Importance of scenario studies on the "by design" proliferation resistance improvement

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Abstract

Recently, there has been a growing belief that the implementation of safeguards at an early stage of the project, of a given "component" of the fuel cycle, will contribute significantly to the improvement of proliferation resistance behaviour. Obviously, a complete fuel cycle should be considered in the assessment of the proliferation resistance, even if during the nuclear reactor (a fuel cycle component) design there is often a lack of design information and specifications on fuel cycle, installations/plants and location, etc.

Moreover, from the reactor designer point of view, proliferation resistance (and physical protection), despite their importance, are not the main technical parameters or constraints considered in the early conceptual design stage.

Because any optimisation freezes the neutron design, in order to be feasible and efficient, proliferation resistance improvement by design at core level, should be based on a clear and self-consistent set of proliferation resistance requirements (like design constraints) given at a very early stage. An approach that could help the core designer to enhance proliferation resistance characteristics will be discussed. It is based on scenario studies and information coming from them, by analysing the performances and the behaviour of physical "observable" parameters. Promising and useful indications and results have been obtained by the application of such an approach to a European level energy demand scenario.

Background

At the international level there is a growing belief that the implementation of safeguards at an early stage of the project of a fuel cycle component, will contribute significantly to the improvement of proliferation resistance (PR) behaviour. In pursuit of this aim, IAEA is co-operating with the international community to develop a new process called "Safeguards by Design" [1], to help ensure that safeguards are fully integrated into the design process of a nuclear facility from initial planning through design, construction, operation, and decommissioning. By taking into account the design features that facilitate the implementation of international safeguards early in the design phase, i.e. Safeguards by Design, the proliferation resistance of the system can be improved.

Several methods for PR evaluation of a nuclear material, facility, system or whole fuel cycle have been proposed, [2-8]. Each of them is characterised by a proper metric and is shown to have proper advantages and disadvantages, but there is, as yet, no method covering concepts-constraints: accuracy and reproducibility of the results. Methodologies based on expert elicitation are accurate but normally not repeatable. In contrast, the methodologies which provide reproducible results (numerical results) are normally not accurate, due to their dependency on questionable "weight factors" simulating the institutional context.

Boundary conditions/Introduction

The following concept, [9], is commonly accepted for the "definition of PR": "Characteristics of existing or proposed Nuclear Energy Systems (NES) in terms of "Intrinsic Features" and "Extrinsic Measures" that may impede the diversion or undeclared production of nuclear material or the misuse of technology". Considering at the same time both "Intrinsic Features" and "Extrinsic Measures", a PR evaluation requires the analysis of the interdependencies between subjective and objective data, which would imply the use of a "multiple" approach. Currently, there is a "limitation" of the existing methodologies in the application of a similar "multiple" approach.

Concerning the neutron designer point of view, it should be highlighted that the core neutron design is the result of an optimisation process based on the knowledge of the materials' behaviour and the application of physical laws within the defined design constraints. Because any optimisation freezes the neutron design, in order to be feasible and efficient, the PR improvement by design at core level should be based on a clear and self-consistent set of PR requirements (like design constraints) given at a very early stage, i.e. before the conceptual design. It is therefore essential for the designer to have at least guidelines to be used in this early phase if proper "requirements" are not available. A further complication comes from the fact that for the reactor designers the PR (& PP=Physical Protection), despite their importance, are not the main technical parameters or constraints considered in the early conceptual design stage. In order to meet the goal of PR improvement at design level, the results of the PR assessment methodology, identification of weaknesses and addressing NPP site specific conditions, etc. should represent a series-of "design constraints".

The scenario studies, including fuel cycle development, could become an interesting tool to improve PR, at early design phase, because the following information and feedbacks for a single reactor or park of reactors can be obtained:

- sustainability of the energy demand scenario, reactor(s) deployment and fuel cycle assumptions;
- front-end and back-end options and interim storage cooling time options;

- reprocessing and fabrication plant options;
- reactor design impact from:
  - Pu and/or MA recycling,
  - fertile assemblies "deployment" strategy ;
- fuel composition U, Pu, MA during the cycle;
- waste management and radiotoxicity evaluations;
- others.

which are "translated" into:

- information for PR behaviour assessment of the system: fuel + reactor (+ cycle);
- detailed fuel isotopic composition: fuel inventory;
- decay heat at different moments:
  - DHRS dimensioning and reliability,
  - shutdown at refuelling/other operating condition,
  - final disposal acceptability;
- neutron source intensity at:
  - final disposal transfer (shielding dimensioning),
  - reprocessing/fabrication phase;
- radiotoxicity (in final disposal) at short, medium and long time.

Naturally, the purpose of this presentation is not to attribute a proliferating or non-proliferating character to a given NES, but to indicate a way useful to the neutron designer about what should be considered, since the earliest stages of the project, in order to increase proliferation resistance. Some elements of the proposed approach, based on physical "observable", will be provided, through an analysis of some simple and transition scenarios using the French scenario Code COSI, [10].

Fuel enrichment

Fuel enrichment, in terms of fissile content -or Pu content- in the fuel, is one of the proliferation risk factors to take into consideration. Figure 1 shows the behaviour of the Pu content in the SFR, deployed in a European Transition Scenario, [11], obtained by COSI code simulations. An increase in the Pu content in the discharged fuel, with respect to the fresh loaded one, is a consequence of the breeding process. In effect the deployed SFR, in the scenario, is designed to be slightly breeder. PR considerations will produce some feedback on the internal breeding gain and on the design of the fertile blankets; axial blankets should be preferred with respect to the radial blankets. The former will be reprocessed simultaneously with the whole fuel assembly, while the latter can be handled separately. So, radial blankets would impose more proliferation concerns.

Generally, the spread of enrichment (and reprocessing) would increase the proliferation risk. The combination "Adiabatic" Fast Reactor (where Pu is working only to "catalyse" the fission of $U_{nat\ or\ depl}$) and associated closed fuel cycle, [12], would ensure a solution to decrease the need for enrichment in the long-term.

Figure 1: ESFR (break-even core) fuel Pu content

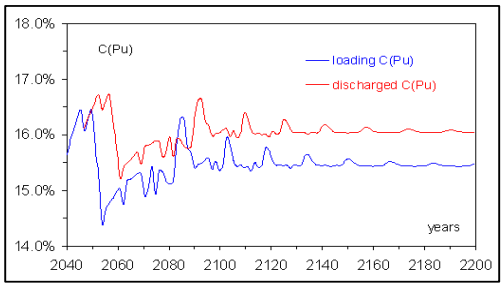
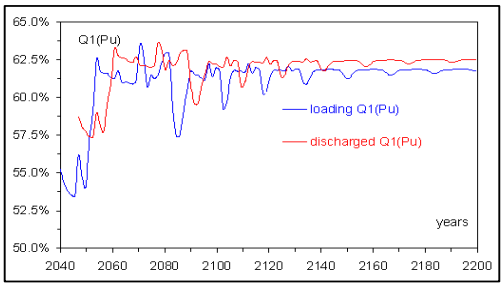


Figure 2: ESFR (break-even core) fuel Pu quality



Quality of the fissile material

One of the main physical observables, for a given NES, concerns the quality of the fissile material which also defines the "attractiveness" of the material itself. Pu isotopic composition is of fundamental importance for PR evaluations because it defines the Pu grade and behaviour with respect to the PR, [13] [14]. According to the IAEA Safeguards Glossary, if a nuclear material includes Pu with ^{238}Pu < 80%, it is confirmed as direct use material that can be used for the manufacturing of a nuclear explosive device without further enrichment. Figure 2 shows the SFR fuel Pu quality, for both loading and discharged fuels, defined as:

$$(^{239}\mathrm{Pu} + {}^{241}\mathrm{Pu}) / \mathrm{Pu},$$

while Table 1 collects the Pu isotopic breakdown, in %, at the scenario equilibrium regime, for the different configurations of the analysed scenario.

Table 1: SFR (break-even core) fuel Pu isotopic composition

	Case 1 Loading	Case 2 Loading	Case 3 Loading	Case 4 Loading	Case 1 Discharged	Case 2 Discharged	Case 3 Discharged	Case 4 Discharged
% Pu239	58.35	58.33	59.96	65.16	58.97	57.29	57.99	63.78
% (Pu239+Pu241)	61.73	60.67	61.75	66.61	62.43	61.49	62.45	67.41
% Pu238	0.54	2.35	0.54	0.36	0.53	2.39	0.53	0.36
% Pu240	34.32	33.25	34.30	30.52	33.87	32.64	33.86	30.07
% (Pu240+Pu242)	37.70	36.98	37.71	33.02	37.04	36.13	37.02	32.23

Case 1: Break-even core, 5-year cooling time; Case 2: Break-even core, 5-year cooling time, MA recycling.
Case 3: Break-even core, 2-year cooling time; Case 4: Breeder core, 5-year cooling time.

Considering that:

- WG-Pu = weapons-grade plutonium, nominally 94% fissile Pu isotopes;
- RG-Pu = reactor-grade plutonium, nominally 70% fissile Pu isotopes;
- DB-Pu = deep burn plutonium, nominally 43% fissile Pu isotopes,

both loading and discharged fuels have reactor grade (RG) Pu for all analysed cases, which according to the GIF – PR&PP WG, [2], has "intermediate" PR characteristics.

Moreover, since ~ 75 tonnes of MOX fuel are required for each SFR, the criticality condition of:

15.5% ≤ C(Pu) ≤ 16.1%; C(Pu) = Pu/(U+Pu) = Pu content

implies a Pu demand of:

~ 11.6 tonnes Pu/reactor or ~ 25.7 kg Pu/FA (= Fuel Assembly) or~ 3 SQ of Pu/FA,

because for Pu containing less than 80% of ^{238}Pu a SQ is ~8 kg of Pu. So, diversion of one FA would be sufficient to obtain about 3 SQ of Pu.

Considering: C(^{A}Pu) = ^{A}Pu / Pu, the results collected in Table 1 show that:

- C(^{238}Pu) < 9% implies a Pu composition weak, [13], with respect to PR;
- C(^{239}Pu) > 50% implies also a RG-Pu composition weak with respect to PR;
- C(^{240}Pu + ^{242}Pu) > 32% means a "good" spontaneous neutron generator Pu composition;
- C(^{240}Pu) > 30% means a RG-Pu composition strong, [14], with respect to PR.

The SFR breeder core obtained by axial blankets introduction shows a "worse" PR by increasing the fissile content and reducing the Pu even isotopes content. In contrast, minor actinide homogeneous recycling does not affect significantly the Pu composition, except C(^{238}Pu), which "increases" the PR through the heating rate.

Concerning fissile materials quality, some interesting information can also be collected from LWR scenarios studies. Considering an energy demand scenario of about 105 TWhe at equilibrium regime, [15], which can be satisfied by three different types of LWR: 8 EPR producing 104.3 TWhe, 40 IRIS producing 105.7 TWhe and 40 CAREM producing 104.1 TWhe, according to the reactors deployment shown in Figure 3, the Pu inventory at the end of the scenario period (80 years) is displayed in Figure 4, while Table 2 collects the spent fuels isotopic composition of all the fuel materials.

Figure 3: Energy demand scenario LWRs

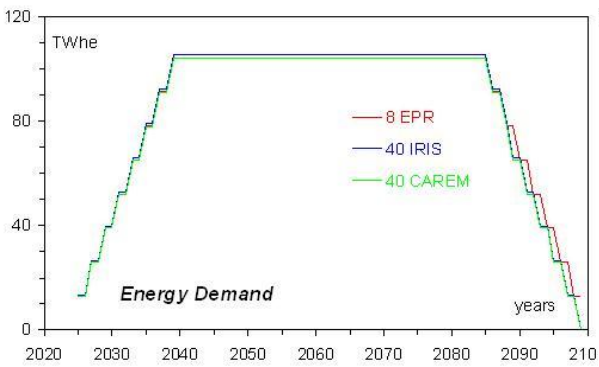
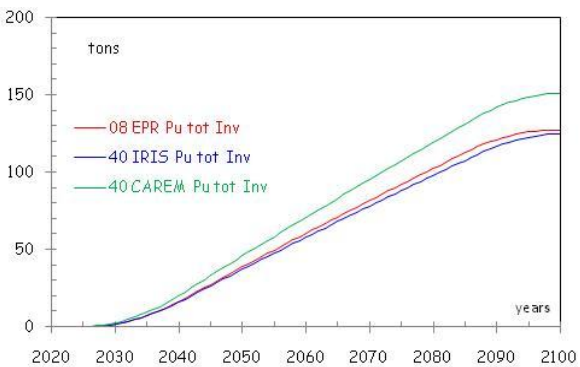


Figure 4: Pu Inventory for the LWR scenario



Large and SMRs behaviour was confirmed for the same scenario boundary conditions. In particular: a weak dependency from the reactor size has been observed. When differences are observed, they mostly depend on the intrinsic characteristics, like: enrichment, burn-up, fuel cycle length, batches number, etc., while the reactor-grade Pu is produced at very low content in ^{238}Pu; a further reduction has been observed, predominant in the CAREM reactor which is characterised by fresh fuel low enrichment and low burnup discharged fuel.

Table 2: LWRs fuel materials isotopic composition

Spent Fuel (=SF) Materials Breakdown (%)

	EPR	IRIS	CAREM		EPR	IRIS	CAREM
U	98.538	98.551	98.913	(^{235}U	0.775	0.823	0.993)
Pu	1.314	1.293	1.013	(^{Fis}Pu	64.170	63.801	69.908)
MA	0.148	0.156	0.074				

SF Pu Isotopic Breakdown (%) :

	EPR	IRIS	CAREM
^{236}Pu	< 1E-4	< 1E-4	< 1E-4
^{238}Pu	3.280	3.586	1.772
^{239}Pu	49.170	48.580	55.478
^{240}Pu	24.086	23.877	22.780
^{241}Pu	15.000	15.221	14.430
^{242}Pu	8.464	8.735	5.540

SF Minor Actinides Breakdown (%) :

	EPR	IRIS	CAREM
MA	**0.148**	**0.156**	**0.074**
Am	24.090	23.202	22.457
^{243}Am	78.348	81.521	74.506
Np	64.149	64.472	70.155
^{237}Np	90.203	89.489	84.315
Cm	11.761	12.326	7.388
^{244}Cm	75.469	76.885	65.022

Multi-Attribute Utility Analysis approach

Similar results are obtained using the Multi-Attribute Utility Analysis (MAUA), [4]. In this approach, which provides numerical results, the PR value for single process in a NES involving i=1, 2, 3, …, I processes is given by:

$$PR_i = \sum_{j=1}^{I} w_j u_j(x_{ij})$$

where:

- w_j = weight for j-attribute;
- u_j = utility function for j-attribute;
- x_{ij} = input value for the utility function for j-attribute in the i-process.

COSI code simulations for the SFR, deployed in the European Transition Scenario, [11], provide the results shown in the next Figures 5 to 8.

Figure 5: SQ Inventory for EPRs

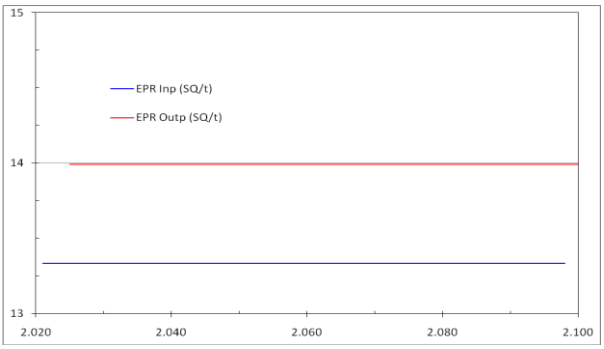
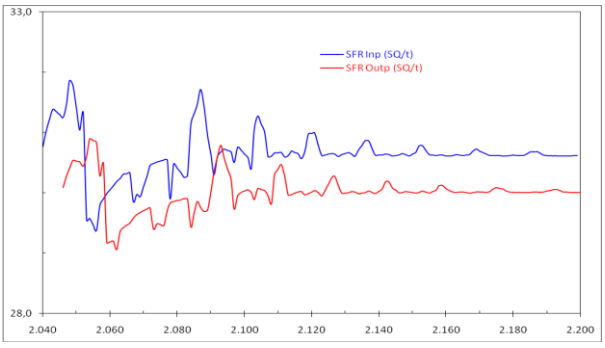


Figure 6: SQ Inventory for SFRs



The fissile material inventory is defined, [4], by:

$$u(x) = \begin{cases} 1, & \text{if } x < 1; \\ \dfrac{(30-x)^{1/3}}{7.18} + 0.574 & \text{if } 1 \le x \le x_{max} = 100; \\ 0, & \text{if } x > x_{max}; \end{cases}$$

where: x is the total (facility) inventory in SQ and x_{max} = the maximum possible inventory, set at 100 SQs, and the results are consistent with the previous ones taking into account the number of the loading batches and the load factor.

Figure 7: SFR heating rate from Pu (W/kg)

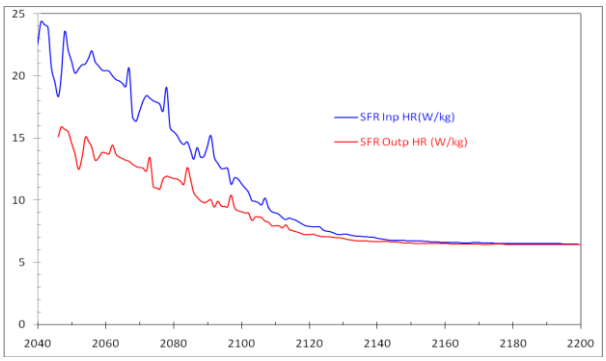
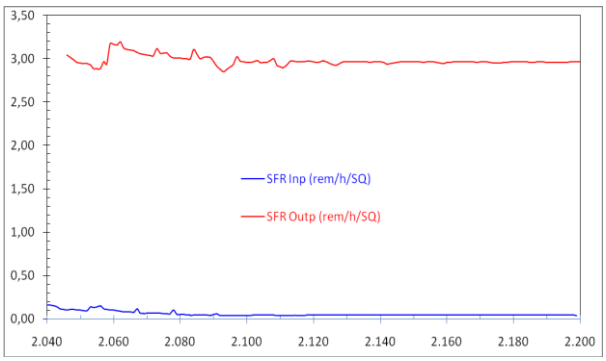


Figure 8: SFR radiation dose rate (rem/h/SQ)



Similarly, Figures 7 and 8 show the heating rate from Pu, per unit mass of FA, and the radiation dose rate, rem/h/SQ, respectively. The physical observables of the last two cases are defined, [4], respectively by:

$$\text{Heating rate (W/kg)} \quad u(x) = 1 - e^{-3\left(\frac{x}{x_{max}}\right)^{-0.8}}$$

$$\text{Dose rate (rem/H/SQ)} \quad u(x) = \begin{cases} 0, & \text{if } x \le 0.2; \\ 0.0520833\,x - 0.010416, & \text{if } 0.2 \le x \le 5; \\ 0.0035714\,x + 0.232143, & \text{if } 5 \le x \le 75; \\ 0.00095238\,x + 0.428571, & \text{if } 75 \le x \le 600; \\ 1, & \text{if } x > 600; \end{cases}$$

The MAUA approach can be applied for further evaluations of the physical observables of a given scenario. Nevertheless, the numerical results, appropriate in a specific context, are only intended to clarify the concept; in effect the aim of the paper is the pursuit of an approach that allows obtaining (at this early design phase) correct PR(&PP) information.

Further considerations

Beyond the numeric answers, the scenario study tool could provide more appropriate information concerning the most promising strategy on the core design in order to improve PR. In the case of FRs deployment (case of a Transition Scenario), the scenario sustainability would be subordinate to the Pu availability; the first answer of the scenario studies would concern this Pu availability.

Figure 9 shows such a transition scenario, [11], while Figure 10 collects the COSI code answers regarding the Pu margin or Pu availability, corresponding to the 4 cases as in Table 1.

Figure 9: High energy demand scenario, [11]

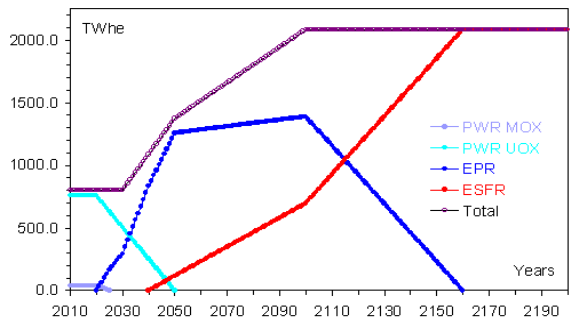
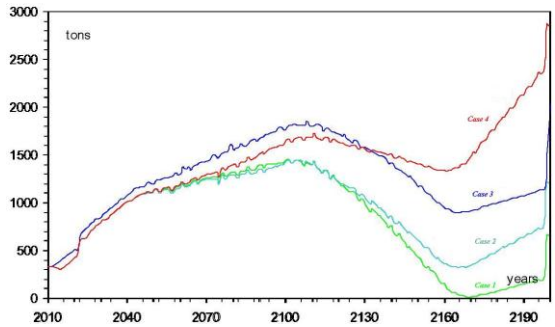


Figure 10: Pu availability for the HED scenario



Case 1: Break-even core, 5-year cooling time; Case 2: Break-even core, 5-year cooling time, MA recycling; Case 3: Break-even core, 2-year cooling time; Case 4: Breeder core, 5-year cooling time.

Concerning the Pu margin for the scenario sustainability, it has been found that:

Case 1: min Pu margin~ 0 tonnes; Case 2: min Pu margin~ 325 tonnes,

Case 3: min Pu margin~ 895 tonnes, Case 4: min Pu margin~ 1339 tonnes,

showing a progressive "worsening" of PR. Some other information has been obtained, as follows:

- Pu content and fissile quality are very close to one another, independently from the energy share;
- zero min Pu margin does not allow scenario sustainability; alternative solutions are requested;
- MA recycling min Pu margin allows scenario sustainability; the breeding gain improvement would allow the scenario sustainability without using fertile assemblies or reducing the cooling time; both of these latter options lead to worsening PR, but the MA recycling option can compromise their safeguard ability, leading to worsening PR;
- since the min Pu margin of ~ 950 tonnes, a reduced cooling time (higher than 2 years) is enough for scenario sustainability, which could improve PR;
- since the large min Pu margin of the breeder core, a specific fertile assemblies strategy can be adopted for scenario sustainability, improving PR.

As for the other components of a NES, which strongly impact the whole PR performance, like enrichment plants, reprocessing plants, fabrication plants, stocks,

temporary storage, transportations, etc. they are not considered in the core design, especially at an early stage.

Conclusions

Even though scenario studies do not provide all the information for detailed PR evaluations, they have become an interesting tool to improve PR features (at an early design phase):

- PR intrinsic features improvement, through breeding gain "modulation", burn-up rate, isotopic detailed fuel composition, fissile materials SQ-inventory, nuclear material attractiveness, heating rate, radiation dose rate, and other information, can be obtained;
- PR extrinsic features improvement, through the whole fuel cycle development, including operative conditions of enrichment, fabrication and reprocessing plants, interim storage, other stocks and waste management, can be obtained;
- PR features improvement, through an "optimised" strategy between different options concerning the design and the operating conditions of all the scenario components, can be obtained.

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