internal structures avoids the significant problems due to the high mechanical stress and the exposition to very high fast neutron flux. This high fast neutron flux would lead to a rapid material degradation caused by irradiation damages.

Based on this new geometry a model for neutron physics calculation is presented in this publication. The major steps are: the modeling of the curved geometry in the unstructured mesh neutron transport code HELIOS and the determination of the real power distribution for the new geometry. The developed model is then used for the calculation of the neutron fluences in the inner and outer wall of the system. In a final step, an optimized neutron shielding strategy for the molten salt fast reactor is developed to keep the fluence in the safety related outer vessel below the limit values over an operation time of 80 years.

2. Known problems in MSRE and new challenges

Some special problems with the materials in molten salt reactor already appeared in the MSRE. "Hastelloy N used for the MSRE was subject to a kind of “radiation hardening due to accumulation of helium at the grain boundaries…it is still desirable to design well blanketed reactors in which the exposure of the reactor vessel wall to fast neutron radiation islimited" (MacPherson, 1985). This described problem with the fluence of fast neutrons will be significantly increased in a MSFR due to several reasons. There are some major differences in the design of a MSFR compared to the MSRE:

- significantly increased power density, thus significantly higher neutron flux level in the core;
- the neutron spectrum in MSFR is significantly harder since there is no moderator existing, thus a higher share of neutrons above 1 MeV appears;
- the fission reactions happen everywhere in the vessel of a MSFR, in contrast to the very well defined core in a thermal molten salt reactor given by the graphite moderator;
- the consequences is that fast neutron are born directly at the wall, no real slowing down can appear before the neutrons hit the vessel wall.

3. The model

The basic arrangement of the main components of a molten salt reactor is given in Fig. 1. All main reactor components (core with blanket, heat exchanger, pumps, and draining tanks) are located in a surrounding steel vessel. The molten salt is fed into the core (area inside the fertile blanket) through perpendicular arranged pipes. This salt is configuring the reactor core by forming a critical mass. In this area the majority of the fission reactions take place. The hot salt is withdrawn at the top of the core and pumped into the heat exchanger and from there back into the core. This basic arrangement has been improved significantly in the frame of the EVOL project. The improved proposal contains of a curved wall geometry with optimized flow conditions. This curved wall separates the blanket from the core. The core and the blanket have to be surrounded by an outer vessel. The reduced 2D geometry for the HELIOS modeling consists only of the core and the blanket surrounded by the outer vessel. All other structures like heat exchanger, pumps, and surrounding reactor vessel are not considered for the neutronic simulation. The 2D geometry is a vertical cut through core and blanket.

The basis for the determination of the fluences which appear in the walls of the inner and outer vessel is a detailed model of the curve wall geometry of the inner vessel and the whole core including fuel, blanket, and outer vessel. These neutron fluences are used for the determination of the feasible operation time of the inner and outer vessel.

For the calculations of the neutron field the HELIOS 1.10 code system with the internal 47 energy group library is used (HELIOS, 2003). The code is a 2D spectral code with wide unstructured mesh capabilities and a transport solver, based on the current coupling collision probability method (Villarino et al., 1992). The staring point for the development is a 15 by 15 cm grid in the center of the system. This grid is surrounded by some compressed cells in the blanket area to open the possibility to increase the blanket size for the shielding studies. The model consists of 37 times 21 cells. Additionally, the curve blanket wall like it has been invented by Rouch and Vu (Rouch, 2012a,b; Rouch et al., submitted for publication) is inserted using the unstructured mesh capabilities. This leads to over all 922 HELIOS calculation regions. The curved wall is approximated by straight line segments. The coordinates for the reproduction of the curved wall are determined with the help of CAD. The visualization of the 2D model forming a vertical cut through core and blanket is given in Fig. 2. The core region with the mix of fuel and fertile salt is given in red. This area is surrounded by the curve blanket wall with a constant thickness of

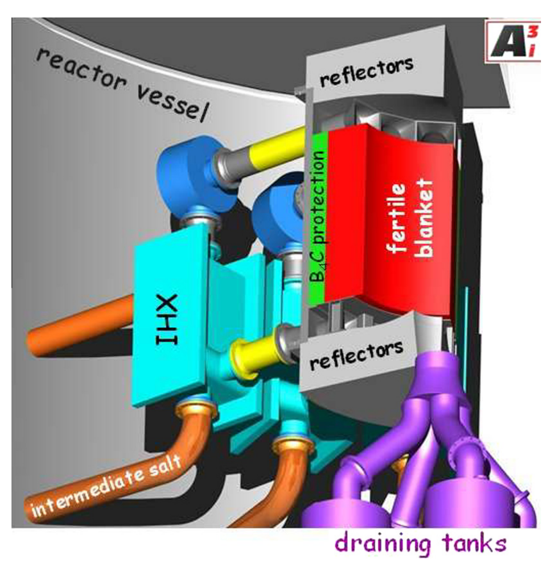

Fig. 1. Basic arrangement of a molten salt fast reactor as given in the EVOL benchmark description (SALT FAST REACTOR, 2011).

20 mm in green. The blanket area with the pure fertile material is given in grey. This area is surrounded by the outer safety related vessel. This is in the draft supposed to have a thickness of 30 mm and is given in dark blue. The outer vessel is surrounded by a 20 cm thick graphite reflector poisoned with 5% natural boron, light blue.

The calculations are based on the material configuration, and the boundary conditions of the EVOL benchmark definition. The reference MSFR is a 3000 MW_{th} reactor with a fast neutron spectrum and based on the thorium fuel cycle. It will be started with TRU elements as initial fissile load. Optimization studies have been performed prior to the beginning of the EVOL project, relying on neutronic considerations (feedback coefficients and breeding capacities), heat evacuation efficiency, and resulting in MSFR configurations with a total fuel salt volume of 18 m^3. Half of the salt is located in the core and half in the external circuits as explained above. The salt's thermal hydraulic behavior is closely coupled to its neutronic behavior, because the salt's circulating time ($\sim$4 s) and the lifetime of the precursors of delayed neutrons ($\sim$10 s) are of the same order of magnitude (EVOL, 2012; MOLTEN SALT FAST REACTOR, 2011).

The salt configuration consists of 77.5% LiF with ThF_4-(Pu-MA)F_3 in the core and with pure ThF_4 in the blanket. The overall fuel salt volume of 18 m^3 contains 30619 kg Th, and 12661 kg TRU following the TRU vector given in Table 1 (MOLTEN SALT FAST REACTOR, 2011).

4. Power and neutron flux distribution

The neutron flux distribution, the neutron flux spectrum, and power for the vertical 2D cut as calculated with HELIOS are given in Figs. 3–5. The elevation of the upper end of the bars, representing each a calculation region of HELIOS, reflect the values in the cells. The fast as well as the epithermal neutron flux distribution (values in Neutrons/cm^2/s) show a clear maximum in the center and a cosine like distribution. The cut-off in this case is 0.1 MeV.

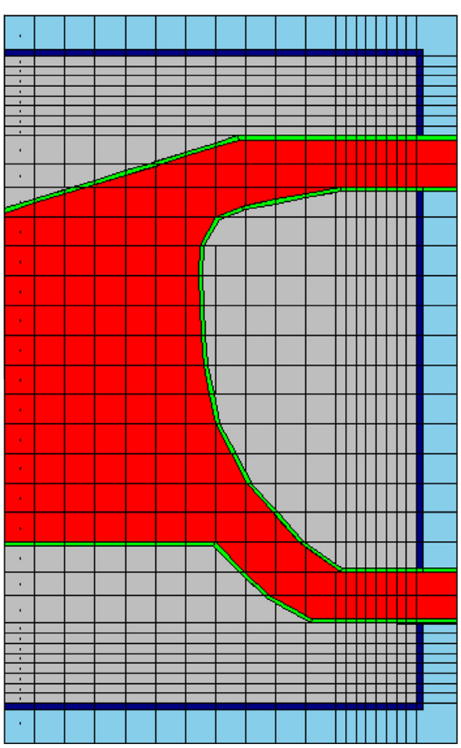

Fig. 2. Model of the advanced molten salt fast reactor developed in HELIOS.

Table 1
TRU vector from PWR fuel (burnup of 60 GWd/tHM) and after five years of storage.

Np 237	6.30%
Pu 238	2.70%
Pu 239	45.90%
Pu 240	21.50%
Pu 241	10.70%
Pu 242	6.70%
Am 241	3.40%
Am 243	1.90%
Cm 244	0.80%
Cm 245	0.10%

The distribution is expected due to the uniform fissile material distribution caused by the homogeneous mixing of the fissile material salt in the LiF carrier salt. Additionally, there is no enrichment variation like it is traditionally appearing in a fast reactor core with solid fuel. This design feature is simply impossible due to the homogeneous mixture. The major difference between the fast neutron flux and the epithermal neutron flux is that the fast neutron flux is strongly bound to the inner vessel with the fissile material. The epithermal neutron flux shows only a small change at the boundary between the inner fissile zone and the fertile zone, which is even hard to observe. This difference demonstrates the effect of the high share of carrier salt and of light materials in the salt.

Both facts cause efficient neutron scattering and thus soften the neutron spectrum, see Fig. 4. The normalized neutron flux spectrum in the molten salt fast reactor (green) is compared with the neutron spectrum in a sodium cooled fast reactor (European Fast Reactor – red) and with a lead cooled fast reactor experiment (GUINEVERE uranium metal fuel in a lead matrix - black). It is obvious, that the neutron flux spectrum in the molten salt fast reactor is significantly softer than in classical liquid metal cooled systems. The peak in the neutron spectrum where the most neutrons appear in the MSFR is between 20 keV and 100 keV. This is in strong contrast to the liquid metal cooled fast reactors where the peat appears between 100 keV and 1 MeV.

The normalized power distribution (see Fig. 5) follows the neutron flux distribution, but the power is only produced in the core inside the inner vessel where the fissile material is available. The power distribution will not depend on the burnup of the fissile material, since the salt is moving and thus the fission products and the fissile material will be continuously redistributed all over the core. A small share of the power is also produced in the pipe like structures where the fuel is flowing in and out of the real reactor core. The power production in the blanket area is very low as long as there is no fissile material bred, but it will increase slightly during operation and will reach an asymptotic value due to the online processing of the salt. The absolute amount will be dependent on the speed of salt processing to separate the bred fissile material from the blanket salt. The bred fissile material will not only stay in the surrounding of the core where it is formed, but it will be distributed in the whole blanket due to the salt movement.

The major information for the material degradation due to neutron irradiation can be drawn from the maximal neutron flux in the highest irradiated cells of the vessel wall. This value can be used to determine the permissible operation time until the fluence limit in the material is reached in this cell. The neutron flux distributions and the maximum values for neutrons with energies higher than 1 MeV in the vessel walls are given in Fig. 6. The neutron flux in the inner vessel wall (left) is highest in the center of the cone which forms the top wall. The highest value appears at this position, since this place is due to the construction of the wall closest to the center of the core, where the maximal neutron flux value appears. Two other areas with high neutron flux exist at the inner

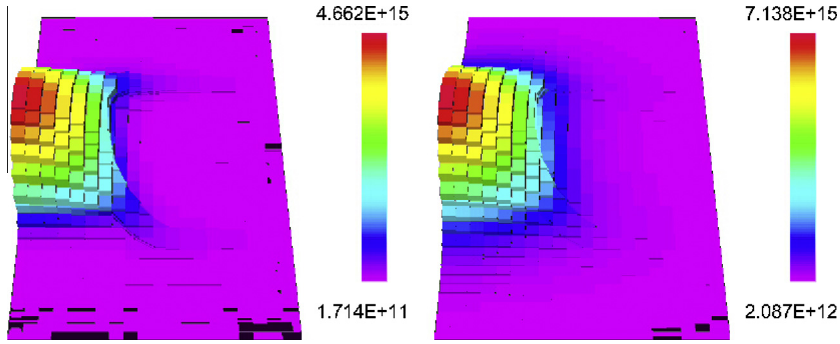

Fig. 3. 2D fast (left) and epithermal (right) neutron flux distribution in the advanced molten salt reactor.

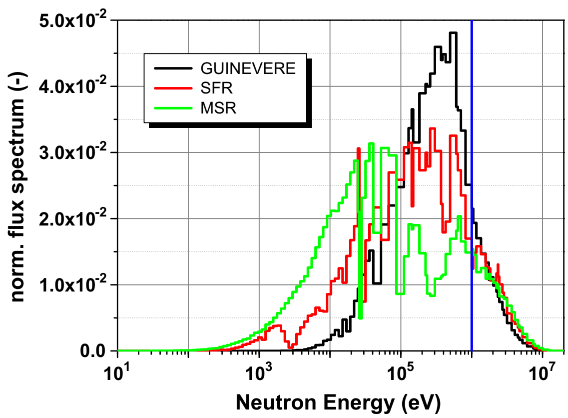

Fig. 4. Average neutron flux spectrum in the advanced molten salt reactor.

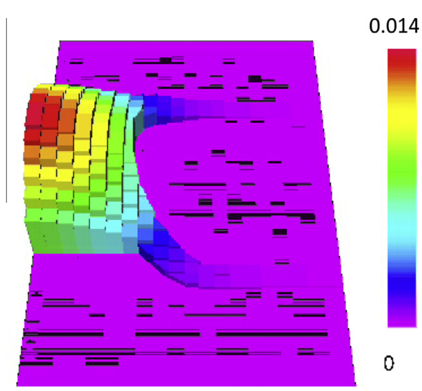

Fig. 5. Normalized 2D Power distribution in the advanced molten salt reactor.

wall. One appears at the center of the bottom part of the vessel and another at the outer wall slightly above the horizontal midplane where the diameter of the torus is smallest. These two positions are the parts of the bottom and the radial wall which are closest to the core center. In all these regions the neutron flux is very high. This leads to a very short time to reach the limit of 10^{20} Neutrons/cm^2 which is usually a limit value for the materials under irradiation in a molten salt reactor as long as this component has a safety related function. "In an MSR the reactor core is in direct contact with its inner container layer. Therefore, the inner container layer serves as a fuel cladding layer that must withstand high levels of neutron flux. High-nickel alloys embrittle when subjected to high neutron fluence (>$\sim 10^{20}$ n/cm^2) at high temperatures (>500 °C). The embrittlement has both thermal and energetic neutron pathways. Thus the reactor vessel must be shielded from high neutron fluxes, or materials other than high-nickel alloys must be employed" (Holcomb et al., 2011). The limit value is already accumulated after some days. Thus, the inner wall cannot carry any safety related function. Additionally, there is no possibility to protect this wall from the high neutron flux, since the fission reactions will occur directly at the wall. Hence, there is no potential to decrease the energy of the fission neutrons by collisions. Based on these boundary conditions it can be stated, that any safety related function has to be taken away from the inner wall since a damage of this wall by the high neutron flux cannot be avoided. Thus a fractional failure resulting in a leak has to be considered. Fortunately, this case is perfectly backed up by the fail safe principle since mixing the fertile material of the blanket with the fissile core material leads to a reduction of the criticality. This leads to a shut down by an inherent passive mechanism. Since the inner vessel cannot fulfill any safety related function, a second outer vessel is required for the safety related confinement of the radioactive material. This function is carried by the reactor vessel, but this vessel contains

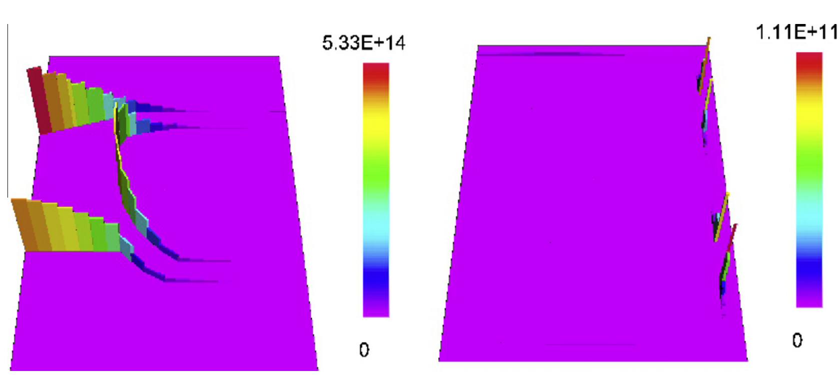

Fig. 6. 2D Neutron flux distribution of neutrons with energy >1 MeV in the inner (left) and the outer (right) vessel wall of the system.

all the primary piping, the pumps, and the heat exchangers. When a failure of the inner vessel has to be taken into account, an additional vessel (her called outer vessel) is required as a safety related system. The pumps and the heat exchanges should not be exposed to the hot salt from the outside in the case of a nonsafety related occurrence. The radiation exposure of this outer vessel is given in the right side of Fig. 6. The outer vessel has the highest exposure to the neutron flux at the penetrations of the pipework of the primary system. This is the only area which cannot be protected by the blanket material. The outer vessel is almost in contact with the fissile material at his position, only the wall of the inner vessel is in between. In the calculated stagnant case, the highest flux appears in the lower pipe but the flux in the upper pipe will increase slightly when the fuel movement is taken into account. The major reason is the movement of the delayed neutron precursors which are carried with the salt flow. The precursors induce delayed neutron production in all pipework depending on the precursor lifetime, but the influence is highest at the outlet since some precursors have a very short half-life. The delayed neutrons induce fissions, which will produce some neutrons, even when the leakage in the pipework is significantly higher than in the core.

5. Shielding optimization

For the optimization of the shielding of the outer vessel is following the strategy given by (MacPherson, 1985). It has been proposed to construct a well blanketed system. The effect of an increase of the blanket size is shown in Fig. 7 as an example starting with a very small outer vessel to highlight the effect of the increase of the blanket size. The visualization of the calculated geometry is given in the upper part. In each step it is indicated by the red circle where the blanket size has been increased. In the lower part of the figure the consequences on the neutron flux distribution in the wall is given. In the first step the upper blanket is increased simply by doubling the dimension of the two blanket nodes. This change leads immediately to a decreased neutron flux at the top part of the outer vessel. The maximum appears at the lower right corner where the primary pipework is attached. In the second step the radial blanket is expanded. The expansion reduces the maximal neutron flux by ~25%. The maximum appears now in the bottom of the safety vessel (see central part of Fig. 7). In the third step the lower blanket is increased in size. This change leads to a reduction of the maximal neutron flux by ~70%. The region with the maximal flux has moved now once more to the area of the connection of the primary pipes to the outer vessel. In contrast to case one, the maximum appears now at the upper part of the lower pipe, not at the lower part anymore. These steps indicate how efficient the strategy of increasing the blanket is. Thus this method will be used in the first step optimization of the shielding for the outer vessel. Nevertheless, it has to be kept in mind that this strategy leads to a rapid increase of core size, thus it is costly and the efficiency will decrease with increasing blanket size.

The optimization of the shielding of the outer vessel (see Fig. 8) starts with a minimal blanket thickness of 30 cm at the upper and lower penetration of the primary pipes through the outer vessel. This leads to thickness of roughly 30–70 cm of the upper and lower blanket and roughly 45–95 cm in the radial direction. In this base case, it takes about 2 years to accumulate a neutron fluence of 10^{20} Neutrons/cm^2 in the region with the highest neutron flux exposure. In the first step, the thickness of the axial and radial blankets is doubled, thus the minimal blanket thickness is now 60 cm. This doubling of the blanket thickness increases the acceptable operation time until the fluence limit is reached to ~6 years. The nextdoubling from 60 cm to 120 cm blanket thickness increases the possible operation time to roughly 20 years. To double the thickness of the blanket once more seems not to be reasonable since the size of the blanket would be than more

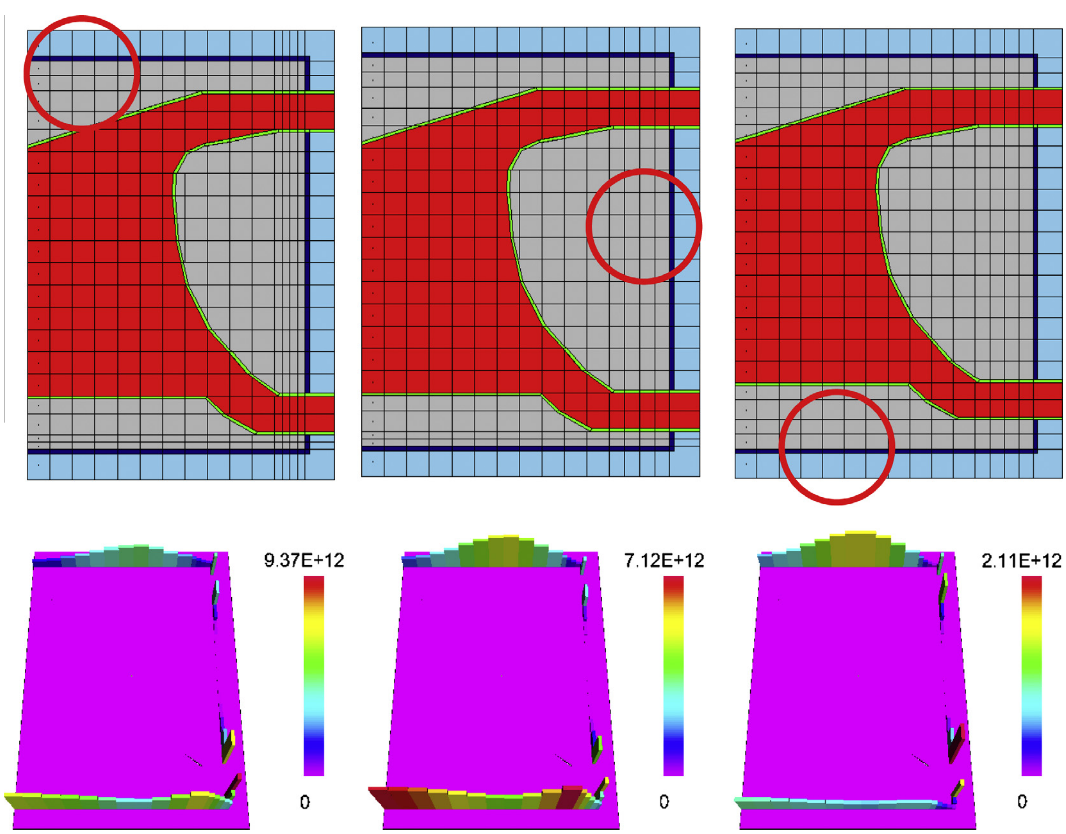

Fig. 7. Effect of stepwise increase of the blanket size.

than the height of the active zone. It would simply be too costly to continue this way for the optimization. A new way in the fast reactor development has been proposed recently by a group of the Indira Gandhi Centre for Atomic Research IGCAR. This group optimized the shielding of the prototype fast breeder reactor (PFBR) design for their next step, the commercial fast breeder reactor (CFBR) design. They proposed the use of ferro boron as shielding material (Sunil Kumar et al., 2010). Using this idea is the next step, the outer 30 cm of the blanket will be filled in a first approximation by ferro boron instead of fertile salt. Thisleads to doubling of the possible operation period of nearly 30 years. Replacing the half of the reflector with ferro boron increases the possible lifetime to 60 years. A detailed analysis of the region where the maximal neutron flux appears indicates still the lower penetration area of the primary circuit pipes. A sleeve made from nickel based alloy is introduced in the next step to improve the situation. The thickness of the sleeve is determined in a first guess to 6 cm. The introduction of the sleeve increases the acceptable operation time until the fluence limit is reached to approximately 68 years. The thickness of the sleeve to almost 10 cm leads to an allowable operation time of 82 years. The sleeve seems to be a very promising solution, but it should not be forgotten, that the sleeve is to be seen as a part of the inner vessel. The sleeve will face an irradiation damagehigher than the outer vessel thus it has to be replaced after the limit operation time like it has to be foreseen for the inner vessel.

All blanket sizes have been increased by the same value in the shielding optimization study above. After fulfilling the acceptable exposure time, the blanket sizes are optimized independently to reduce the system size as much as possible. The maximal exposure with neutrons with energies higher than 1 MeV appears in all steps at the salt inlet pipe, where the outer vessel is attached to the inner vessel. With this knowledge, the upper and lower blankets have been reduced successively until the exposure of the upper and lower part of the outer vessel is comparable. The neutron flux at the different positions after the optimization is given in Fig. 9. The neutron flux at the highest exposed parts of the top and the bottom of the vessel is now in the same range than in the highest exposed lower inlet of the primary system and the lifetime is ~81 years. For better validation of the results, this final optimization step has been recalculated with an improved transport approximation. This increases the calculation time by roughly a factor of 30. Thus the method is not affordable for an optimization, but for the final solution. The use of the improved transport method increases the flux slightly and leads to a final lifetime of ~ 77 years.

A visualization of the most challenging points in the outer vessel in correlation with the inner vessel wall is given in Fig. 10. The inner vessel wall is given in red. The sleeves can be recognized as thick red areas at the right side, where the pipes penetrate the outer wall. The high flux values in the outer walls appear exactly close to the sleeves. Additionally, high flux values occur at the transition of the core to the pipe like structures. There the blanket size is lowest and these areas are close to the big volume of the active core. Generally, the problem of neutron shielding to protect the vessels against high neutron damage is not dependent of the general structure of the core – rectangular EVOL benchmark configuration or optimized flow geometry, but rather a problem of the high neutron flux and the deep penetration behavior of fast neutrons.

The geometry resulting from the optimization is given in Fig. 11 including the dimensions of the outer vessel. The upper and lower blanket (grey) and the regions with the ferro boron absorber material (cyan) can be reduced significantly. These blankets are finally less than half as thick as the radial blanket. The inner vessel with the curved wall geometry has a height of 2 m and 30 cm and the diameter of the structure is 3.3 m. Additionally, this inner structure has been slightly optimized by extending the inclined inlet pipe like part. This change reduces the neutron streaming from the core through the primary pipe to the most exposed part where the outer vessel is attached to the inner vessel. The final dimension of the outer vessel is 3.2 m height and a diameter of 5.2 m. This dimension of the core is of the same range as for a classical SFR. The dimension of the PFBR core is given with a diameter of 6.3 m and a height of 1.7 m for a core with 1200 MW_{th} (Chetal and et al., 2006).

The neutron fluence in the outer vessel can be reduced significantly by the use of a optimized blanket and an absorbing layer, but there is no possibility to do this for the inner vessel (see the neutron fluence distribution given in Fig. 12). There the fission reactions occur directly at the wall, see the power production given

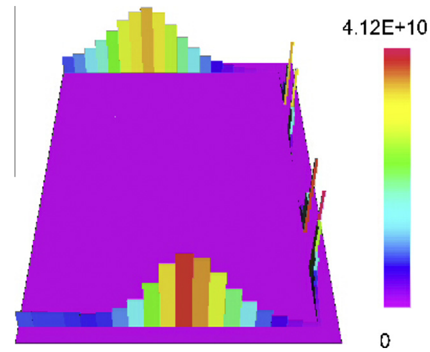

Fig. 9. 2D Neutron flux distribution of neutrons with energy >1 MeV in the outer wall of the system for the optimized case.

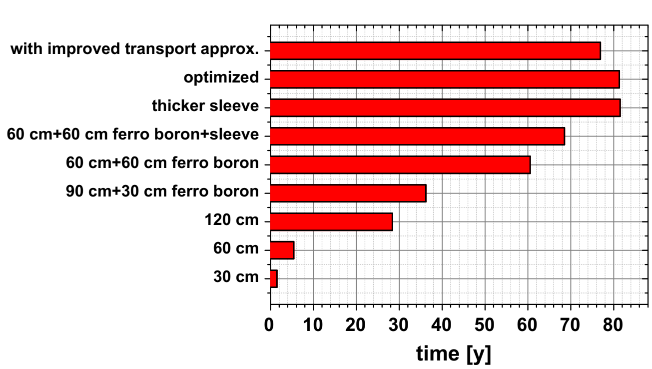

Fig. 8. Time to reach the fluence limit in the outer vessel after stepwise change of the system configuration.

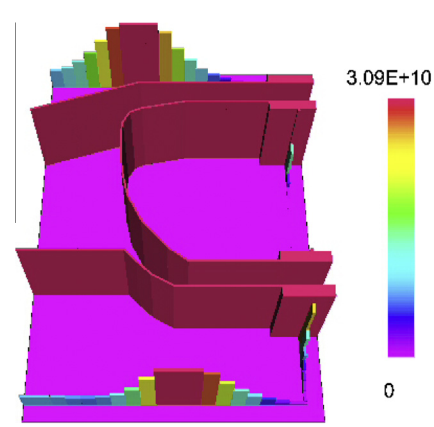

Fig. 10. 2D Neutron flux distribution of neutrons with energy >1 MeV in the outer wall in correlation with the inner structure.

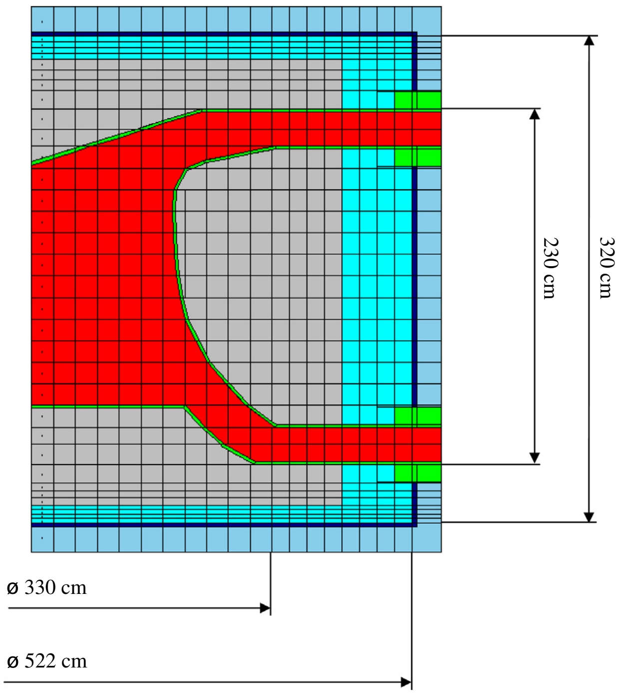

Fig. 11. Model of the optimized geometry for an expected time of 80 years to reach the fluence limit.

in Fig. 5. The problems caused by this configuration have already been discussed. The problem is not changed at all due to the blanket. A literature survey identifies the maximal experienced neutron fluence in an experiment using nickel based material. The test has been performed in the fast breeder test reactor (FBTR) in India as basis for the use of nickel based reflector elements. In this irradiation experiment the nickel material has been exposed to a neutron fluence of $1.09 * 10^{23}$ neutrons/cm^2 (Muralidharan et al., 2011). The post irradiation examination has shown swelling of nickel blocks, condition of the collapsible tube, and dimensional changes in the nickel blocks.

The conclusion of the experiment was:

- Maximum volumetric swelling of the nickel block is 3.6%.
- Radial gap between nickel blocks and wrapper has reduced from 3 mm to 2 mm.
- Residual life of reflector subassemblies can be extended further without any concern on mechanical interaction between nickel blocks & wrapper (Muralidharan et al., 2011).

Based on this data for neutrons above 0.1 MeV a first guess can be given, that the time to reach experience limit is about 4 years. Thus a significant improvement in radiation exposure experience will be required for this component like it is requested for several high exposed parts for all types of fast systems foreseen in GEN-IV.

It has to be kept in mind that all calculations are preformed on a 2D level. Some changes would appear in the power and flux distribution in the walls when a 3D code is used. For the current state of the project the 2D approximation is appropriate to ensure that the shielding problem can be solved and the shielding is not a criterion for exclusion for the whole development of a molten salt fast reactor.

The results of the HELIOS calculations have been confirmed by a comparison with MCNP. Reasonably good agreement has been achieved for a one dimensional deep penetration problem representing a horizontal cut through the center of the system.

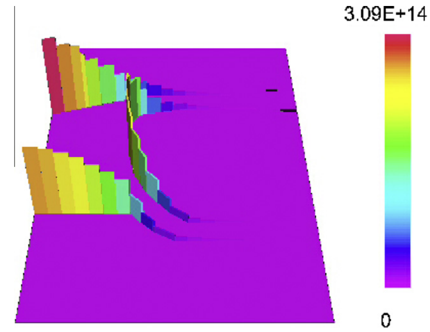

Fig. 12. 2D Neutron flux distribution of neutrons with energy >1 MeV in the outer wall of the system.

The results of HELIOS at the outer wall are around 30–40% higher for the neutron flux. Translating this difference into dimensions, roughly 8% less distance using the same material configurations would be required to achieve the same neutron flux in the outer wall relying on the MCNP calculations than for using the HELIOS results. With a view on the current project status, this difference in completely acceptable if compared with the other uncertainties still immanent in the whole design process.

Moreover the direct irradiation by high energy neutrons leading to material damage is not the only irradiation problem. In nickel based alloys at high temperatures like they appear in a MSR a second path of damage becomes important. This is the embrittlement due to helium formation caused by neutron irradiation and accumulation at the grain boundaries. This process is even caused by low energy neutrons (Angeliu et al., 2006). The only way to solve this kind of problem is to keep the temperature in the outer vessel low (frozen wall technology) (McNeese, 1972). When the frozen wall technology is applied, it could even be possible to use high temperature resistant steel material, since corrosion caused by the frozen salt on the steel will be limited.

6. Conclusions

The molten salt reactor technology has gained some renewed interest which was focused in the EURATIOM project MOST and two recent important projects which have been launched EVOL (EVOL – Evaluation and Viability of Liquid Fuel Fast Reactor, 2012; Renault and Delpech, 2010) and MOSART (Ingatiev et al., 2012,2007). The reactor concepts of these two projects are significantly different than the historic molten salt reactor experiment (MSRE) at the Oak Ridge National Laboratory. Both concepts are proposed as molten salt fast reactor (MSFR) like it has discussed in the generation IV international forum. The problem of shielding the reactor vessel against high neutron irradiation has already been recognized and discussed in the MSRE.

The challenge in the MSFR is even more serious due to several reasons. The power density and thus the neutron flux is significantly higher, the neutron spectrum is harder, the fission neutrons are born close to the vessel, thus there is no space for slowing down the neutrons.

One very interesting and innovative result of the EVOL project up to now is the fluid flow optimized design of the inner reactor vessel using curved blanket walls. The developed structure leads to a uniform flow distribution without using internal flow guiding. A model for this new design has been built for the HELIOS code. The neutron flux and power distributions are discussed and the neutron flux distributions in the inner and outer walls are analyzed. It has been demonstrated with the help of 2D HELIOS calculations that it is possible to build a well blanketed system and to keep the fluence in the outer vessel within the limit of 10^{20} neutrons/cm^2 for a reasonable operation time of 80 years. Nevertheless, the limitations of the 2D modeling have to be kept in mind. The results obtained by HELIOS have been verified against Monte-Carlo calculations with very satisfactory agreement for a deep penetration test problem

With this result it is demonstrated that an outer vessel has to be foreseen in the design, since this outer vessel has to carry the safety related function containing the highly radioactive materials. The outer vessel can be shielded by suited structures and thus the radiation damage does not have the potential to be a show-stopper for the molten salt fast reactor. However, a successful shielding of the fast neutrons increases the size of the MSFR core/blanket system significantly. The dimensions will be of the same level as for a sodium cooled fast reactor. Nevertheless, the neutron fluences at the inner vessel are still a challenging problem like it appears for all highly irradiated components in GEN-IV systems.

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