

Article

## Safety-Related Optimization and Analyses of an Innovative Fast Reactor Concept

Barbara Vezzoni \*, Fabrizio Gabrielli, Andrei Rineiski, Marco Marchetti, Xue-Nong Chen, Michael Flad, Werner Maschek, Claudia Matzerath Boccaccini and Dalin Zhang

Karlsruhe Institute of Technology (KIT), Hermann-von-Helmholtz-Platz 1, Eggenstein-Leopoldshafen 76344, Germany; E-Mails: [fabrizio.gabrielli@kit.edu](mailto:fabrizio.gabrielli@kit.edu) (F.G.); [andrei.rineiski@kit.edu](mailto:andrei.rineiski@kit.edu) (A.R.); [marco.marchetti@kit.edu](mailto:marco.marchetti@kit.edu) (M.M.); [xue-nong.chen@kit.edu](mailto:xue-nong.chen@kit.edu) (X.-N.C.); [michael.flad@kit.edu](mailto:michael.flad@kit.edu) (M.F.); [werner.maschek@kit.edu](mailto:werner.maschek@kit.edu) (W.M.); [claudia.matzerath@kit.edu](mailto:claudia.matzerath@kit.edu) (C.M.B.); [dalin.zhang@kit.edu](mailto:dalin.zhang@kit.edu) (D.Z.)

\* Author to whom correspondence should be addressed; E-Mail: [barbara.vezzoni@kit.edu](mailto:barbara.vezzoni@kit.edu); Tel.: +49-721-608-22446; Fax: +49-721-608-23824.

Received: 4 May 2012; in revised form: 11 June 2012 / Accepted: 12 June 2012 /

Published: 15 June 2012

---

**Abstract:** Since a fast reactor core with uranium-plutonium fuel is not in its most reactive configuration under operating conditions, redistribution of the core materials (fuel, steel, sodium) during a core disruptive accident (CDA) may lead to recriticalities and as a consequence to severe nuclear power excursions. The prevention, or at least the mitigation, of core disruption is therefore of the utmost importance. In the current paper, we analyze an innovative fast reactor concept developed within the CP-ESFR European project, focusing on the phenomena affecting the initiation and the transition phases of an unprotected loss of flow (ULOF) accident. Key phenomena for the initiation phase are coolant boiling onset and further voiding of the core that lead to a reactivity increase in the case of a positive void reactivity effect. Therefore, the first level of optimization involves the reduction, by design, of the positive void effect in order to avoid entering a severe accident. If the core disruption cannot be avoided, the accident enters into the transition phase, characterized by the progression of core melting and recriticalities due to fuel compaction. Dedicated features that enhance and guarantee a sufficient and timely fuel discharge are considered for the optimization of this phase.

**Keywords:** fast reactors; sodium void effect; severe accident; recriticality

---

## Abbreviations

CDA: core disruptive accident	CRs: control rods
ULOF: unprotected loss of flow	CSD: control and shutdown device
MAs: minor actinides	DSD: diverse shutdown device
SFR: sodium cooled fast reactor	SASSs: self-actuated shutdown systems
CMR: controlled material relocation	USS: upper steel structure
BOL: beginning of life	SVRE: sodium void reactivity effect
XSSs: cross sections	GEMs: gas expansion modules
SAs: subassemblies	EOC3: end of cycle 3
UAB: upper axial blanket	EOI: end of irradiation
UGP: upper gas plenum	CRGT: control rod guide tube
LAB: lower axial blanket	PSI: Paul Scherrer Institute
LGP: lower gas plenum	KIT: Karlsruhe Institute of Technology

## 1. Introduction

For the long term nuclear energy sustainability, the transition to a fast reactor based fleet and the adoption of closed fuel cycles is envisaged, as indicated by several international studies [1–3]. In fact, the adoption of fast systems has positive effects both for resources optimization and waste reduction [4–6].

The fuel cycle actually implemented is a once-through cycle based on uranium consuming reactors with thermal spectrum. In this cycle, only ~1% of the uranium extracted is utilized and the spent fuel is sent to disposal without reusing the fissile material (e.g., Pu239). This leads to a high demand on disposal capacities in terms of masses, radiotoxicity and heat load.

Due to specific features of fast spectrum reactors, fuels containing a fraction of minor actinides (MAs), can be loaded into their cores and closed (or partially closed) fuel cycles can be implemented thus providing an option for MAs transmutation [4,5,7,8]. Advances in fuel cycle and sustainability should not lead to a lower safety level.

In order to address these goals [2], a new generation of fast reactors has to be designed where safety and competitiveness are the main issues. Among the several fast reactor concepts actually considered in Europe [9], the sodium cooled fast reactor (SFR) is associated with the broadest experience in Europe, where seven operated experimental and prototype reactors operated (including Phénix and Superphénix in France) in the past.

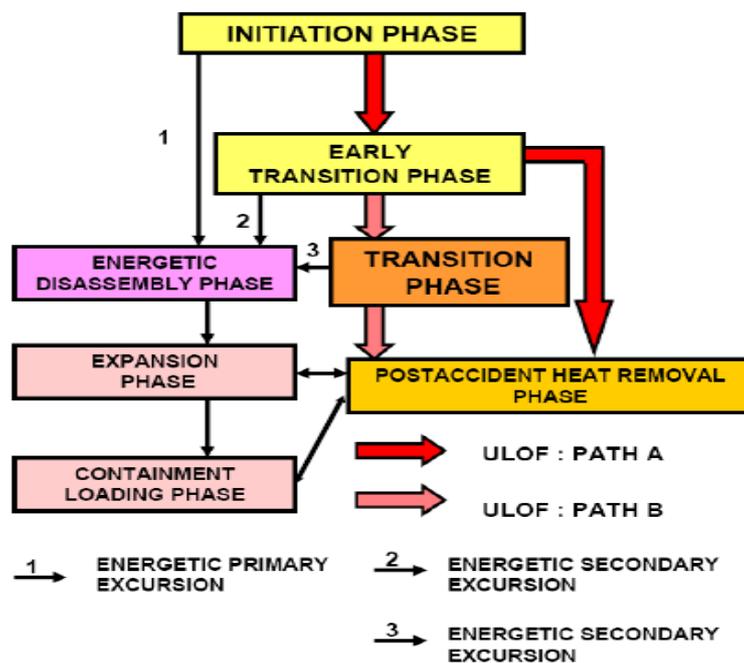
Recently France has launched an initiative for the construction of an advanced SFR prototype, ASTRID by 2020 [10]. Within the European framework, an industrial size (3600 MWth) advanced SFR is being investigated in the CP-ESFR project [11]. After the reference design has been established, foremost efforts will be oriented to improve the safety of the system.

In this paper, we analyze an innovative fast reactor concept developed within the CP-ESFR European project [11] focusing on phenomena affecting the initiation and the transition (to full core melting) phases of an accident due to unprotected loss of coolant flow (ULOF). The experience gained in the past by numerous in-pile and out-pile experiments performed, e.g., [12,13], has shown that

studies of hypothetical Core Disruptive Accidents (CDAs), where fuel may compact and prompt criticality which may be achieved, are important for the safety assessment of SFRs. These studies have considerably improved the understanding of the phenomena related to the fuel failure and fuel relocation under accident conditions.

Traditionally in analyzing severe CDAs, the accident evolution is broken down into different phases (see Figure 1) distinguished by a set of several physical key processes which evolve during accident progression, e.g., due to electrical break-down accompanied by non functioning of the available shut-down systems. Coolant flow reduction after some seconds leads to sodium temperature increase up to the saturation level (boiling onset). Due to the positive sodium void effect, a CDA with a primary core power excursion is initiated and generalized core degradation occurs. The formation after the primary excursion of medium and large scale molten pools may occur (transition phase) and then re-compaction phenomena may potentially generate secondary power excursions (recriticality events) that lead to energetic disassembly and expansion phases of the fuel/steel pool or sodium vapor bubble [14–16] endangering the integrity of the reactor pressure vessel and/or even of the reactor containment.

**Figure 1.** Typical phase diagram for core disruptive accidents (ULOF transient).



Safety researches are traditionally oriented to the initiation phase and in particular to the sodium void worth reduction in order to control the energetic potentials of the accident [17–19]. However, new solutions to “control” the later accident phases introducing design measures that enable a Controlled Material Relocation (CMR) have been studied. These features aim at avoiding (or limiting) recriticalities by a sufficient and timely fuel discharge (beyond the natural removal path usually not sufficient to prevent recriticalities) that influences and ‘brakes’ the recriticality path [12,20–22].

The results obtained by past studies have been applied to the ESFR model considered in the current paper. Preliminary results have been presented in [13,23,24].

Emphasis has been devoted to the initiation phase by slightly modifying the design for reducing the Beginning of Life (BOL) positive void worth of the system. The selected measures help in reducing the void worth by about \$2–3 through increasing the leakage term, without affecting the other characteristics of the systems, e.g., the power distribution or the breeding capability [23,24].

In order to further reduce the void worth, other solutions have been investigated and introduced in the framework of the CP-ESFR project by [25]; e.g., adoption of pins of diluents or empty pins in each subassembly. These attempts have a large impact on the other parameters (mainly power distribution) but they have been considered because some advantages (preferential path for corium relocation) are expected. For this reason in the paper, a special section is dedicated to the investigation of potential effects during the late phases of the accidents.

For this purpose, SIMMER-III analyses have been performed for a slightly different SFR model using 19 empty pins implemented into each fuel subassembly. The obtained results are briefly summarized here; more details concerning the transition phase studies are reported in [13,23].

In order to improve the transmutation performance of the system, a few percent of MAs can be loaded in the core adopting a homogeneous or heterogeneous (blankets) strategy. The effects on void worth concerning loading of MAs have been investigated in the paper. Preliminary results were presented in [24].

Neutronic analyses have been performed by means of the deterministic ERANOS code systems [26] using JEFF3.1 data library [27]. The effective neutron cross-sections (XSs) have been processed by means of the ECCO cell code by employing actual geometries and fine-group energy structure (1968 groups). The 1968 energy groups effective XSs have been collapsed to 33 groups XSs for the flux calculation. The neutron flux distribution has been calculated using the TGV/VARIANT code embedded in the ERANOS code [28,29].

The trends obtained at BOL have been confirmed by preliminary analyses performed by means of the SIMMER-III code systems [30]. The SIMMER-III (2D) and SIMMER-IV (3D) [31] multi-physics code systems have been developed primarily to analyze transients and accidents in fast reactors with liquid metal cooling (LMFRs) and actually are used as reference codes for severe accident simulation.

## 2. Model Description

The model considered has been proposed within the European Project CP-ESFR [11]. Within the project, two 3600 MWth core designs, loaded with oxide and carbide fuels, respectively, were proposed by CEA, France [11]. Both cores show positive void worth at BOL.

The activity described in the present paper is oriented to improve the oxide core characteristics by employing measures to reduce the positive void worth without affecting the other characteristics of the system. Preliminary studies have shown that the same measures can also improve the carbide core characteristics [25].

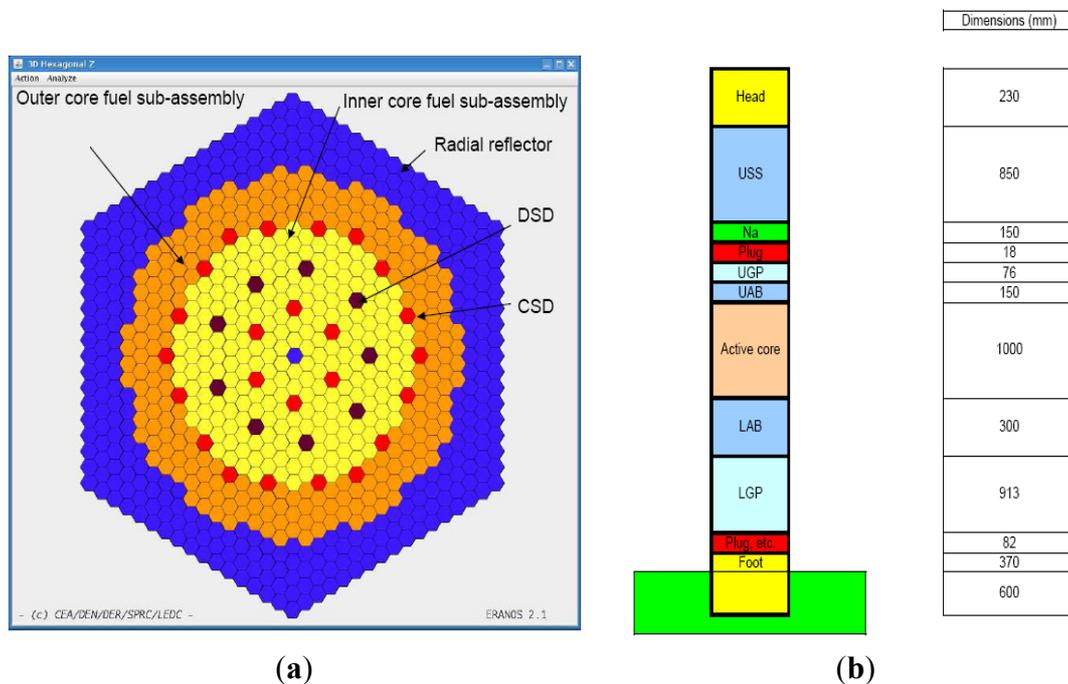
The oxide core layout is presented in Figure 2a and it will be referred to in the following as reference configuration or REF.

The core is composed of 453 fuel subassemblies (SAs) subdivided into two zones in order to flatten the core power profile in the equilibrium cycle. The average Pu content is 14.5% wt. for the inner zone and 16.9% wt. for the outer zone. The active height is 100 cm.

Above the active zone, an upper axial blanket (UAB, 7.6 cm height) and upper gas plenum (UGP, 15 cm height) are placed. Just above the gas plenum, there are plugs, a 15 cm height Na plenum zone, and the upper steel structure (Figure 2b). The lower part is composed of a lower axial blanket (LAB, 30 cm height) and a lower gas plenum (LGP, 91.3 cm height). In the reference configuration the blankets are made of steel in order to improve the reflection of neutrons towards the core.

The average burn up of 100 GWd/tHM is reached after 2050 equivalent full power days (efpd) for a power density of 206 W/cm<sup>3</sup>. In Figure 2a, the positions of the nine Diverse Shutdown Devices (DSD, containing B<sub>4</sub>C with 90% of <sup>10</sup>B) and of the 24 Control and Shutdown Device (CSD, containing natural boron carbide) are also indicated. In the calculations, the CRs are considered withdrawn and the follower (wrapper with Na inside) has been modeled for the active height part.

**Figure 2.** ESRF-OXIDE reference configuration [11]: (a) core layout (ERANOS model); (b) axial structure.



Two figures of merit are considered: the core sodium void reactivity effect (SVRE) and the extended SVRE. The SVRE refers to the voiding of the total active height for the inner and outer fuel zone and the extended SVRE refers, in addition to the active height, to the voiding in the above structures (UGP, UAB, plugs and Na plenum regions). For both figures only voiding inside the wrapper is considered (*i.e.*, sodium between SAs is not removed).

For the reference configuration, both figures of merit are positive: +1532 pcm for the SVRE and +1211 pcm for the extended SVRE. These positive values lead to a reactivity jump up at the beginning of the transient, as shown later on by the SIMMER study.

The positive void effects (+1211 pcm SVRE) indicated above can be partially compensated by other negative feedbacks, as the Doppler constant (*ca.* -1239 pcm) and axial and radial thermal expansion (*ca.* -200 pcm). In the ERANOS model, we considered an axial thermal expansion coefficient of 1.00566 (driven by the clad assuming  $\Delta T$  from 20 °C to average coolant temperature, 470 °C) and a radial expansion coefficient of 1.00666 (driven by the diagrid assuming  $\Delta T$  from 20 °C to inlet coolant

temperature, 395 °C). In the present study, the negative feedback effect due to control rod drive line expansion is not included along with design features and measures that might mitigate transients and facilitate passive reactor shut-down, like self-actuated shutdown systems (SASSs), such as flow levitated absorbers, Curie-point latches or gas expansion modules (GEMs) as indicated e.g., in [32], have not been considered in the present study because they are beyond the aim of the project [25].

### 3. Initiation Phase: Measures for Void Worth Reduction

According to the literature (e.g., [17–19]), the measures suggested and studied in the past for reducing the positive SVRE can be subdivided into two main categories: (1) measures oriented to increase the neutron leakage under voided conditions and (2) measures oriented to soften the neutron spectra (e.g., by inserting diluents).

Within the CP-ESFR project, a parametric study has been performed by comparing different measures from the two categories mentioned above. In order to propose an optimized configuration (referred to in the following as CONF-2) the measures selected are the ones that mostly reduce void effect at the BOL without changing the other core parameters. Preliminary activities are summarized in [23,33].

For reducing the SVRE, the most effective way is to increase the leakage term under voided conditions by modifying the region above the core [23,34]. The adoption of a larger Na plenum with an absorber layer of B<sub>4</sub>C above (in order to reduce the neutron backscattering to the core from the upper reflector, USS) can significantly reduce the extended SVRE as also indicated by [23,33,34]. In order to further increase the leakage term, the Na plenum is shifted close to the core by eliminating the UAB and by reducing the UGP height.

For further increasing the leakage term, the LAB has been replaced by a fertile blanket (depleted U oxide). Therefore, under voided conditions the neutrons are not reflected back to the core but they are absorbed by U238. The use of the fertile blanket also improves the breeding performance of the system.

As reported in [24], a small fraction of AmO<sub>2</sub> can be added to the lower axial fertile blanket in order to degrade the produced Pu vector without affecting the void effect. According to [35], from the proliferation resistance point of view, it is advisable to have a Pu238 content in the discharged material larger than 12% wt., a value that can be reached by loading 5% vol. of AmO<sub>2</sub> in the blanket [24,25].

In particular, for the optimized CONF-2 configuration, the Na plenum height has been increased from 15 to 60 cm (value confirmed by the parametric studies performed by PSI with different Na plenum heights [33]), the UAB removed, the UGP reduced from 7.6 to 5 cm, and an absorber layer (natural B<sub>4</sub>C with same geometrical structure of the above structure) of 30 cm height has been added.

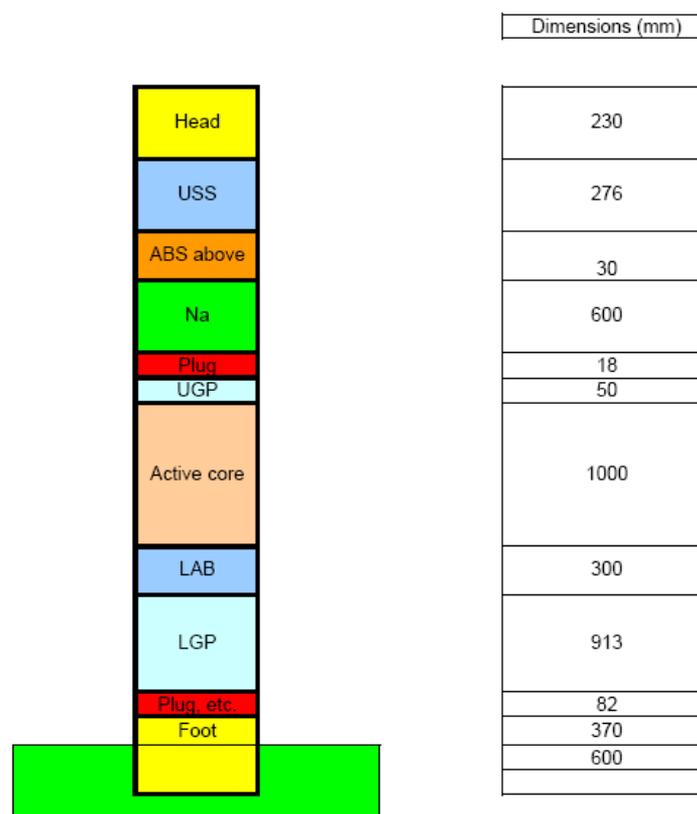
The core layout is the same as in Figure 2a. The axial structure of the optimized CONF-2 is represented in Figure 3 and the design modifications mentioned above can be identified by comparison with Figure 2b.

In order to maintain the criticality level as the reference configuration at BOL (assuming maximal acceptable variation of 200 pcm) and therefore to compensate the leakage under full Na conditions, the Pu content has been homogeneously increased in the inner and outer core zones. For CONF-2, the Pu content has been set up to 14.76% wt. and to 17.15% wt. for the inner and outer core, respectively.

The measures we have considered help in reducing the BOL extended SVRE by about \$2–3 ( $\beta_{\text{eff}} = 393$  pcm) going to +1211 pcm for the REF configuration to +496 pcm for the CONF-2 configuration.

The SVRE is also reduced but not significantly (ca. 100 pcm) if compared to the REF case. The results are summarized in Table 1. For CONF-2, results in Table 1 also show the deterioration of void worth and Doppler with the burn-up. After 1230 efpd (at End of Cycle 3, EOC3) the SVRE is increased by 600 pcm and the extended SVRE by 750 pcm, remaining slightly below the extended SVRE at BOL for the REF configuration. In the study, we have considered EOC3 composition (*i.e.*, the composition after three irradiation cycles assuming reshuffling every 410 efpd) in order to quantify the deterioration of safety coefficients for the equilibrium composition (EOC3 can be considered a composition representative of beginning of equilibrium cycle [25]).

**Figure 3.** ESRF-OXIDE optimized CONF-2 configuration: axial structure.



As further comparison, a 3D MCNP model has been assessed for CONF-2 [36] using same data library, temperatures, expanded dimensions, and compositions as in ERANOS. As shown in Table 1, the MCNP results are in good agreement with the ERANOS results, as expected from previous studies [37].

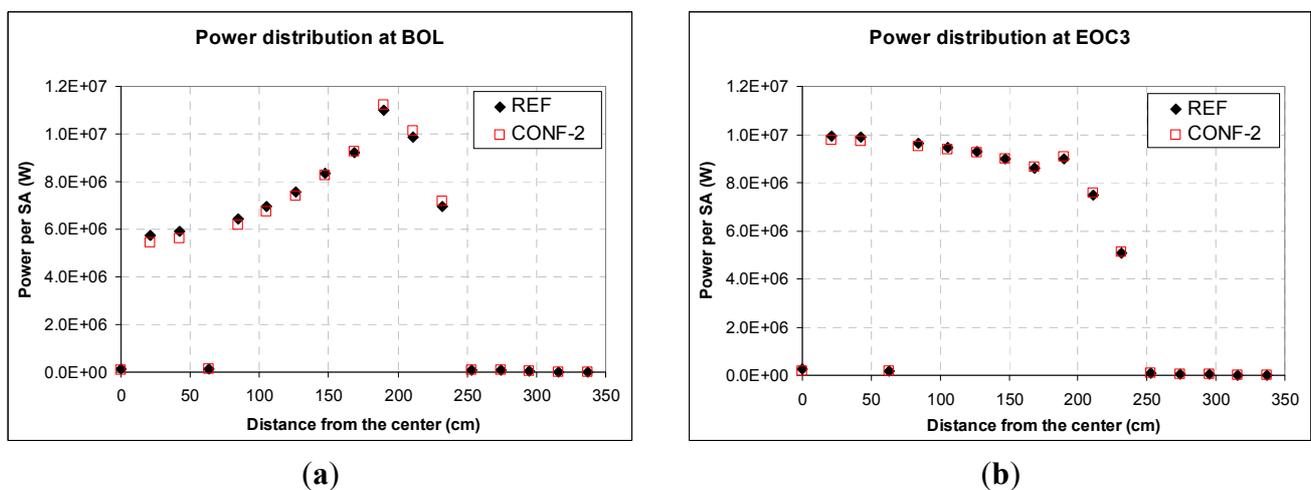
The measures considered do not affect the power distribution. Results show that the radial power distributions at BOL (Figure 4a) and EOC3 (Figure 4b) for REF and CONF-2 are in good agreement.

The use of a lower fertile blanket leads to a less pronounced reactivity swing as indicated in Figure 5. The  $\Delta k$  between BOL and the end of irradiation (EOI) is about 2500 pcm and about 1960 pcm for REF and for CONF-2, respectively.

**Table 1.** Void and Doppler effects for REF and CONF-2.

	REF	CONF-2		
	BOL	BOL	EOC3	
	ERANOS	ERANOS	MCNP	ERANOS
	pcm			
SVRE	+1532	+1423	+1365 ± 60	+1951
Extended SVRE	+1211	+496	+422 ± 60	+1170
Doppler constant, $K_D$ (pcm)	-1239	-1158	-	-843
$k_{eff}$	1.00930	1.01141	1.01264	1.00414

**Figure 4.** Radial power distribution comparison: (a) at BOL; (b) at EOC3 (1230 efpd).



**Figure 5.** Reactivity swing for REF and CONF-2.

The CONF-2 configuration has some margins for improvement for what concerns the void effect reduction. One possibility is the adoption of attempts that make the neutron spectrum softer under voided conditions. This kind of approach has been proposed and developed in the past within the CAPRA/CADRA project [38] where several B<sub>4</sub>C diluents elements were placed in the core (22 SAs).

The use of diluent material, as B<sub>4</sub>C or empty pins (pin filled only with He), has been proposed also for the mitigation part of the later accident phases [22,39], because it enhances and guarantees a sufficient and timely fuel discharge. Some details are summarized in the later part of the paper.

In agreement with other studies [40], other options have been considered in the project for reducing the BOL void reactivity effects [25] such as the introduction of an internal fertile blanket or the variation of the height over diameter ratio (H/D). These modifications reduce the Na void reactivity effects (as studied in the past [41,42]) but they have a large impact on the other core parameters (e.g., power distribution), therefore, it has been decided not to include them in the present study [25].

In order to confirm that the proposed core modifications influence the core behavior under transient conditions in the expected direction, preliminary analyses of an Unprotected Loss of Flow (ULOF) transient for the ESRF reference and for the CONF-2 configurations have been performed at KIT by means of the SIMMER III ver. 3D code [30,31]. The ULOF transient has been simulated in SIMMER considering a coolant mass flow halving time constant of 10 seconds. Since for SIMMER calculations, the mass flow can not be specified as input, because it is a calculated result as a response to pump head coast down in order to produce an input for SIMMER simulation. The coast down curve of the pump pressure head is defined as the following in the scheme  $\Delta p = \Delta p_0 / (1 + t / t_{1/2})^2$ , where  $\Delta p$  is the pump pressure head,  $\Delta p_0$  is  $\Delta p$  at the nominal condition,  $t$  is the transient time and  $t_{1/2}$  is mass flow rate halving time ( $t_{1/2}$  is equal to 10 s for the present design With coolant velocity,  $v$ , and coolant mass flow being proportional to  $p^{1/2}$ ).

As indicated in Figure 6a, SIMMER results for the CONF-2 core show that the reactivity drops down when Na starts to boil (*ca.* 31 s after pump trip) while the reactivity jumps up for REF configuration. This behavior is qualitatively in agreement with the results obtained with ERANOS. However differences are expected due to the different models adopted (3D HEX-Z in ERANOS, 2D RZ in SIMMER), different energy groups (1968 in ERANOS and 18 in SIMMER), and the different transport equation solvers (TGV/VARIANT in ERANOS and TWODANT in SIMMER). In addition the thermal expansion contribution has not been simulated in SIMMER.

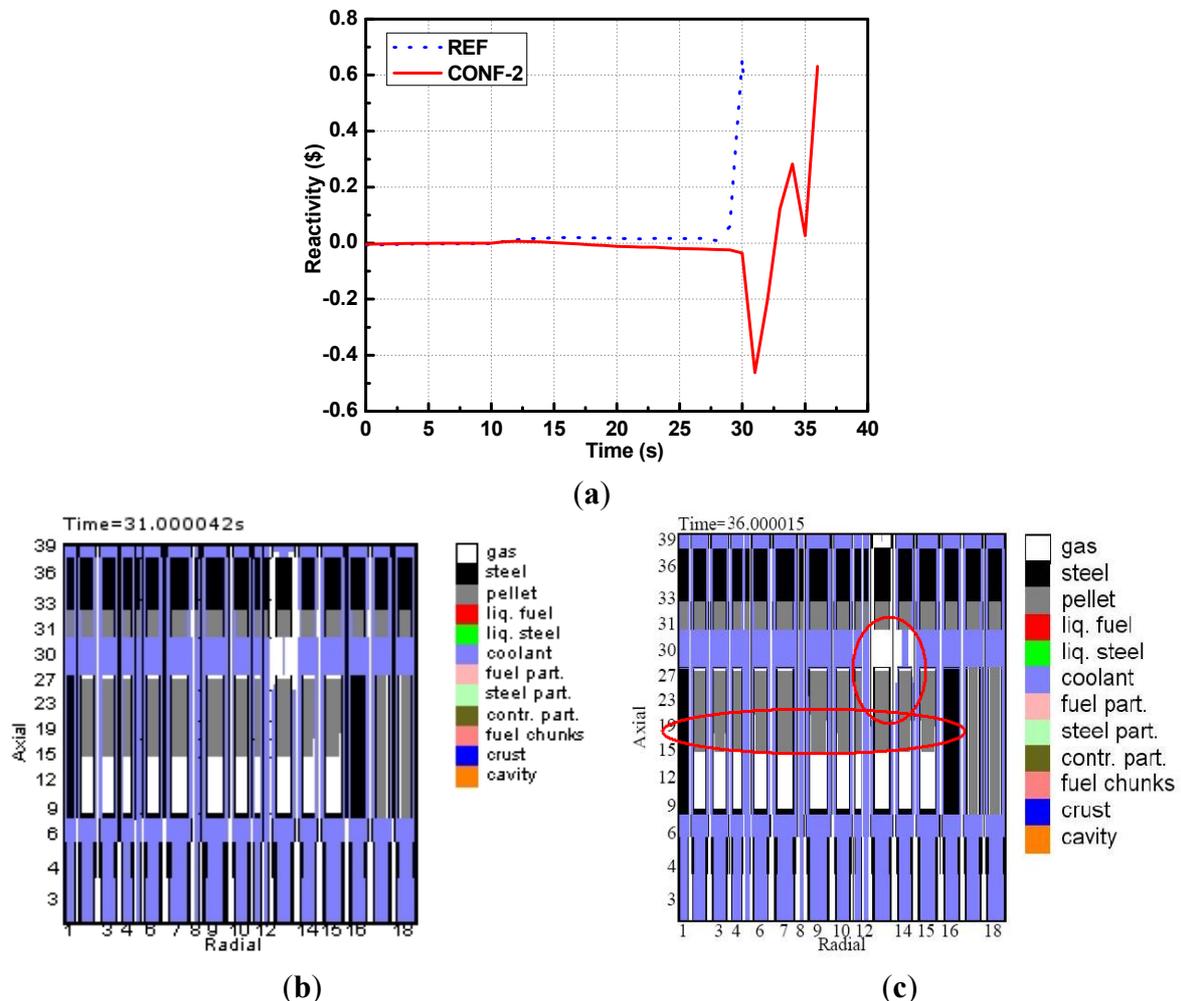
It is also interesting to underline that, in accordance with the power distribution shown in Figure 4a, the boiling onset happens, at time *ca.* 31 s, in the hottest channel (ring 13 in Figure 6b) and it expands initially in the Na plenum zone as indicated in Figure 6,b. However, when the boiling spreads axially and radially into core regions, as shown in Figure 6c, also for CONF-2 the total reactivity (all contributions except thermal expansion are considered) jumps up but it remains well below \$1. In Figure 6-b and Figure 6-c, the material distribution, i.e. the volume fraction of different materials and not the geometrical arrangement are represented as obtained by SIMMER.

The preliminary SIMMER results confirmed the better behavior of the CONF-2 configuration compared to the reference model (In the SIMMER model, constant orifices coefficients have been applied for the present study). The CONF-2 design shows potential for additional optimization in terms of void. Some proposals are under discussion for the following in the CP-ESFR project. Similar results have been obtained by other partners, e.g., [43].

In particular, the low extended SVRE at BOL offers the opportunity for introducing and burning MAs in the system, considering either homogeneous (small content of MAs loaded in core) or

heterogeneous (higher content of MAs loaded in the radial blanket) strategies. These options have been studied and the results are encouraging in terms of core behavior and fuel cycle [4,24,44,45].

**Figure 6.** ULOF transient: (a) Total Reactivity variation *versus* time in REF and CONF-2; (b) Material distribution in CONF-2 at sodium boiling onset (*ca.* 31 s after ULOF).



The addition of Am (different contents) in the core has been preliminarily analyzed at Karlsruhe Institute of Technology (KIT) [24]. In the next paragraph, a summary of the main results obtained is shown.

#### 4. Homogeneous Am loading in the Core

In order to study the effect on safety parameters of the introduction of homogeneous loading of MAs, the CONF-2 model has been considered as reference. Different contents of Am have been homogeneously introduced in the core and in the lower axial blanket. For the study, only Am has been considered because it is the main MA isotope which is accumulated in the cycle. In addition, the transmutation of Am isotopes (mainly Am241) together with Pu has some positive effects in terms of long term radiotoxicity and heat load [4,45].

Two core configurations have been compared with the CONF-2 model:

- 1) the CONF-2 (2%) configuration, where 2% wt. Am (Am241:Am243, 76%:24%) has been introduced in the lower axial blanket and ~1.9% wt. Am has been homogeneously loaded in the core;
- 2) the CONF-2 (4%) configuration, where 4% wt. Am (Am241:Am243, 76%:24%) has been introduced in the lower axial blanket and ~3.8% wt. Am has been homogeneously loaded in the core.

The safety parameters (sodium void worth and Doppler constant), the kinetics parameters ( $\beta_{\text{eff}}$  and  $\Lambda$ ), and the reactivity and heavy metal (HM) mass variation have been evaluated *versus* burn-up.

In Table 2, the computed SVRE, the extended SVRE, and the Doppler constant are shown at BOL and EOC3. The results show that the homogeneous introduction of Am in the core leads to a deterioration of the sodium void worth with respect to CONF-2. As expected the Doppler constant is also deteriorated.

**Table 2.** Void and Doppler effects for configurations considered.

	CONF-2		CONF-2 (2%)		CONF-2 (4%)	
	BOL	EOC3	BOL	EOC3	BOL	EOC3
	<b>pcm</b>					
<b>SVRE</b>	+1423	+1951	+1636	+2029	+1821	+2104
<b>Extended SVRE</b>	+496	+1170	+781	+1290	+1031	+1407
<b>Doppler constant, <math>K_D</math> (pcm)</b>	-1158	-843	-904	-785	-712	-600
<b><math>k_{\text{eff}}</math></b>	1.01141	1.00414	1.00963	1.01330	1.00796	1.02105

The SVRE and the extended SVRE for CONF-2 (4%) are 400 pcm and 500 pcm, respectively, which are larger than the values at BOL for CONF-2 but comparable with the corresponding values for the REF configuration (Table 1).

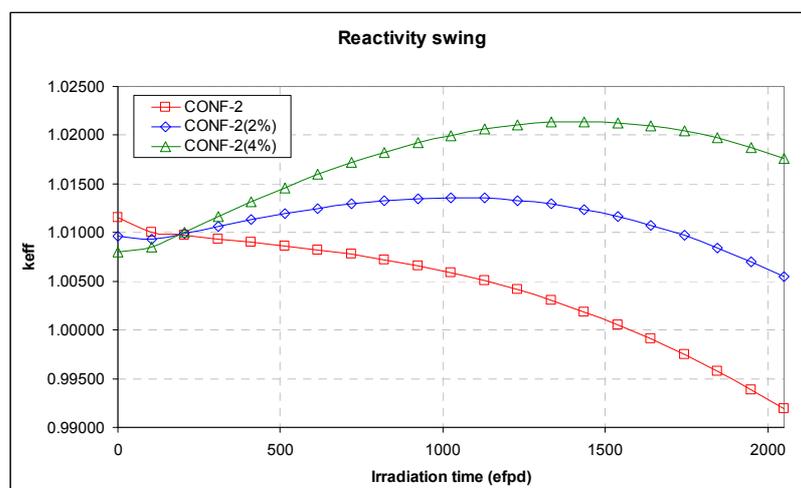
The effective delayed neutron fraction ( $\beta_{\text{eff}}$ ) and the mean neutron generation time ( $\Lambda$ ) have been evaluated at BOL by employing an extended ERANOS version developed at KIT. Results (Table 3) show that both kinetics parameters decrease as the Am content in the core increases, as expected.

**Table 3.** Kinetics parameters at BOL computed by ERANOS.

	CONF-2	CONF-2 (2%)	CONF-2 (4%)
<b><math>\beta_{\text{eff}}</math> (pcm)</b>	393	377	361
<b><math>\Lambda</math> (<math>\mu\text{s}</math>)</b>	0.4406	0.3816	0.3345

The effect of MAs has been also evaluated with respect to the burn-up. A single irradiation period of 2050 efpd has been simulated by recalculating the neutron flux every 102 efpd.

The use of Am loading in the core has advantages in term of reactivity swing as indicated in Figure 7. The CONF-2 (2%) model shows a roughly constant reactivity during the irradiation time mainly due to the Am241 transmutation. For the CONF-2 (4%) configuration, an increase in reactivity *versus* burn-up is underlined.

**Figure 7.** Reactivity swing.

As previously indicated, AmO<sub>2</sub> loaded in the blanket has advantages also in terms of proliferation resistance due to the Pu238 content increase at discharge [35].

Table 4 shows Pu vectors in the core, in the blanket, and in the systems (core + blanket) for the three configurations we have considered. The shares of Pu240, Pu241, and Pu242 in the system remain unchanged. The Pu238 share in the system is increased from 1.97% to 5.98% and therefore, the Pu239 share is slightly reduced from 56% for CONF-2 to 52% for CONF-2 (4%). In the blanket, the Pu238 share is increased at the expense of the Pu239 share: for CONF-2 (2%) the Pu238 content increases up to 6.3% after 2050 efpd. In the case of CONF-2 (4%), the Pu238 content increases up to 11.7%, close to the 12% wt. considered for proliferation resistance [35].

**Table 4.** Plutonium vectors in core, blanket and system at EOI.

	CONF-2	CONF-2 (2%)	CONF-2 (4%)
	Core/Blanket/Total	Core/Blanket/Total	Core/Blanket/Total
	(wt.%)		
<b>Pu238</b>	2.07/ <b>0.01</b> /1.97	3.9/ <b>6.32</b> /4.02	5.73/ <b>11.68</b> /5.98
<b>Pu239</b>	54.12/93.08/55.99	52.42/86.25/53.97	50.8/80.59/52.09
<b>Pu240</b>	30.09/6.59/28.97	29.55/5.70/28.46	29/4.82/27.95
<b>Pu241</b>	5.00/0.31/4.77	4.92/0.21/4.71	4.85/0.15/4.64
<b>Pu242</b>	8.72/0.01/8.30	9.19/1.50/8.84	9.63/2.76/9.33

The transmutation performances of the three configurations considered have been compared. In Table 5 the burning capabilities of the systems, expressed in kg/TWh<sub>th</sub>, are compared assuming 3600 MW<sub>th</sub> and 2050 efpd. All systems have comparable Pu production (*ca.* 7–8 kg/TWh<sub>th</sub> as indicated in Table 5) but the CONF-2 (4%) model burns roughly 50% more MAs with respect to the CONF-2 (2%) configuration.

The isotopic contributions to the burning rates (Pu and main MAs isotopes) are included in Table 5. The separated effect of core and blanket is also taken into account.

The good burning performance and the reasonable void worth of the CONF-2 (4%) configuration suggest some possibilities to improve the system, e.g., by the reduction of the active height. Studies are on-going at KIT to investigate these options.

**Table 5.** Burning capability of the systems (rates at EOI).

	CONF-2		CONF-2 (2%)		CONF-2 (4%)	
	Core	Blanket	Core	Blanket	Core	Blanket
	kg/TWh <sub>th</sub>					
<b>Pu238</b>	-1.00	0.00	0.28	0.35	1.62	0.64
<b>Pu239</b>	4.96	5.33	4.08	4.82	3.28	4.39
<b>Pu240</b>	0.52	0.38	0.26	0.32	0.03	0.26
<b>Pu241</b>	-2.15	0.02	-2.23	0.01	-2.30	0.01
<b>Pu242</b>	-1.05	0.00	-0.71	0.08	-0.36	0.15
<b>Am241</b>	0.44	0.00	-2.80	-0.68	-6.13	-1.21
<b>Am242m</b>	0.04	0.00	0.18	0.05	0.34	0.09
<b>Am243</b>	0.90	0.00	-0.09	-0.18	-1.02	-0.33
<b>Cm242</b>	0.05	0.00	0.21	0.05	0.38	0.09
<b>Cm243</b>	0.00	0.00	0.03	0.00	0.04	0.01
<b>Cm244</b>	0.28	0.00	0.89	0.15	1.47	0.27
<b>Cm245</b>	0.03	0.00	0.12	0.01	0.19	0.02
<b>Pu</b>	1.28	5.72	1.68	5.59	2.27	5.44
<b>MA</b> s	1.75	0.00	-1.47	-0.60	-4.73	-1.07
	<b>Core+Blanket</b>		<b>Core+Blanket</b>		<b>Core+Blanket</b>	
<b>Pu</b>	7.00		7.27		7.72	
<b>MA</b> s	1.75		-2.06		-5.79	

## 5. Recriticality Prevention and Mitigation by Controlled Material Relocation

An important focus of safety research on the fast reactor is the mitigation or even elimination of specific severe accident routes avoiding core disruption and fuel re-compaction phenomena that can lead to secondary power excursions (recriticality events) and energetic disassembly and move the accident toward the so-called “expansion phase” with consequent loading of the structures (*i.e.*, reactor vessel).

The phenomena associated with the transition phase are quite complex and interconnected. The competition between fuel losses from the core and in-core fuel compaction determines the energetic potentials of the accident. Here, the phenomenology associated with the transition phase will not be treated in detail because they are beyond the aim of this paper. We only focus on the measures that may enhance the fuel removal from the core.

In the past, it was proposed to introduce by design, dedicated measures that enhance and guarantee a sufficient and timely fuel discharge by a controlled material relocation (CMR) to enable control of recriticalities and energetic potentials of the accident [12,21,22].

A CMR measure investigated here follows ideas [12,13,39] that have been promoted during the CAPRA/CADRA project [38].

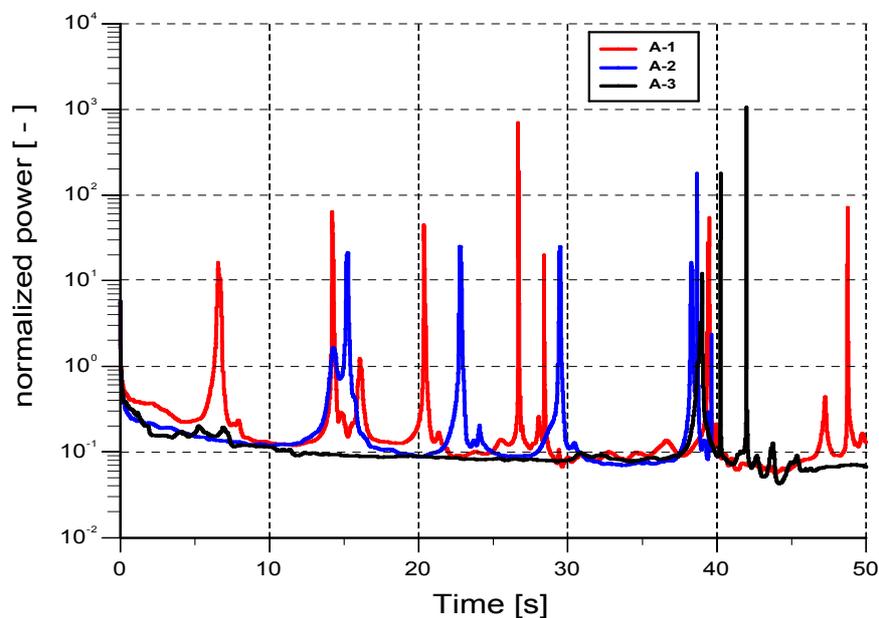
As indicated above, the use of diluents or empty pins (filled only with gas) in a SA has been investigated for reducing the coolant void worth by softening the spectrum. These special pins are characterized by a lower melting temperature than MOX fuel (2600 K for B<sub>4</sub>C with respect to *ca.* 3000 K). During the accident it is therefore expected that they may melt at first and, together with the control rod guide tubes (CRGT), they provide a preferential path for the corium. The CRGTs, indeed, are destroyed within the first few seconds allowing radial motion in the fuel pool and providing natural material relocation paths.

The efficiency of CMR measures based on empty pins has been analyzed with the SIMMER-III code by simulating a ULOF transient for a slightly different SFR configuration [13,23,42]. In this case, 19 empty pins (out of 271 pins) were loaded and grouped ring-wise in each fuel SA.

Three cases have been compared: (1) reference case without CMR measures (Case A-1); (2) adoption of empty pins in each subassembly (Case A-2); and, (3) same conditions as Case A-2, but fuel can be released also through CRGTs (Case A-3).

The results obtained for this SFR configuration show that a final recriticality still takes place in the case of CMR measures but with lower energetics than for the reference case (Figure 8). The CMR measures enable an increase in fuel release. However, this early fuel release can result in some damage to the lower subassembly structures, as indicated in [13,23].

**Figure 8.** Transition phase power traces: reference case (A-1), with CMR (A-2) and with CMR plus CRGTs (A-3).



The same trends can be expected for the ESFR core. Investigations are on-going at KIT as part of the CP-ESFR project both for the REF configuration and for the optimized CONF-2 configuration.

The investigations here discussed show that the recriticality potential and its energetics can be diminished without a too large change in the SA structure while taking into account the CRGTs. The results are encouraging in providing a better understanding of the effect of these CMR measures.

## 6. Conclusions

For long term nuclear energy sustainability, the transition to a fast reactor based fleet and the use of closed fuel cycles is envisaged in many countries. New fast reactor concepts are being developed to assure high safety goals and to mitigate (or even eliminate) severe accident routes that lead to core destruction and recriticalities. Activities in this direction are on-going also within the CP-ESFR project.

In this paper, the main attention has been devoted to the reduction of the positive sodium void worth and therefore to the optimization of the initiation phase of the accident. The measures selected are the

ones that mostly reduce the void effects at BOL without changing the core design. In particular, the adoption of a larger Na plenum shifted close to the core, as well as an absorber layer above the Na plenum, help in reducing the positive void worth of the system by about \$2–3. The introduction of a lower fertile blanket further reduces the void worth by increasing the leakage term and improves the core characteristics, as breeding performance and reactivity swing.

The low value of the extended SVRE at the BOL offers an opportunity for introducing and burning MAs in the systems. A homogeneous addition of Am (between 2% to 4% wt.) to the core and blanket has been studied. The results on core behavior under accident conditions and transmutation performance are encouraging. The preliminary SIMMER analysis has shown that also for the optimized configuration the accident (ULOF) enters the transition phase. Therefore, measures oriented to improve fuel relocation for reducing the possibilities of recriticalities can be envisaged.

In this paper, the adoption of 19 empty pins in each fuel sub-assembly is presented as a possible controlled material relocation measure. Preliminary SIMMER results, obtained for a slightly different SFR configuration, have shown the potential of these attempts.

Further studies are planned at KIT aiming at improving safety (by avoiding core degradation) and transmutation potential of fast reactor designs proposed in Europe.

### Acknowledgments

This work was carried out as part of the CP-ESFR project, which was co-funded by the European Commission under the Euratom Research and Training Programme on Nuclear Energy within the Seventh Framework Programme - EURATOM, contract number 232,658.

The authors acknowledge support by Deutsche Forschungsgemeinschaft and Open Access Publishing Fund of Karlsruhe Institute of Technology.

The authors would like to thank also Edgar Kiefhaber, retired colleague at KIT/IKET, for the careful revision of the paper.

### Conflict of Interest

The authors declare no conflict of interest.

### References

1. SNE-TP. *Strategic Research Agenda*; EU: Brussels, Belgium, 2009. Available online: <http://www.snetp.eu> (accessed on 13 June 2012).
2. GIF IV. *A Technology Roadmap for Generation IV Nuclear Energy Systems*; GIF-002-00; U.S. DOE Nuclear Energy Research Advisory Committee, Washington, DC, USA, 2002. Available online: <http://www.gen-4.org/> (accessed on 3 September 2010).
3. IAEA. *Technical Report—Nuclear Energy Development in the 21st Century: Global Scenarios and Regional Trends*; No. NP-T-1.8; IAEA: Vienna, Austria, 2011.
4. Vezzoni, B.; Salvatores, M.; Gabrielli, F.; Schwenk-Ferrero, A.; Romanello, V.; Maschek, W.; Forasassi, G. Analysis of a hypothetical Italian fuel cycle: Transition to fast reactors. *Trans. Fusion Sci. Technol.* **2012**, *61*, 167–173.

5. NEA/OECD. *Transition towards a Sustainable Nuclear Fuel Cycle: A World Scenario Analysis*; NEA/OECD: Paris, France, 2012, in press.
6. Romanello, V.; Salvatores, M.; Schwenk-Ferrero, A.; Gabrielli, F.; Maschek, W.; Vezzoni, B. Comparative study of fast critical burner reactors and subcritical accelerator driven systems and the impact on transuranics inventory in a regional fuel cycle. *Nucl. Eng. Des.* **2011**, *241*, 433–443.
7. Salvatores, M. Medium and Long-Term Scenarios for Fission Nuclear Energy and Role of Innovative Concept. In *Proceedings of the Int. Workshop Nuclear Reaction Data and Nuclear Reactor*, ICTP, Trieste, Italy, 23 February–27 March 1998.
8. NEA/OECD. Actinide and Fission Product Partitioning and Transmutation 11th Information Exchange Meeting, San Francisco, USA, 1–4 November 2010. Available online: <http://www.oecd-nea.org/pt/iempt11/> (accessed on 13 June 2012).
9. SNE-TP. *The European Sustainable Nuclear Industrial Initiative, ESNII*; Technical Report; EU-EC: Brussels, Belgium, 2010.
10. Rouault, J.; Serpantié, J.P.; Verwaerde, D. French SFR Program and the ASTRID Prototype. In *Proceedings of the Int. Conf. on Fast Reactors (FR09)*, Kyoto, Japan, 7–11 December 2009.
11. Fiorini, G.L. The Collaborative Project on European Sodium Fast Reactor (CP ESFR). In *Proceedings of the Int. Conf. on Euratom Research and Training in Reactor Systems (FISA2009)*, Prague, Czech Republic, 22–24 June 2009.
12. Maschek, W.; Struwe, D. Accident analyses and passive measures reducing the consequences of a core-melt in CAPRA/CADRA reactor cores. *Nucl. Eng. Des.* **2000**, *202*, 311–324.
13. Maschek, W.; Flad, M.; Matzerath Boccaccini, C.; Wang, S.; Gabrielli, F.; Kriventsev, V.; Chen, X.-N.; Zhang, D.; Morita, K. Prevention and Mitigation of Severe Accident Developments and Recriticalities in Advanced Fast Reactor Systems. In *Proceedings of the Int. Conf. on Innovative Nuclear Energy Systems (INES-3)*, Tokyo, Japan, 1–4 November 2010.
14. Bohl, W.R. Some Recriticality Studies with SIMMER-II. In *Proceedings of the Int. Mtg. on Fast Reactor Technology*, Seattle, WA, USA, 19–23 August 1979.
15. Maschek, W.; Asprey, M.W. SIMMER-II recriticality analyses for a homogeneous core of 300 MWe class. *Nucl. Technol.* **1983**, *63*, 330–336.
16. Theofanous, T.G.; Bell, C.R. *An Assessment of CRBR Core Disruptive Accident Energetics*; NUREG/CR-3224T; LANL: New Mexico, USA, 1984.
17. Barthold, W.P.; Beitel, J.C.; Lam, P.S.K.; Orechwa, Y.; Su, S.F.; Turski, R.B. Low Sodium Void Cores. In *Proceedings of the Int. Conf. ENS/ANS Topical Meeting on Nuclear Power Reactor Safety*, Brussels, Belgium, 16–19 October 1978.
18. Rachi, M.; Yamamoto, T.; Jena, A.K.; Takeda, T. Parametric study on fast reactors with low sodium void reactivity by the use of zirconium hydride layer in internal blanket. *Nucl. Sci. Technol.* **1997**, *34*, 193–201.
19. Hill, R.N.; Khalil, H. An Evaluation of LMR Design Options for Reduction of Sodium Void Worth. In *Proceedings of the Int. Conf. on the Physics of Reactors: Operation, Design & Computation*, Marseille, France, 23–26 April, 1990.
20. Ieda, Y.; Niwa, H.; Uto, N.; Kondo, S. Assessment of Proposed Passive Prevention and Mitigation Measures for Future Breeders Reactors. In *Proceedings of the Int. Conf. on Advanced Reactors Safety (ARS-94)*, Pittsburgh, PA, USA, 17–20 April 1994.

21. Tobita, Y.; Morita, K.; Kawada, K.; Niwa, H.; Ninokata, N. Evaluation of CDA Energetics in the Prototype LMFBR with Latest Knowledge and Tools. In *Proceedings of the Int. Conf. on Nuclear Engineering (ICONE-7)*, Tokyo, Japan, 19–23 April 1999.
22. Sato, I.; Tobita, Y.; Konishi, K. Elimination of Severe Recriticality Events in the Core Disruptive Accident of JSFR Aiming at In-Vessel Retention of the Core Materials. In *Proceedings of the Int. Conf. on Fast Reactors (FR09)*, Kyoto Japan, 7–11 December 2009.
23. Vezzoni, B.; Chen, X.-N.; Flad, M.; Gabrielli, F.; Marchetti, M.; Maschek, W.; Matzerath Boccaccini, C.; Rineiski, A.; Zhang, D. Optimization of safety parameters and accident mitigation measures for innovative fast reactor concepts. *Trans. Fusion Sci. Technol.* **2012**, *61*, 155–160.
24. Rineiski, A.; Vezzoni, B.; Zhang, D.; Chen, X.-N.; Gabrielli, F.; Marchetti, M. Sodium Void Effect Reduction and Minor Actinide Incineration in ESFR. In *Proceedings of the Int. Conf. ANS Annual Meeting 2011*, Hollywood, CA, USA, 26–30 June 2011.
25. Rineiski, A.; Vezzoni, B.; Gabrielli, F.; Maschek, W.; Marchetti, M.; Omenetto, A.; Chen, X.-N.; Zhang, D.; Tsige-Tamirat, H.; Sun, K.; *et al.* *Synthesis of Options to Optimize Feedback Coefficients*; CP-ESFR Project deliverable, SP2.1.3.D1; CP-ESFR Project: 2012.
26. Rimpault, G. The ERANOS Code and Data System for Fast Reactor Neutronic Analyses. In *Proceedings of the Int. Conf. on the Physics of Reactors (PHYSOR 2002)*, Seoul, Korea, 7–10 October 2002.
27. NEA/OECD. *The JEFF-3.1 Nuclear Data Library*; JEFF Report 21, NEA No. 6190 OECD-NEA; NEA/OECD: Paris, France, 2006.
28. Carrico, C.B.; Lewis, E.E.; Palmiotti, G. Three-Dimensional variational nodal transport methods for cartesian, triangular and hexagonal criticality calculations. *Nucl. Sci. Eng.* **1992**, *111*, 168–179.
29. Dilber, I.; Lewis, E.E. Variational nodal methods for neutron transport. *Nucl. Sci. Eng.* **1985**, *91*, 132.
30. Kondo, S.; Tobita, Y.; Morita, K.; Brear, D.J.; Hamiyama, K.; Yamano, H.; Fujita, S.; Maschek, W.; Fischer, E.A.; Kiefhaber, E.; *et al.* Current Status and Validation of the SIMMER-III LMFR Safety Analysis Code. In *Proceedings of the Int. Conf. ICONE-7*, Tokyo, Japan, 19–23 April 1999.
31. Yamano, H.; Fujita, S.; Tobita, Y.; Kamiyama, K. *SIMMER-IV: A Three-Dimensional Computer Program for LMFR Core Disruptive Accident Analysis*; JNC TN9400 2003-070; Japan Nuclear Development Institute: Tokyo, Japan, 2003.
32. Tentner, A.M.; Parma, E.; Wei, T.; Wigeland, R. *Severe Accident Approach—Final Report Evaluation of Design Measures for Severe Accident Prevention and Consequence Mitigation*; ANL-GENIV-128; ANL: Chicago, USA, March 2010.
33. Sun, K.; Krepel, J.; Mikityuk, K.; Chawla, R. An Optimization Study for the Safety and Performance Parameters of a 3600 MWth Sodium-cooled Fast Reactor. In *Proceedings of the Int. Conf. on Advances in Nuclear Power Plants (ICAPP2011)*, Nice, France, 2–5 May 2011.
34. IAEA. *Evaluation of Benchmark Calculations on a Fast Power Reactor Core, Final Report of an International Benchmark Program*; IAEA-TECDOC-731; IAEA: Vienna, Austria, 1994.
35. Rineiski, A.; Kessler, G. Proliferation-resistant fuel options for thermal and fast reactors avoiding neptunium production. *Nucl. Eng. Des.* **2010**, *240*, 500–510.

36. X-5 Monte-Carlo Team. *MCNP—A General Monte Carlo N-Particle Transport Code, Version 5*; Tech. Report; LANL: New Mexico, USA, 2003.
37. Gabrielli, F.; Rineiski, A.; Maschek, W. Computation of Heterogeneity Model Parameters for Analyses of Reactor Transients. In *Proceedings of the Jahrestagung Kerntechnik*, Hamburg, Germany, 27–29 May 2008.
38. Languille, A.; Garnier, J.C.; Lo Pinto, P.; Verrier, D.; Deplaix, J.; Newton, T.; Sunderland, R.E.; Kiefhaber, E.; Maschek, W.; Struwe, D. CAPRA Core Studies-the Oxide Reference Option. In *Proceedings of the Int. Conf. on Evaluation of Emerging Nuclear Fuel Cycle Systems (GLOBAL '95)*, Versailles, France, 11–14 September 1995.
39. Maschek, W. A preventive/Mitigative Measure Avoiding Recriticalities in Liquid Metal Reactors. In *Proceedings of the Int. Conf, TOPSAFE'95*, Budapest, Hungary, 24–27 September 1995.
40. Stauff, N.E.; Driscoll, M.J.; Forget, B.; Hejzlar, P. Resolution of proliferation issues for a sodium fast reactor blanket. *Nucl. Technol.* **2010**, *170*, 371–382.
41. Barthold, W.P.; Tzanos, C.P.; Fieitel, J.C. Potential of Large Heterogeneous Reactors. In *Proceedings of the Int. Conf. on Optimization Sodium-Cooled Fast Reactors*, London, U.K, 28 November 1977.
42. Takeda, T.; Kuroishi, T. Optimization of internal blanket configuration of large fast reactor. *J. Nucl. Sci. Technol.* **1993**, *30*, 481–484.
43. Sun, K.; Chenu, A.; Mikityuk, K.; Krepel, J.; Chawla, R. Coupled 3D-Neutronics/Thermal-Hydraulics Analysis of an Unprotected Loss-of-Flow Accident for a 3600 MWth SFR Core. In *Proceedings of the Int. Conf. Advances in reactor Physics (PHYSOR2012)*, Knoxville, Tennessee, TN, USA, 15–20 April 2012.
44. Varaine, F.; Buiron, L.; Boucher, L.; Verrier, D. Overview on Homogeneous and Heterogeneous Transmutation in a French New SFR: Reactor and Fuel Cycle Impact. In *Proceedings of the Int. Conf. on Actinide and Fission Product Partitioning and Transmutation (IEMPT-11)*, San Francisco, CA, USA, 1–4 November 2010.
45. Coquelet, C.; Kieffer, C. Comparison of Different Scenarios for the Deployment of Fast Reactors in France—Results Obtained with COSI. In *Proceedings of the Int. Conf. GLOBAL 2011*, Makasuri, Japan, 11–15 December 2011.