PHYSOR 2010 – Advances in Reactor Physics to Power the Nuclear Renaissance
Pittsburgh, Pennsylvania, USA, May 9-14, 2010, on CD-ROM, American Nuclear Society, LaGrange Park, IL (2010)

DEFINITION OF BREEDING GAIN FOR MOLTEN SALT REACTORS

K. Nagy, J. L. Kloosterman, D. Lathouwers and T. H. J. J. van der Hagen
Delft University of Technology
Department of Radiation, Radionuclides and Reactors
Mekelweg 15, 2629 JB Delft, The Netherlands
k.nagy@tudelft.nl; j.l.kloosterman@tudelft.nl; d.lathouwers@tudelft.nl;
t.h.j.j.vanderhagen@tudelft.nl

ABSTRACT

The graphite-moderated Molten Salt Reactor (MSR) is a potential breeder reactor using the thorium fuel cycle. The MSR has unique properties due to the possibility of making changes to the salt composition during operation. Most important is the extraction of protactinium, which separates the fissile uranium production into two volumes: the reactor core and the external stockpile. The paper focuses on the definition of breeding gain in such a system. The prospects of using breeding gain expressions defined for solid fuel reactors are investigated and new definitions are given which incorporate the processes occurring in the reactor core and the external stockpile. The difference of the growth rate of the mass of fissile material and breeding gain is pointed out. The new definitions are applied to an optimization study of the graphite-salt lattice of a breeder MSR.

Key Words: molten salt reactor, thorium fuel cycle, breeding gain

1. INTRODUCTION

There is a renewed interest in molten salt reactors (MSR) since the design was chosen by the Generation IV International Forum. The MSR uses fluid fuel which consists of actinide salts and other salts providing a low melting point and good heat transfer properties. The concept of the MSR has advantages on fuel fabrication, neutron economy and high temperature operation. It is also possible to change the composition of the salt during operation by extracting fission products or refueling the reactor on-line. Because of these advantages the MSR is a promising breeder reactor. Our study considers a moderated MSR which is a possible breeder on the thorium fuel cycle.

The research of an MSR running on the thorium cycle was started in the 1960s at the Oak Ridge National Laboratory. The goal of the project was to develop a graphite-moderated breeder reactor (MSBR) [1]. That research proved the feasibility of such a reactor after which the project was cancelled due to fierce opposition with the sodium cooled fast reactor. Recently, the concept was re-assessed in the frame of the MOST project [2]. An extensive study is carried out in France at Laboratoire de Physique Subatomique et de Cosmologie (LPSC), where first a design of a moderated reactor was developed on the basis of the MSBR [3], currently the project focuses on a fast breeder reactor [4].

The thorium fuel cycle works similar to the uranium one. The fertile isotope is ^{232}Th which forms ^{233}U after a neutron capture and 2 beta decays. The in-between steps are ^{233}Th and ^{233}Pa. The important difference between the two cycles is the half-life of the isotopes. ^{233}Pa has a half-life of 27 days, which gives a high chance to capture a neutron before it decays leading to ^{234}U, a non-fissile isotope. To avoid this, it is possible in a MSR to extract part of the protactinium from the salt during operation and to store it outside the core. All the ^{233}Pa removed will decay to ^{233}U. In this way the ^{233}U production can be enhanced. This also means that an external stockpile of protactinium and uranium is formed. Since the amount of protactinium left in the core does not produce enough ^{233}U to keep the reactor critical (the core itself is a converter) a part of the ^{233}U formed in the stockpile has to be sent back into the core to maintain criticality. In other words the core and the stockpile strongly depend on each other, forming one system.

As one wants to maximize the breeding performance of such a reactor, one has to define the proper performance parameters for the optimization, based on the physical processes in this system. In order to estimate the performance of breeder reactors, several parameters were defined for solid fuel reactors, the so called conversion ratios and breeding gains (BG). In this paper we investigate if some of these definitions are applicable for a moderated MSR and give new definitions which accurately describe the long-term fissile fuel evolution of such a reactor. Based on these definitions an optimization study of a breeder core is performed.

2. METHODOLOGY

The calculations presented in this paper are performed with the XSDRN and KENO neutron transport codes of the SCALE code package [5] and an in-house developed burnup code which is able to take into account continuous removal and feed processes as well as the material evolution of stockpiles outside the core.

The BG definitions are investigated and compared on the basis of one MSR case. The reactor is described in Table I. The salt mixture used is FLiBe with 12 mol% of ThF_4. The concentration of UF_4 is continuously adjusted to keep the reactor critical. The reactor is started up with ^{233}U. The same salt mixture is considered for the optimization study. Non-soluble and soluble fission products are removed and protactinium extraction is applied unless stated otherwise. The assumed efficiencies of the removal processes are listed in Table II [6]. In the calculations all the uranium used to refuel the reactor is assumed to be extracted from the external stockpile and all the uranium produced by the decay of the extracted protactinium is first placed in the stockpile. Thorium is continuously fed to sustain a constant concentration of fertile material in the core.

Table I. Parameters of the core

Height (m)	Diameter (m)	Fuel channel diameter (cm)	Volume ratio	Power (MW)	Power density (MW/m3)	Initial U-233 load (kg)
5	5	6	2.75	490	5	630

Table II. Efficiency of extraction [6]

	Gaseous	Non-soluble	Soluble	Protactinium
Efficiency (%/s)	1	$5.8\ 10^{-3}$	$3.172\ 10^{-8}$	$1.269\ 10^{-7}$

3. INTEGRAL BREEDING GAIN DEFINITIONS

Breeding gain and fuel growth-rate definitions are widely used in fast breeder reactor studies [7]. Integral definitions of breeding gain compare the amount of fuel at the beginning and the end of an irradiation cycle. These expressions cannot be used unambiguously since in a MSR no such cycles can be defined due to its continuous fission product (FP) removal and actinide feeding processes. Any other reference points than the first startup and the last shutdown of the reactor is artificial. Simple definitions as the ratio of total mass of uranium at a certain time of operation to the startup, or the ratio of the amount of fissile material produced during operation to the amount of fissile material destructed give misleading results in case of a MSR.

First consider a MSR without protactinium removal. It was shown [8] that depending on the efficiency of FP removal the MSR will get subcritical after some time of operation. This happens because of the build-up of FP and non-fissile uranium in the salt, which changes the ratio between Σ_f and Σ_a even if the mass of ^{233}U is increasing in the core. In other words the reactor produces more uranium than it destroys but this is not enough to maintain criticality.

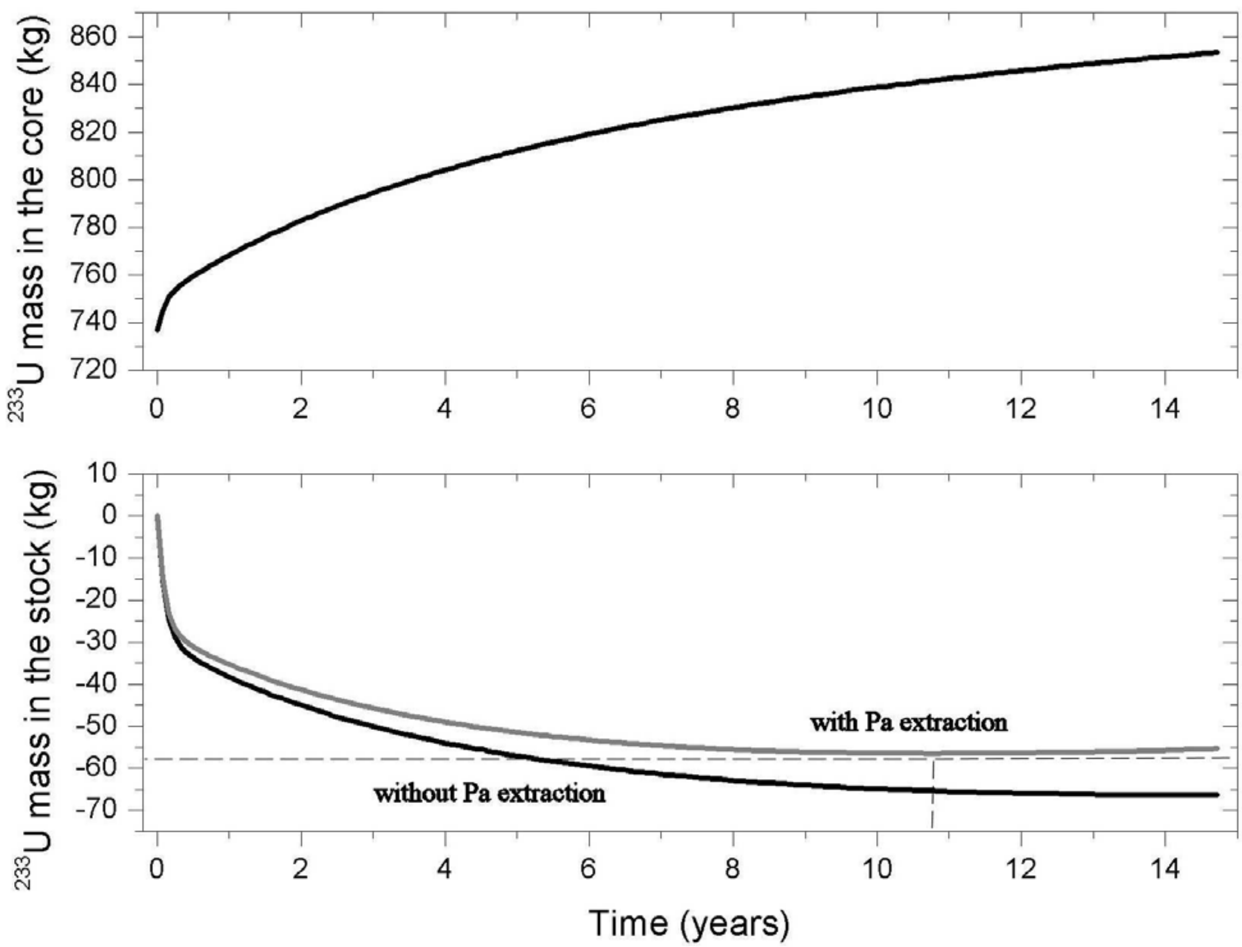

Figure 1. Time variation of fuel mass with and without Pa extraction.
Differences in the mass of ^{233}U in the core are negligible. The efficiency of the removal of soluble FP is set to one third of the original one.

Consider now a reactor with Pa extraction. The MSR is operated with as low reactivity reserve as possible. To compensate the build-up of absorbers, part or all of the ^{233}U formed in the external stockpile is sent back into the core to increase Σ_f to keep the reactor critical. This also means that although breeding occurs and the total amount of fissile material is increasing, the produced uranium can not be considered as excess that can be used in other reactors, because it is needed to maintain criticality of the original reactor. Even more fissile material may be needed than is produced in the core and the stockpile together, so the growth rate of fissile material is lower than the growth rate needed to keep the reactor critical.

The behavior of the reactor without and with Pa extraction is compared on a case considering a reactor described in Table I. To amplify the effect of FP extraction, the efficiency of the extraction of soluble FP is set to one third of the original efficiency listed in Table II. The results are shown on Figure 1. To keep the reactor critical the mass of uranium in the core has to increase continuously to compensate for the build-up of the absorber nuclei. The increase is almost the same in the two cases, the difference is 3 kg after 10 years of operation. This is due to the absorption of neutrons by protactinium left in the core. The reactor with protactinium removal is a converter in the first 11 years and after this time it turns into a breeder, the stockpile of uranium starts to grow. Without protactinium removal the stockpile is monotonously decreasing, the reactor is a converter in this case. In the latter case the uranium mass in the core increased with 116 kg after 15 years but only 66 kg of it was extracted from the external stockpile, the other 50 kg was produced in the core. The increase can not be used in other reactors, it is not excess. It is clear that BG expressions purely based on the change of the mass of fissile material are not suitable for the MSR.

4. WEIGHTED INTEGRAL BREEDING GAIN DEFINITIONS

In fast reactor studies different isotopes are weighted according to their contribution to the reactivity of the reactor. In order to define an integral BG definition using these weights one has to consider a performance parameter *R(t)* based on the material composition of the reactor and weights of these isotopes.

$$R(t) = \left\langle \overline{w}, \overline{N} \right\rangle \tag{1}$$

Here $\overline{N}$ is the time-dependent nuclide inventory of the reactor and $\overline{w}$ is the time dependent vector of the corresponding weights of the nuclides. The bracket indicates the scalar product of the vectors. In a MSR the fuel is continuously mixing in the piping so a homogeneous distribution of nuclides in the core and spectrum-averaged, only time-dependent weights are assumed. On the basis of this parameter BG can be defined as follows [9], basically comparing weighted amounts of the isotopes.

$$BG_{weighted}(t) = \frac{R(t) - R(0)}{R(0)} \tag{2}$$

We take the first startup of the reactor as reference point 0 and weight only the key nuclides of the thorium cycle and the uranium isotopes present in the core. To define the weights the following formula is considered, which gives normalized weights so ^{232}Th and ^{233}U have weight of 0 and 1, respectively.

$$w_i = \frac{\sigma_i^* - \sigma_{02}^*}{\sigma_{23}^* - \sigma_{02}^*}, \tag{3}$$

where the two digit subscripts denote the isotopes with the last digit of their atomic and mass number respectively and σ_i^* for a certain nuclide is defined as the traditional definition of reactivity weights [10]:

$$\sigma_i^* = \nu_i \sigma_{f,i} - \sigma_{a,i} \tag{4}$$

The list of nuclides and the corresponding weights are listed in Table III, averaged for the core. The nuclide inventory of the stockpile is considered in the BG calculation as well. The ^{233}Pa in the stockpile has the same weight as ^{233}U as it is outside of the neutron flux and will decay to uranium without neutron capture.

Table III. Weights of the nuclides

Th-232	Pa-233	U-233	U-234	U-235	U-236
0	-0.284	1	-0.271	0.688	-0.0956

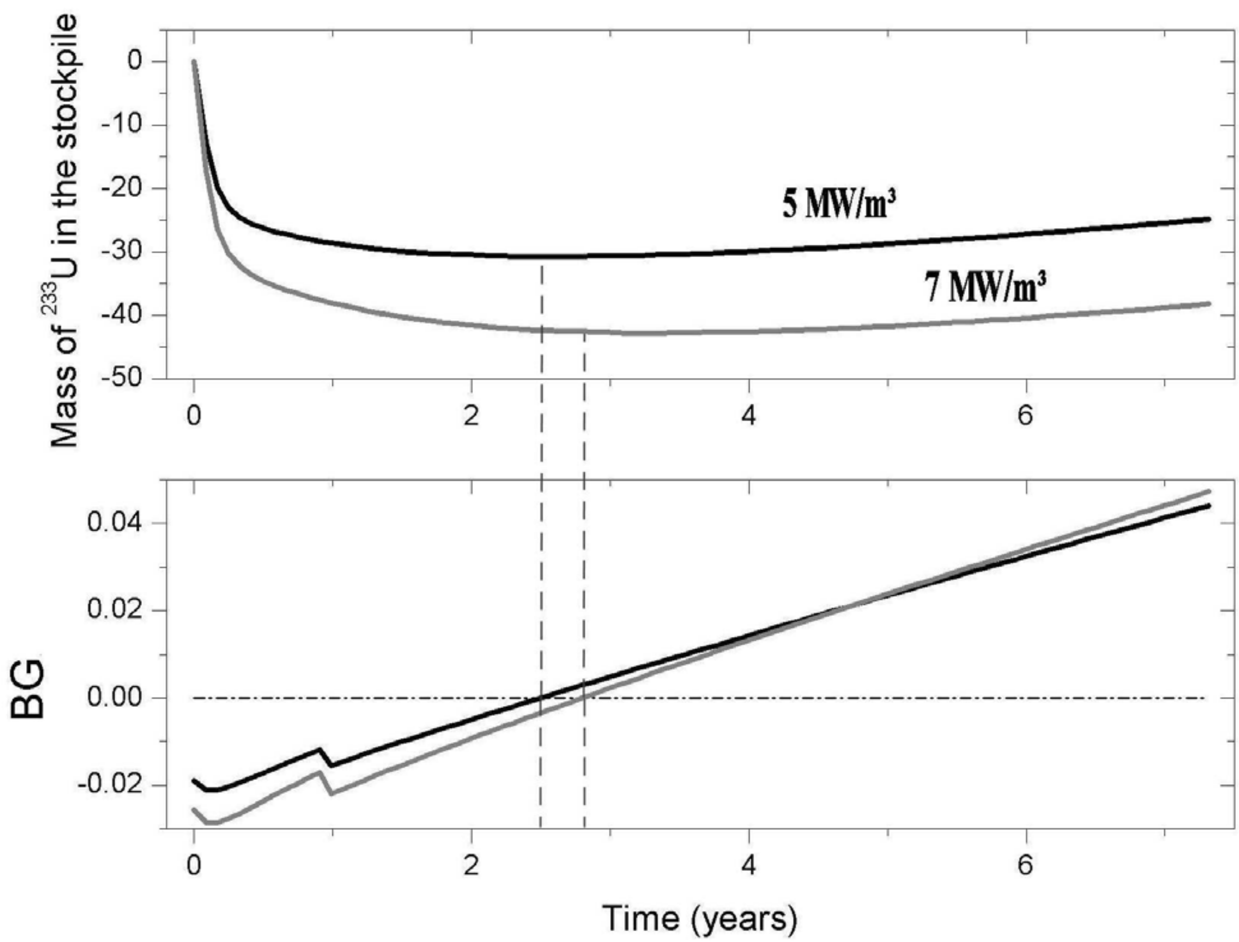

Figure 2. Time variation of the integral breeding gain definition. Black and grey lines correspond to the case with 5 and 7 MW/m³ power density, respectively.

Results of the BG calculation are shown in Figure 2. The geometry of the reactor is the same as used before and listed in Table I. Removal and storage of protactinium is assumed. In this case two different power levels were applied: 5 and 7 MW/m³. This is only an arbitrary choice to show the behavior of this definition. In the BG calculation the external stockpile is normalized to have a starting mass of 0 kg. As long as the mass of uranium is decreasing in the stockpile the reactor consumes more uranium than it produces in the core and in the stockpile together. This way the reactor is a converter. If one would like the BG definition to predict the converter and breeder regimes of the reactor the value of the BG should be negative in the first case and turn into positive when the mass of U-233 in the stockpile starts to grow. In the case of 5 MW/m³ this is the case, the BG calculation properly describes the behavior of the reactor. In the case of 7 MW/m³ power density however, the BG predicts breeding earlier than it actually occurs. The same behavior of the definition can be observed with different reactor setups, the definition can predict the transition between converter and breeder operation with an error up to 6 months. Other weights were tested but these change only the values of the calculated breeding gain, not the predicted point of transition. We can conclude that this definition give a good approximation but does not describe the system properly.

5. DIFFERENTIAL BREEDING GAIN DEFINITIONS

One way to define breeding gain (BG) is based on the reaction rates. Using these, BG can be calculated as the production rate of fissile material over the total destruction rate. This definition seems to suit the MSR since there is no need to define irradiation cycles. Breeding gain calculations must be performed on the core and external stockpile together. The simplest BG definition is the net production of ^{233}U over the destruction of ^{233}U (eq. 5).

$$BG_1 = \frac{R_c^{02} - R_c^{13}}{R_a^{23}} - 1 \tag{5}$$

In the equations *R* denotes reaction rate, subscript *c* and *a* refer to capture and absorption. The numerator of eq. 5 holds only at equilibrium but changing that term to the decay rate of ^{233}Pa gives the actual production. If protactinium removal is applied, both the protactinium inside and outside the core must be accounted for. It is clear that at equilibrium the numerator of eq. 5 should equal the decay rate of all protactinium. During operation the uranium vector – which contained only ^{233}U at startup - changes, after 50 years the composition is 0.678-0.212-0.076-0.034 ^{233}U to ^{236}U by weight, so the definition should be extended with the production and loss of ^{235}U as well (eq. 6).

$$BG_2 = \frac{R_c^{02} - R_c^{13} + R_c^{24}}{R_a^{23} + R_a^{25}} - 1 \tag{6}$$

As mentioned above, the reactor core may need a fissile fuel from the external stockpile to sustain criticality. The problem with these definitions is that they do not account for this. The use of a feeding term in the BR definition, as shown in eq. 7, seems to be useful. In this way the numerator describes the net production of fissile uranium in the core and the net production of ^{233}U outside the core. The subscript *fe* refers to the feeding of the reactor core from the stockpile.

$$BG_3 = \frac{R_c^{02} - R_c^{13} - R_{fe}^{23} + R_c^{24}}{R_a^{23} + R_a^{25}} - 1 \tag{7}$$

Since the amount of fissile material in the core can be considered just enough to run the reactor, BG can be defined on the production and loss rates of ^{233}U outside the core (eq. 8). The subscript *d* refers to decay while the superscript *13_s* denotes the ^{233}Pa in the stockpile.

$$BG_4 = \frac{R_d^{13\_s}}{R_{fe}^{23}} - 1 \tag{8}$$

This definition describes the evolution of the fissile material not used in the core. This part of the produced uranium can be used to start up other reactors. As a drawback, this definition does not describe the growth of the total fissile material inventory.

The aforementioned definitions are compared on a case considering the reactor described in Table I. Removal and storage of protactinium is assumed as well. Results are shown in Figure 3. The external stockpile of ^{233}U is plotted, normalized to a starting stock of 0. It reaches minimum of -30.8 kg after 2.5 years. This time corresponds with the build-up time of protactinium inside the core and in the stockpile. During this period the system (core plus stockpile together) can be considered as a converter. After 2.5 years, more ^{233}U is produced in the stock by the decay of protactinium than sent back to the core so the reactor system is effectively a breeder. The surplus fissile material shows up in the ^{233}U stockpile.

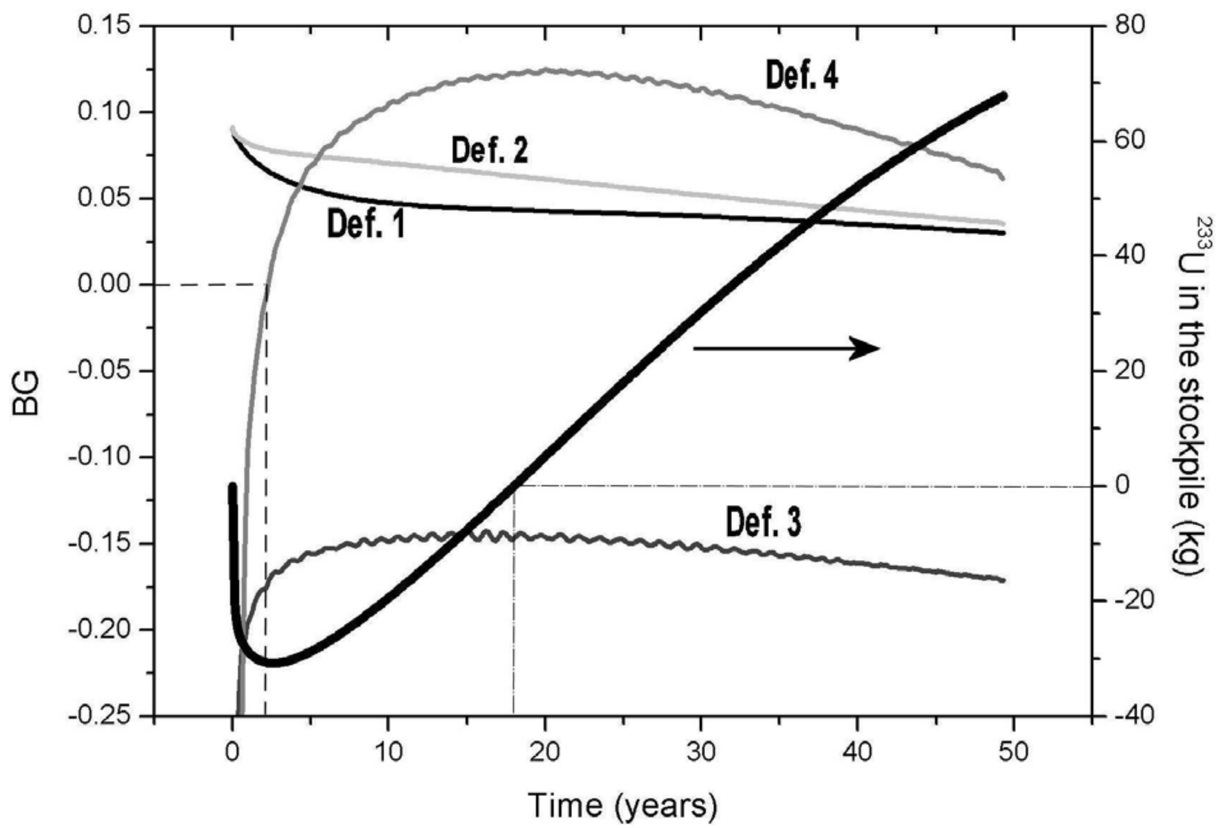

Figure 3. Time variation of differential breeding gain definitions and the mass of uranium in the stockpile. Def. 1, 2 and 3 never change sign so these definitions do not predict the change between converter and breeder operation.

The uranium needed in the converter phase is reproduced 18 years after the startup. The 4 BG definitions are shown on the same figure. Definition 1 and 2 (eq. 5 and 6) clearly don't follow the behavior of the system, they don't predict any converter phase. Since the power density and thus the denominator of the expressions are fixed, they show only the change in capture rate of fertile material. It is clear from the two definitions above that the feeding of ^{233}U should be taken into account, but definition 3 (eq. 7) is negative for the whole time of operation. Counting the feeding as a loss for the whole system does not describe the processes properly. The last definition (eq. 8) follows the evolution of the ^{233}U in the stockpile. It is negative in the converter phase and turns to positive when the stock reaches its minimum. This definition is able to predict the amount of excess fissile material and can be used to determine the doubling time of the reactor.

6. GROWTH RATES FOR CORE AND STOCKPILE

Since eq. 8 describes the net ^{233}U gain in the external stockpile normalized to the feeding rate, a similar mass balance can be established for the core. Eq. 9 gives the change of mass of the fissile material in the core as the net gain normalized to the total destruction.

$$BG_{core\_total} = \frac{R_d^{13} + R_c^{24} + R_{fe}^{23} - R_a^{23} - R_a^{25}}{R_a^{23} + R_a^{25}} \tag{9}$$

Here the capture in ^{232}Th minus the capture in ^{233}Pa cannot be used to express the production of ^{233}U since that equals the decay of all ^{233}Pa. A part of protactinium is extracted from the core and we are interested only in the part remaining in the core. The decay rate of the ^{233}Pa in the core gives the exact ^{233}U production rate. The feeding from the stock is a gain in the mass balance for the core, although that uranium is basically lost from the external stockpile and thus from the excess fissile material. At this point it is worthwhile to change the denominator of eq. 8, normalizing it to the total destruction rate of fissile material, according to eq. 9:

$$BG_s = \frac{R_d^{13-s} - R_{fe}^{23}}{R_a^{23} + R_a^{25}} \tag{10}$$

In this way the two separate definitions can be summed to give the BG related to the change of mass of ^{233}U and ^{235}U in the whole system. In the sum the mass transfer between the core and stockpile vanishes and the expression left almost equals eq. 6:

$$BG = BG_{core\_total} + BG_s = \frac{R_d^{13} + R_d^{13-s} + R_c^{24}}{R_a^{23} + R_a^{25}} - 1 \cong BG_2 \tag{11}$$

Here instead of the capture in ^{232}Th minus capture in ^{233}Pa term the decay rate of ^{233}Pa both in the core and in the stockpile is used. This gives a difference at startup until the equilibrium concentration of protactinium is reached in the core and in the stockpile. It is clear that eq. 6 has a maximum if the ^{233}Pa concentration is 0 while eq. 11 has a minimum in this case. This way eq. 6 suggests a large increase of uranium mass at the startup of the fuel cycle while eq. 11 predicts the actual decrease of mass. We will use one more expression which contains only the processes occurring in the core:

$$BG_{core} = \frac{R_d^{13} + R_c^{24}}{R_a^{23} + R_a^{25}} - 1 \tag{12}$$

In fact these definitions in eq. 9, 10 and 11 can be considered as the derivative of the function describing the time evolution of the mass of ^{233}U and ^{235}U in the core, ^{233}U in the stockpile and the whole system, respectively. In order to do this, the reaction rates have to be weighted according to the masses of the different isotopes. They fulfill the following equations describing the time variation of the total fissile (^{233}U and ^{235}U) inventory [11]:

$$m_{fiss}(t) = m_{fiss}^{core}(0) + m_{fiss}^{stock}(0) + \int_{t=0}^{t} \left(\frac{dm_{fiss}^{core}(t)}{dt} + \frac{dm_{fiss}^{stock}(t)}{dt} \right) dt \tag{13}$$

$$\frac{dm_{fiss}^{core}(t)}{dt} = BG_{core\_total}(t) \cdot R \, ; \quad \frac{dm_{fiss}^{stock}(t)}{dt} = BG_s(t) \cdot R \tag{14}$$

Here the superscripts refer to the reactor core and the external stockpile and *R* is the destruction rate of fissile nuclei in the core, the denominator of eq. 9 and 10. As the reactor is operated at a certain power level, one can consider the fission density to be independent of time. Figure 4 shows the calculated values for these BG definitions. The BG of the core (eq. 9) is always positive as it is continuously fed with ^{233}U from the external stockpile. It is high after the startup, because there is almost no uranium production until the ^{233}Pa reaches its equilibrium concentration in the core but high feeding rate of ^{233}U is needed to maintain criticality. BG_{core} without the feeding term (eq. 12) is always negative, showing that the reactor core itself is only a converter due to the fact that part of protactinium is removed from the core and the ^{233}U produced by that protactinium shows up in the stockpile. The BG of the stock (eq. 10) starts from large negative values since a large amount of ^{233}U is sent into the core after the startup. It is extracted from the external stockpile while the protactinium did not reach its equilibrium amount in the stock. After 2.5 years it turns into positive, from this point the reactor is a breeder. This is also the point where the sum of BG_{core_total} and BG_s is higher than BG_{core_total}.

Eq. 10 describes the time evolution of ^{233}U in the external stockpile and as we assume that the core contains just enough fissile material to make it critical, it is a measure of the breeding performance of the whole system. As eq. 12 considers the production and loss only in the core it is the measure of the breeding performance of the reactor core itself. Eq. 9 and 11 give the growth rate of fissile fuel mass for the core and for the core plus the stockpile as defined in eq. 13 and 14 while eq. 10 gives the growth rate for the stockpile as well. It is also clear now that eq. 5 and 6 do not describe breeding gains but the growth rate of ^{233}U and fissile uranium for the whole system, respectively. Since in this case the growth rates do not describe the actual breeding performance of the reactor, we are going to use eq. 10 and 12 for optimization studies.

Based on the differential BG an integral one can be defined: as eq. 10 gives the BG for the reactor, its integral, the mass of ^{233}U in the stockpile is an integral BG definition. It can not be used for direct comparison of different cases because the initial fuel load of the core may differ. The mass of the stockpile normalized to the mass of the initial load is a proper measure for comparisons.

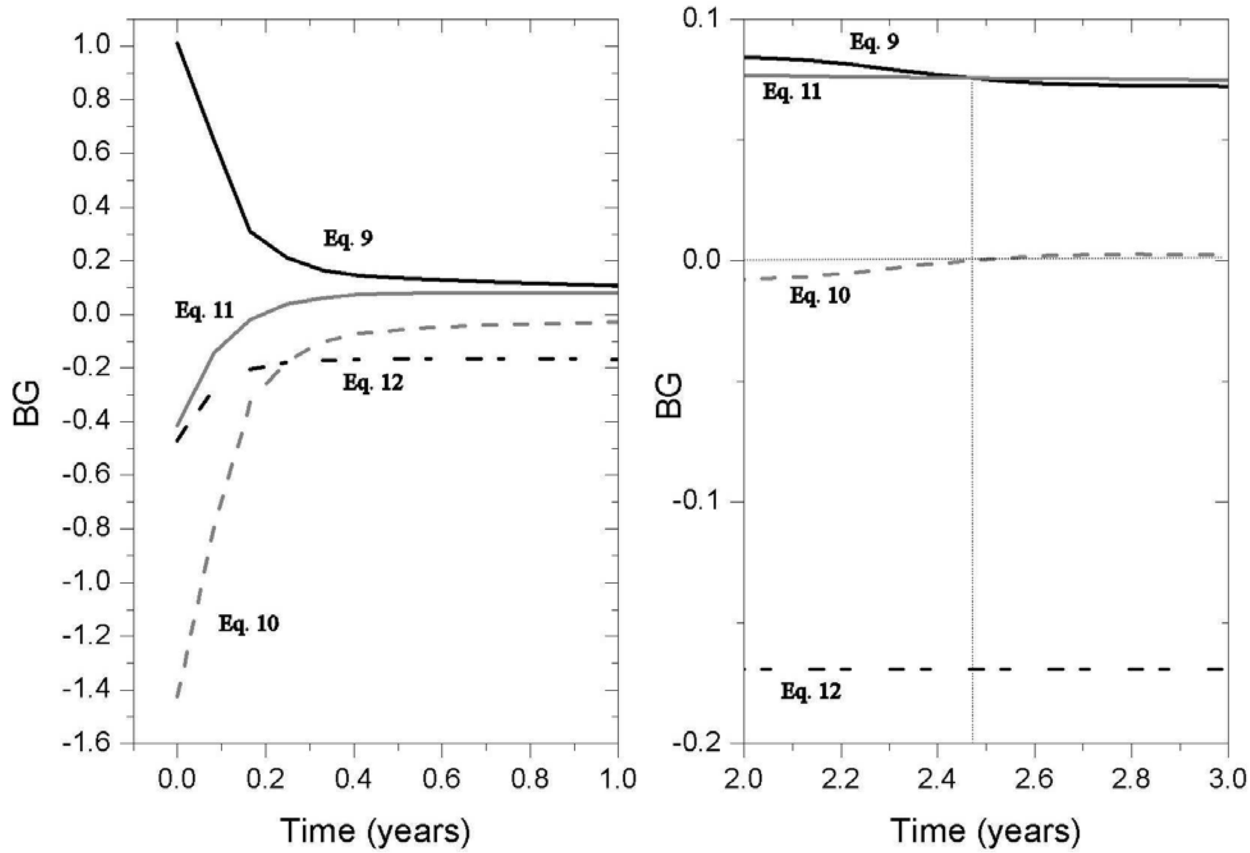

Figure 4. Time variation of breeding gains based on growth rates. The start-up of the reactor and the transition from converter to breeder operation is plotted.

7. OPTIMIZATION OF CORE CONFIGURATION

In the optimization study the graphite-salt lattice is optimized in order to achieve the best configuration for a breeder reactor. The parameter maximized is the breeding gain as defined in eq. 10, other BG definitions, eq. 12 and the integral definition mass of the stockpile normalized to the mass of the initial load is calculated as well. Only the channel diameter and volume ratio are varied; other parameters are considered as boundary conditions. The outer size of the core and the power density are kept constant, the values are the ones listed in Table I. As the parameters vary the number of channels is changed to fill the volume of the core. The efficiencies of the applied chemical processes are listed in Table II.

Before starting the actual optimization one has to have a look on the temperature feedback coefficients because in a moderated MSR positive temperature feedback coefficients can occur depending on the properties of the lattice. The feedback coefficients are computed as follows. First the k_{eff} for a temperature of 1000K is calculated then it is calculated again at a different reactor temperature. Dilatation of the graphite is neglected. For this study the margin of -1.5 pcm/K is set and only those lattices which fulfill this criterion are considered. This margin limits the possible channel diameter and volume ratio combinations which are interesting for this study.

In order to show the behavior of the BG definition in the first case the graphite-to-salt volume ratio is set constant at 2.2 and the channel diameter is varied. The parameter used for this figure is the mass of the stockpile normalized to the mass of the initial load, after 50 years of operation. Results are shown on Figure 5. The gain as the function of channel diameter clearly has a maximum. The best core configuration has a positive feedback. This is true for all volume ratios.

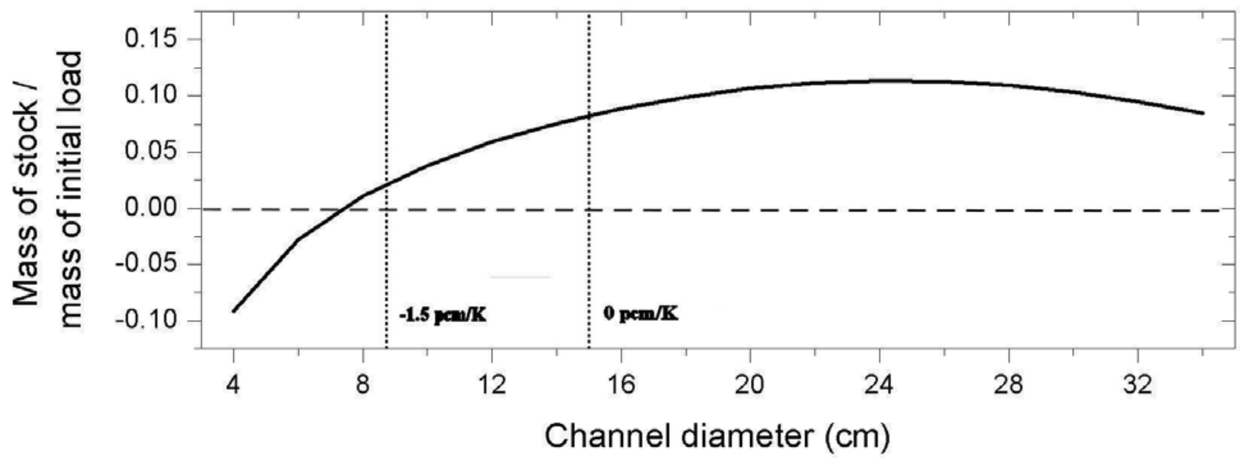

Figure 5. Breeding gain of different channel diameters as the volume ratio is set at 2.2

Table IV shows the possible combinations for breeders within the safety margin. Results are obtained after 50 years of operation, breeding gains are averaged values. The initial loads are assumed to be twice the mass of uranium in the core to account for the salt in the primary loop and the processing plant. The maximized parameter is BG_s (eq. 10) which has its maximum at small channel diameter and relatively large volume ratio. This configuration allows a lower initial load and a better homogenized core than other configurations. The other definitions calculated have their maximum at the same configuration. The core is a converter in each case. Small channel diameters have higher amount of excess uranium although the total increase of the mass of ^{233}U is the lowest in case of these configurations. This is due to the different amount of fuel needed to keep the reactors with different core configurations critical.

Table IV. Results of optimization. First two rows are set parameters.

Channel diameter (cm)	**8**	**7**	**6**	**5**	**4**
Volume ratio	**2.2**	**2.4**	**2.6**	**2.8**	**3.0**
Initial load (kg)	1570	1444	1348	1272	1220
Stock (kg)	8.5	35.3	53.2	65	72.3
Load (kg)	1958	1776	1638	1536	1466
BG_{core} (eq. 12)	-0.175	-0.174	-0.174	-0.173	-0.172
BG_s (eq. 10)	$9.02\ 10^{-4}$	$5.64\ 10^{-3}$	$9.28\ 10^{-3}$	$1.21\ 10^{-2}$	$1.42\ 10^{-2}$
Stock / initial load	0.0055	0.0245	0.0345	0.051	0.0595
Total increase of ^{233}U mass (kg)	396.5	367.3	343.2	329	318.3

7. CONCLUSION AND DISCUSSION

The time evolution of fissile material in the core and in the external stockpile of a MSR with protactinium extraction was shown, and on the basis of this the difference between produced fissile material and excess fissile material in such a reactor was pointed out. A part of the produced fissile material or all of it is needed to compensate the growing neutron absorption of the salt, keeping the reactor critical. Comparison between masses of fuel at different times is not useful because of the increasing inventory of the reactor core. The possibility to use BG defined for solid fuel fast reactors was investigated. The modified version of it – counting for the separate external stockpile – gives approximate results about the behavior of a MSR. Several differential BG definitions were calculated, both counting the inventory of the core and the stockpile as one bulk, and treating them separately. If the definition does not treat the core and the stockpile separately, one obtains the growth rate of fissile fuel mass of the reactor instead of a gain which can be connected to doubling time. Using these definitions one cannot distinguish between converter and breeder operation. New definitions – both differential and integral – were proposed for the reactor core and the external stockpile separately, which describe the breeding gain and the doubling time of the reactor. These definitions can describe the unique behavior of MSR such as the transition from converter to a breeder reactor. In an optimization calculation the best configuration for a breeder MSR was proposed on the basis of the new definitions. The results of the study show that more homogeneous cores produce more excess ^{233}U although the total increase of fuel mass is lower in these cases.

REFERENCES

1. R. C. Robertson, "Conceptual Design Study of a Single-Fluid Molten-Salt Breeder Reactor", ORNL-4541, Oak Ridge National Laboratory (1978)
2. C. Renault et al., "The MOST project: Key-Points and Challenges for the Feasibility of Molten Salt Reactors", *Proceeding of the ICAPP 2005 International Conference,* Seoul, South Korea, paper 5208 (2005)
3. L. Mathieu et al., "The thorium molten salt reactor: Moving on from the MSBR"; *Progress in Nuclear Energy,* **48,** 664-679 (2006)
4. E. Merle-Lucotte et al., "Optimization and simplification of the concept of non-moderated Thorium Molten Salt Reactor", *Proceeding of PHYSOR 2008 International Conference,* Interlaken, Switzerland, paper 355 (2008)
5. SCALE, 2005. "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations, vols I-III". ORNL/TM-2005/39, version 5, Oak Ridge National Laboratory
6. V. Ignatiev, et al., 2007. "Progress in Development of Li,Be,Na/F Molten Salt Actinide Recycler & Transmuter Concept", *Proceeding of the ICAPP 2007 International Conference,* Nice, France, paper 7548 (2007)
7. K. O. Ott and R. C. Borg, "Fast Reactor Burnup and Breeding Calculation Methodology", *Progress in Nuclear Energy,* **5,** 201-235 (1980)
8. K. Nagy et al., "Parametric studies on the fuel salt composition in thermal molten salt breeder reactors", *Proceeding of PHYSOR 2008 International Conference,* Interlaken, Switzerland, paper 277 (2008)
9. W. F. G. van Rooijen, et al., "Definition of Breeding Gain for the Closed Fuel Cycle and Application to a Gas-Cooled Fast Reactor", *Nuclear Science and Engineering,* **157,** 185-199 (2007)
10. M. Salvatores, "Fast Reactor Calculations", *Handbook of Nuclear Reactor Calculations,* CRC Press, Bota Raton, Florida (1986)
11. A. A. Harms, "Fissile Fuel Dynamics of Breeder/Converter Reactors", *Annals of Nuclear Energy,* **5,** 65-73 (1978)