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Editorial

On the Genesis of the Special Issue on Proliferation Resistance

Giacomo G.M. Cojazzi
European Commission, Joint Research Centre

Civil Nuclear Energy Systems (NES) might constitute a potential nuclear proliferation threat, and this should be carefully addressed during their design process.

According to the International Atomic Energy Agency report (IAEA-STR-332), Proliferation Resistance is defined as:

"That characteristic of an NES that impedes the diversion or undeclared production of nuclear material or misuse of technology by the Host State seeking to acquire nuclear weapons or other nuclear explosive devices.

The degree of proliferation resistance results from a combination of, inter alia, technical design features, operational modalities, institutional arrangements and safeguards measures."

Since the nineteen seventies, both the IAEA, International Nuclear Fuel Cycle Evaluation (INFCE) study, and the US driven Non-Proliferation alternative systems Assessment Program (NPAM) initiative, stated that an absolute intrinsic Proliferation Resistance, although desirable, is not achievable in the foreseeable future. After thirty years this statement is still valid. While future nuclear energy systems are not required to be proliferation-proof, they are targeted to exhibit enhanced proliferation resistant features.

Proliferation resistance is currently being investigated within the IAEA *International Project on Innovative Nuclear Reactors and Fuel Cycles* (INPRO) and the Generation IV International Forum (GIF) activities. The driver for this renewed interest is the forecast of a nuclear renaissance in the near to mid future.

Generation IV International Forum considers Proliferation Resistance and Physical Protection as one of the four goal areas in which future nuclear energy systems will have to excel together with Sustainability, Safety & Reliability, and Economics. INPRO shares similar goals in the context of innovative nuclear reactors and fuel cycles.

During the 29TH ESARDA Annual Meeting, Symposium on Safeguards and Nuclear Material Management, held in Aix En Provence, France, on 22-24 May 2007, two sessions on Proliferation Resistance were organized and I had the honor of co-Chairing them with Ms. Caroline Jorant of AREVA.

In view of the quality of the presentations and of the final papers on proliferation resistance appeared in the Symposium proceedings (EUR 22863 EN) and considering the continuous interest in this topic demonstrated within several ESARDA working groups, I launched the idea of a Special Issue of the ESARDA Bulletin dedicated to this argument. The ESARDA Editorial Committee strongly supported this proposal and gave me the go-ahead to the project.

I decided to focus this Special Issue on the material contained in the ESARDA Symposium Proceedings. Most of the original papers which have been presented at the ESARDA Annual Meeting have been retained as candidate papers for this special issue and the corresponding authors were contacted seeking their commitment and support to the project. All contacted authors welcomed the initiative and agreed to support it. The original papers contained in the ESARDA Symposium Proceedings were circulated among all the contributors for comments and cross-reviewing. It has been then the responsibility of the contributors to revise the papers and to update them as necessary to reflect advancements in the field in the last year.

It is with great pleasure that I now present this Special Issue. The papers here included are arranged in two groups, respectively dealing with *Proliferation Resistance Aspects and Safeguards* and with

Methodologies for Proliferation Resistance Evaluation. I believe that they give a rather comprehensive panorama of these topics.

I would like hereby to express firstly my thanks to all contributors to this Special Issue of the ESARDA Bulletin, for accepting this proposal and contributing to it. I also would like to thank all members of the ESARDA Editorial Committee for the support received and for the careful check and timely revision of all the papers. Finally my thanks are for my colleague Cristina Versino, Editor of the ESARDA Bulletin, for all the work done and, in particular, for the efficient interfacing with the ESARDA Editorial Committee.

Proliferation Resistance Aspects and Safeguards

Proliferation Resistance and Physical Protection Robustness Characteristics of Innovative and Advanced Nuclear Energy Systems

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Abstract

Since the early nineteen seventies, a lot of effort has been put into trying to define and evaluate the proliferation resistance of nuclear energy systems and their associated nuclear fuel cycles. Past studies put in evidence how it was not possible to conceive a proliferation-free nuclear fuel cycle (hence the need of a suitable safeguards system), but also stressed that not all of the available options are equivalent.

The topic has become of renewed interest, in the context of the innovative reactor and nuclear energy systems design concepts presently under development. New reactors will have to exhibit and demonstrate enhanced features with respect to the existing ones.

It is common practice to classify Proliferation Resistance and Physical Protection characteristics of a system as either intrinsic, i.e. belonging to the system, or extrinsic, such as those related to the application of international safeguards.

This paper will summarise in a critical way some of the Proliferation Resistance & Physical Protection (PR&PP) intrinsic features that have emerged so far, in a number of studies and reports available in this field, and can contribute to provide a first input to designers to brainstorm on a number of possible requirements. This survey is part of a JRC activity contributing to the Generation IV International Forum (GIF).

Keywords: Proliferation Resistance, Physical Protection, Robustness, Intrinsic characteristics.

1. Introduction

The proliferation resistance of innovative nuclear energy systems and their associated fuel cycles is being reconsidered after various studies developed in the 1970s [1].

The Generation IV International Forum (GIF) aims to develop a set of promising reactor concepts, to be studied and developed in time for deployment in the years 2020-2030. Indeed, Proliferation Resistance & Physical Protection (PR&PP), Safety, Economics and Sustainability are the four goal areas where innovative nuclear energy systems will have to excel, according to the Generation IV International Forum roadmap project report developed in 2002 [2].

Both for proliferation resistance and physical protection, it is common practice to distinguish between intrinsic characteristics of the system (i.e., inherent features of the system design, lay-out and interfaces), and extrinsic measures, (i.e., features related to local standards and requirements and to the application of international safeguards to the system [3]).

The present paper will summarise, in a critical way, some of the PR&PP intrinsic features of innovative nuclear energy systems, focussing on GEN-IV reference designs. These have emerged from a number of studies and reports available in this field, including the GEN-IV International Forum; IAEA; and other scientific publications.

This survey is done in the context of JRC's contribution to the GIF PR&PP Working Group.

2. Proliferation Resistance and Physical Protection of Nuclear Systems

Proliferation resistance and physical protection are requirements that must be ensured for the whole fuel cycle and nuclear system's entire lifetime.

According to [3] and to the definitions adopted by the GIF PR&PP Methodology Rev.5 [4]:

- **Physical protection (robustness)** *is that characteristic of a nuclear energy system that impedes the theft of materials suitable for nuclear* *explosives or radiation dispersal devices (RDDs) and the sabotage of facilities and transportation by sub-national entities or other non-Host State adversaries.*

- **Proliferation resistance** *is that characteristic of a nuclear energy system that impedes the diversion or undeclared production of nuclear material and the misuse of technology by the Host State seeking to acquire nuclear weapons or other nuclear explosive devices.*

The main intrinsic PR&PP features of a nuclear system to be considered have been identified by the GIF, international activities like IAEA/INPRO [5], as well as by other studies [6].

Physical protection requires adequate security of the energy production plants and of related fuel cycle plants, which can be facilitated by the adoption of suitable intrinsic physical protection features, i.e., design choices that rely as much as possible on passive safety systems, able to operate without electric power as might be the situation in case of sabotage. But also the vulnerability of fuel transport must be taken into account, linked to the fuel cycle design choices (central or co-located facilities).

Fissile material could be diverted from the system at any stage, i.e., through removal of fresh or spent fuel from the reactor, during reprocessing, if applied, or even during transport to reprocessing or final conditioning before disposal in the open-cycle case.

Intrinsic proliferation resistance features depend on the strategic choices for the system to develop (e.g., reactor type, fuel cycle, material qualities) and the design adopted to cope with technical requirements and difficulties. These include design features that can increase technological difficulties for diversion of fissile material and fabrication of weapons, such as:

- Type, accessibility and inventory of feed fuel
- Evidence of separated fissile material throughout the fuel cycle, which is linked to the reprocessing process
- Spent fuel characteristics (e.g., burn-up, radiation barriers, fissile material isotopic composition, heat generation rate, neutron emission, critical mass, radiation signature for detectability).

In this respect, Appendix D in the Addendum of the PR&PP methodology study report [7], a paper in the ESARDA 2007 Symposium proceedings [8] and this special issue, discuss the concept of safeguardability of advanced nuclear energy systems as an assessment method.

It appears evident that proliferation resistance and physical protection issues are both linked to the system's intrinsic features resulting from design choices, playing a key role both in making the system a non-attractive route to diversion, and in facilitating the implementation of safeguards. For this reason, it is of paramount importance to take them into account as early as possible in the design.

3. Summary of Innovative Reactor Designs

A large number of innovative reactor systems is being studied worldwide. Different designs still have the chance to be actually developed, as the possible future reactor fleet will be formed by different complementary types of nuclear systems.

In comparison to the existing second and third generation reactors, Generation IV reactors should be able to exploit as much as possible the fissile and fertile properties of uranium, and possibly thorium, at the same time minimizing waste generation and its radiological issues by recycling Minor Actinides (MA), enhancing proliferation resistance and physical protection.

The classification of innovative nuclear energy systems is usually done according to the reactor types:

- Water-cooled
- Gas-cooled
- Liquid Metal-cooled
- Non-conventional

3.1. GIF Roadmap

Several reactor system concepts and the associated nuclear fuel cycle options were considered during the GEN-IV Roadmap project (2001-2002) [2].

Four goal areas of excellence were defined for Generation IV nuclear energy systems:

1. Sustainability, (SU);
2. Economics, (EC);
3. Safety and Reliability, (SR);
4. Proliferation Resistance and Physical Protection (PR&PP)

In order to assess the proposed designs, the four goal areas of excellence were assigned eight equally important goals:

- Resource Utilisation (SU1)

- Waste Minimization and Management (SU2)
- Life Cycle Cost (EC1)
- Risk to Capital (EC2)
- Operational Safety and Reliability (SR1)
- Core Damage (SR2)
- Offsite Emergency Response (SR3)
- Proliferation Resistance and Physical Protection (PR1)

The goals were worked out into 15 weighted criteria and, finally, into 26 metrics. For what concerns in particular the PR&PP goal, the GIF Roadmap project adopted the hierarchy of goals, criteria and metrics reported in Table 1.

Goal area	Goal	Criteria	Metrics
		PR1-1 Susceptibility to diversion or undeclared production.	Separated materials
Proliferation resistance and physical protection	PR1 Proliferation resistance and physical protection		Spent fuel characterization
		PR1-2 Vulnerability of Installations.	Passive safety Features

Table 1: PR&PP Roadmap evaluation criteria and metrics.

As reported in [9] - [12], four GIF Technical Working Groups of experts, one per reactor system type, analysed and screened the proliferation resistance potential of a total of 124 Innovative Nuclear Energy Systems (38 Water-Cooled, 21 Gas-Cooled, 33 Liquid Metal-Cooled and 32 Non-Conventional) of various sizes.

The evaluation was done on the basis of the weighted criteria and metrics, by comparing the systems' features with those of a typical third generation system reference (Advanced Light Water Reactor, ALWR) and led to the further identification of the most suitable nineteen nuclear energy system concepts [13], ranked according to a systematic procedure.

A huge amount of material has been generated during this evaluation process. Most of this material was disseminated only through the worldwide web, on which it was openly available, and not reported at conferences, nor in the scientific literature.

3.2. IAEA Study on Small and Medium Sized Reactors

Besides the innovative designs considered by the GIF in its Roadmap, about 50 Small and Medium Reactor (SMR) concepts remain under consideration in more than 15 IAEA Member States. Small reactors are defined as those with electric power less than 300 MWe, whereas medium ones are in the range 300-700 MWe.

A recent Technical Document issued by the IAEA [14], dedicated to the design status of SMRs in 2005, describes designs of 13 Water-Cooled, 6 Gas-Cooled, and 6 Liquid Metal-Cooled reactors as well as of 1 Non-Conventional reactor. Half of the design concepts presented in the document also appear in the lists of the GIF.

Paragraphs 4 and onwards contain descriptions of the main technical and PR&PP characteristics of the four groups of innovative nuclear systems designs taken from the GIF Roadmap and IAEA's lists, focussing on the six designs considered as GEN-IV references.

4. Water-Cooled Reactor (WCR) Systems

38 Water-cooled reactor systems and fuel cycle concepts were considered by the GIF in its Roadmap TWG1 [12]. These included most of the concepts belonging to the largest family of SMRs reported by the IAEA (thirteen concepts, 50% of the total), the majority of which is formed by Light-Water-type reactors (six Pressurized Water Reactors (PWR), three Boiling Water Reactors (BWR), one indirect BWR, two innovative pool type reactors) and one Advanced Heavy Water Reactor (AHWR).

According to [14], some water-cooled SMRs, like SMART[1] , IRIS, MARS, and IMR, present longer operational cycles and a reduced number of inspections, which could be seen as simplifying the implementation of safeguards. IRIS is characterized by regional or centralized reprocessing, a burn-up which at a later stage could attain 120 GWd/tHM and degradation of secondary plutonium isotopic composition. Besides the interesting VBER-300 floating nuclear power plant, there are also noticeable designs foreseeing a closed nuclear fuel cycle like the Reduced Moderation Water Reactor (RMWR). The basic AHWR design is characterized by a once-through fuel cycle with Pu /Th /233U, but can also be extended to a closed fuel cycle.

1 For the meaning of the systems' acronyms, see the list of acronyms at the end of the paper.

The screening for proliferation resistance potential yielded the following reactors (GEN-IV references highlighted):

- W1-LWR Integral Primary System Reactor Concept Set
- W2-Large Simplified Boiling Water Reactor (ESBWR)
- W3-NG (Next Generation) CANDU - With Low Enriched Uranium (LEU) Once-Through Cycle
- W4-SCWR Supercritical Water Reactor (SCWR) – Thermal Spectrum (GEN-IV reference)
- W5-SCWR Supercritical Water Reactors - Fast Spectrum (SCWR-Fast) (GEN-IV reference)
- W6 High Conversion Advanced Boiling Water Reactor, ABWR-II

4.1. GEN-IV Supercritical-Water-Cooled Reactor System (SCWR)

Amongst Water-Cooled reactors, GIF's selection prioritized supercritical fission reactors [Figure 1], to be developed by 2025, whose characteristics are the following:

- Operation at high temperatures and pressures (above the light water critical point of 374 °C, 221 bar)
- High thermal efficiencies (45-50 %)
- Target burn-up 45-50 GWd/tHM
- Very compact nature of the physical plant, with lower coolant mass inventory
- Simpler design than LWRs (no steam separators and generators)
- Higher heat transfer rate per unit mass flow (large specific heat above the critical point)
- Single-phase fluid with no re-circulation
- Both direct and combined direct/indirect cycles
- Both light and heavy water moderated concepts
- High coolant outlet temperatures allowing potential for hydrogen production.

Currently, two designs are considered: Pressure-Tube (PT) and Pressure-Vessel (PV).

The PT option could foresee both on-line and off-line re-fuelling, also in batches. For the PV reactor only off-line re-fuelling is envisaged. The neutron energy spectrum is thermal (with uranium oxide fuel enrichment slightly above 5%), or fast (MOX fuel). All the options are still at the conceptual design stage.

The difference between a thermal and a fast supercritical water-cooled reactor is in the lattice pitch

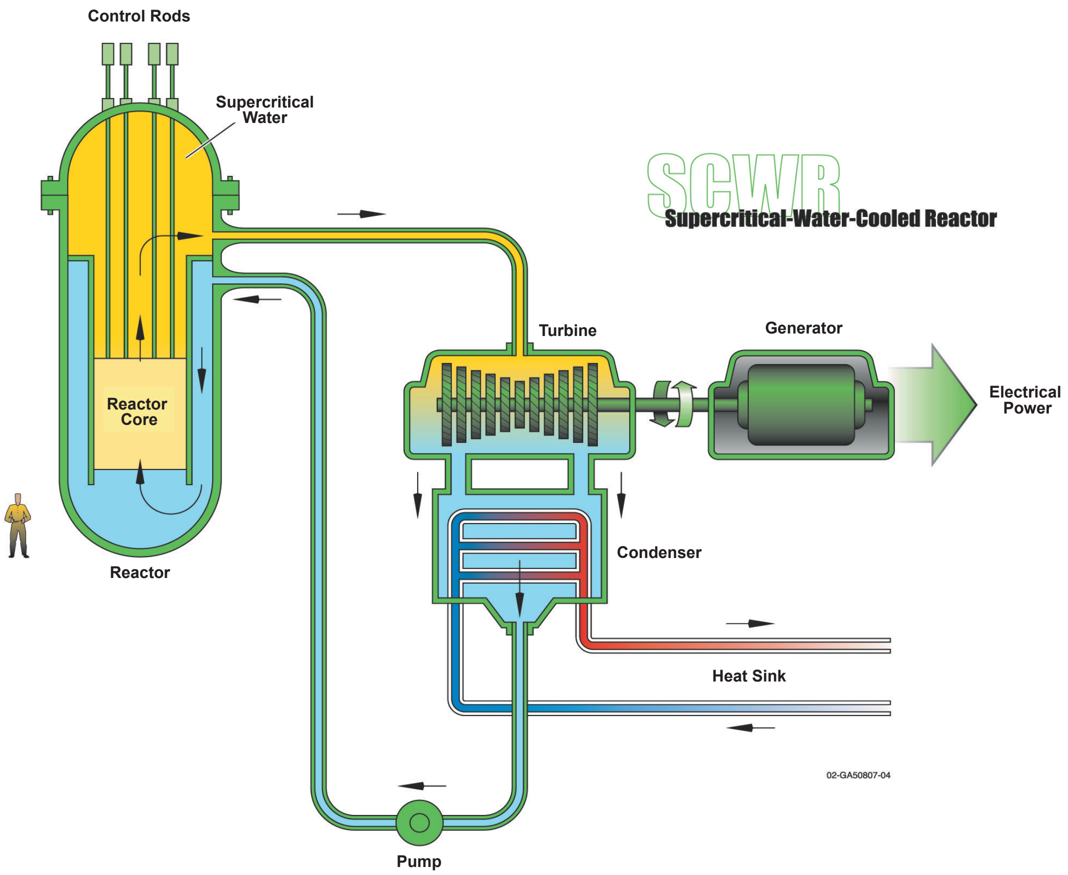

Figure 1: Supercritical-Water-Cooled Reactor - GEN-IV Roadmap's pictorial view [2].

and in the use of moderating material. The fast spectrum reactors use a tight lattice and no additional moderator material, whereas the thermal spectrum reactors need both a loose lattice and additional moderator material in the core.

The thermal option foresees an open once-through cycle, while advanced aqueous reprocessing at a central fuel cycle facility (see 9.2) is foreseen in the case of the fast option closed cycle with MOX fuel.

4.2. WCR Proliferation Resistance Intrinsic Features

In general, water-cooled innovative reactors present features similar to existing PWRs and BWRs:

- Low enrichment uranium dioxide fuel in fresh and spent fuel
- Once-through fuel cycle (most designs)
- Unattractive isotopic composition of plutonium in discharged fuel
- Radiation barriers provided by the spent fuel.

The option of a closed U-Pu cycle is applicable to fast reactors, with issues related to advanced aqueous reprocessing. According to the information reported in a joint workshop between GIF PR&PP WG members and Generation System Steering Committees SSC members [15], a proliferation resistance assessment of SCWR concepts has not yet been made.

4.3. WCR Physical Protection Intrinsic Features

Passive safety features are in general similar to the reference ALWR system. Passive safety of SCWR is facilitated by the lower heat content of the reactor coolant system, which results in lower containment loadings during a design-basis Loss-Of-Coolant Accident (LOCA).

The SCWR Pressure-Tube option could enhance the role of the moderator as a passive heat sink.

5. Gas-Cooled Reactor (GCR) Systems

Twenty-one high-temperature GCR system concepts were contributed to GEN-IV Roadmap's TWG-2 [10], grouped into:

- Modular Pebble Bed Reactor Systems (PBR)
- Prismatic Fuel Modular Reactor Systems (PMR)
- Very-High-Temperature Reactor Systems (VHTR)
- Gas-Cooled Fast Reactor Systems (GFR)

The main property of all GCRs is the use of inert gas, which avoids possible cliff-edge effects, due to phase transitions possible with other fluids, and decouples thermal-hydraulics and neutronics.

GCRs based on a direct Brayton gas turbine cycle are expected to attain enhanced sustainability and economics.

The screening for proliferation resistance potential yielded the following reactors (GEN-IV reference highlighted):

- G1 PBR Modular Pebble Bed Reactor - Once Through
- G2 PMR Prismatic Fuel Modular Reactor - LEU Open Cycle
- G3 VHTR Very High Temperature Reactor - LEU Open Cycle *(GEN-IV reference)*
- G4-Generic HTGR - Closed Synergistic Flexible Fuel Cycle
- G5- Gas-Cooled Fast Reactor GFR - Closed Cycle *(GEN-IV reference)*

We focus hereinafter on the main features and proliferation resistance aspects of the two GEN-IV reference designs.

5.1. GEN-IV Very-High-Temperature Reactor System (VHTR)

Based on PBR or GT-MHR concepts, but with a higher inert helium coolant outlet temperature (above 950°C), it is an advanced, high-efficiency reactor system, which can be used in energy-intensive, non-electric processes (e.g., hydrogen production) as well as supplying process heat to a broad spectrum of high temperature applications. Its main features are:

- High thermal efficiency (45-50%)
- High burn-up, hence reduced waste production and disposal burden
- Larger scope of potential waste applications, for example, coal gasification and metallurgic processes
- Improved intrinsic proliferation resistance due to refractory coated fuel, low fissile inventories and open fuel cycle
- Increased passive safety due to refractory fuel precluding damage under all operating and accident conditions
- Thermal neutron-spectrum and once-through uranium cycle

- Flexibility to adopt U/Pu fuel cycles and improve waste minimization.

VHTR is the nearest-term hydrogen production system [Figure 2], foreseen by 2020, but still needs R&D on high-temperature resistant alloys, fibre-reinforced ceramics or composite materials, and zirconium-carbide fuel coatings, including confirmation of fuel behavior under accident conditions.

5.1.1. Fuel Design

Two options exist for the fuel design:

- Prismatic block fuel, where TRISO-coated particles are mixed with a matrix and formed into cylindrical fuel compacts, with a foreseen burn-up of up to 200 GWd/tHM.
- Pebble fuel, where TRISO-coated micro-spheres are contained in a 6 cm ball configuration called pebble, with a foreseen burn-up of up to 100 GWd/tHM.

TRISO-coated particles (650 microns to about 850 microns) consist of a spherical kernel of fissile (standard UO_2, or UCO), encapsulated in multiple coating layers (SiC or ZrC). The multiple coating layers form a highly corrosion-resistant barrier, essentially impermeable to the release of gaseous and solid fission products up to high temperatures of 1,600 oC. This enables the fuel to withstand high temperatures arising in case of accident or sabotage.

Some pebble bed gas-cooled reactor concepts rely on continuous re-fuelling within an annular core or in channels (PBR, APBR types), whereas others employ prismatic pin-in-block fuel (GT-MHR types) in graphite channels. With this fuel arrangement, GCRs can accommodate a wide variety of mixtures of fissile and fertile materials without any significant modification of the core design. The solid moderator in GCRs also avoids the positive void coefficient of reactivity, which limits the plutonium content of LWR MOX fuels.

The operating characteristics of the GCRs accommodate the use of a wide range of fuel cycles without changing the basic reactor system design. The applicable fuel cycles range from LEU to thorium-uranium to plutonium alone. An option of closed fuel cycle could be foreseen for the GT-MHR, with MOX or hybrid U-Th fuel. Reprocessing, either aqueous or pyro-chemical, is still an open issue (see par. 9.2).

5.1.2. VHTR Proliferation Resistance Intrinsic Features

Intrinsic proliferation resistance features common to this category include:

- Once-through fuel cycle
- High fuel burn-up, with low residual plutonium inventory and high content of Pu-240
- Difficulty to process fuel (e.g., TRISO)
- High spent fuel radiation barriers
- Low ratio of fissile-to-fuel volume, both in compact and pebble type.

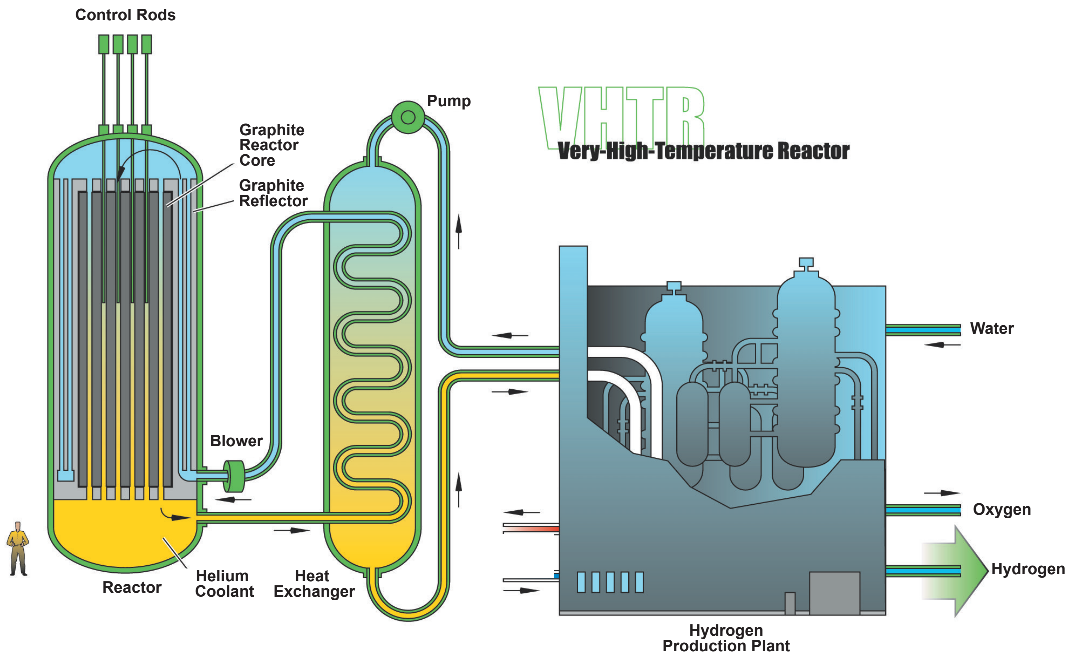

Figure 2: Very-High-Temperature Reactor - GEN-IV Roadmap's pictorial view [2].

A relevant proliferation resistance characteristic of pebble bed reactors is the huge number of pebbles that would have to be diverted to accumulate enough nuclear material suitable for weapons production. For example, given an initial uranium enrichment of 8-10%, to obtain 5 kg of ^{235}U, one would have to divert either 5-10 blocks of prismatic fuel, or more than 5,000 pebbles [15]; obtaining plutonium from spent fuel would be even more cumbersome (hundred of thousands of pebbles [10]). This poses also a difficulty in nuclear material accountancy and control, but remains a considerable advantage over LWRs, in which this same amount of plutonium could be retrieved from only two spent fuel assemblies.

The high proliferation resistance is primarily due to the refractory coated fuel form, hard to access, and the low fissile fuel volume fraction. The refractory coatings provide a containment from which it is difficult to retrieve fissile materials. The technology for the reprocessing of the TRISO fuel has not yet been fully established, thus providing a considerable increase in proliferation resistance. In fact, PUREX cannot be directly applied to TRISO particles, because silicon carbide coating layers are not dissolved by acid mixtures and, therefore, require additional mechanical treatment.

The production of plutonium per MWd produced is lower than in an LWR, resulting in a higher proliferation resistance. Moreover, VHTR systems are able to reach very high burn-ups, which are far beyond the possibilities offered by other thermal reactors (with the exception of molten salt reactors). In particular, the GT-MHR with optimized TRISO kernels prismatic fuel in graphite channels, could achieve a high-burn-up capability allowing for essentially complete ^{239}Pu fission and transmutation of 90% of all TRU waste in a single burn-up. This minimises the proliferation risk in the use of this fuel form, as well as limiting the generation of secondary waste [16].

5.1.3. VHTR Physical Protection Intrinsic Features

In terms of passive safety in case of loss-of-coolant events, the VHTR presents the following characteristics:

- Helium coolant, which is single phase, inert, and has no reactivity effects
- For the GT-MHR design, a graphite core with high heat capacity and structural stability at very high temperatures
- Refractory coated particle fuel, retaining fission products up to temperatures much higher than normal operation
- Negative temperature coefficient of reactivity, which inherently shuts down the core above normal operating temperatures
- Low power density core (6 kW/l) in a steel reactor vessel surrounded by a natural circulation reactor cavity cooling system, as in GT-MHR
- Removal of decay heat by heat conduction, thermal radiation and natural convection, keeping fuel particle temperatures below damage limits.

5.2. GEN-IV Gas-Cooled Fast Reactor System (GFR)

Gas-Cooled Fast reactors are potentially highly sustainable and economically competitive, combining the advantages of high temperature with breeding fuel and burning actinides. They exhibit the same safety features as thermal VHTRs, with the advantages of a closed and integrated fuel cycle reducing proliferation issues and the needs for mining and transport of nuclear materials.

The first reference concept in 2002 [Figure 3] was a 600 MWth / 288 MWe helium-cooled reactor system, operating with an outlet temperature of about 850 °C, using a direct Brayton cycle gas turbine, with a thermal efficiency estimated around 48% and potential for hydrogen production [10].

More recently [17], the reference design has become the indirect combined cycle, with three steam generators, a thermal power in the range 2,000 - 3,000 MWth and an efficiency of 45-48%.

The GFR fuel design foresees, at present, a plate fuel element with an actinide compound/solid solution content (UPuC or UPuN), encased by a tightness refractory liner and a SiC/SiC matrix ceramic cladding.

GFR enables plutonium breeding and minor actinides transmutation. The fuel composition includes 15-20% enrichment in plutonium, with 1% minor actinides and balance uranium (depleted or natural), with total quantities respectively 11,000 kg, 700 kg and 55,000 kg, in the case of a thermal power of 2,400 MWth. The burn-up should reach 10% FIMA.

The GFR development is foreseen to be completed by 2025, with a cycle lifetime of 5-10 years and a global breeding ratio greater than one.

5.2.1. GFR Proliferation Resistance intrinsic Features

These are mainly based on the idea of not separating trans-uranium materials and on the following features [15]:

- Fissile materials are diluted in the fuel ceramic matrix.
- No enriched uranium is used (only natural or depleted).
- Plutonium is low grade.
- Fresh fuel elements or sub-assemblies possibly incorporate minor actinides to increase radiation level.
- Fuel element is not separate from sub-assembly on reactor site.
- Spent fuel sub-assemblies are difficult to transport due to the intrinsic radiological barrier.

Using a closed cycle, proliferation resistance aspects are then mainly relevant in fuel handling and on site storage, prior to fuel reprocessing.

For the latter, there is no specific project on the GFR fuel cycle in the GIF.

A proliferation resistance weakness could be represented by fresh fuel, if pure UPu was not loaded with minor isotopes.

5.2.2. GFR Physical Protection Intrinsic Features

The power density is about 90 kW/l, i.e., much higher than in VHTR, but lower than in a LMFBR.

Mainly based on the idea that a robust containment building should protect the core from external hazards, hence a pre-stressed concrete containment building is included.

From the point of view of material choice, it has already been pointed out that helium as inert coolant is favourable, causing no chemical reactions (fire, explosion).

In case of safe shutdown following sabotage, decay heat removal can be achieved by natural circulation of the gas. Moreover, the refractory encasing of the

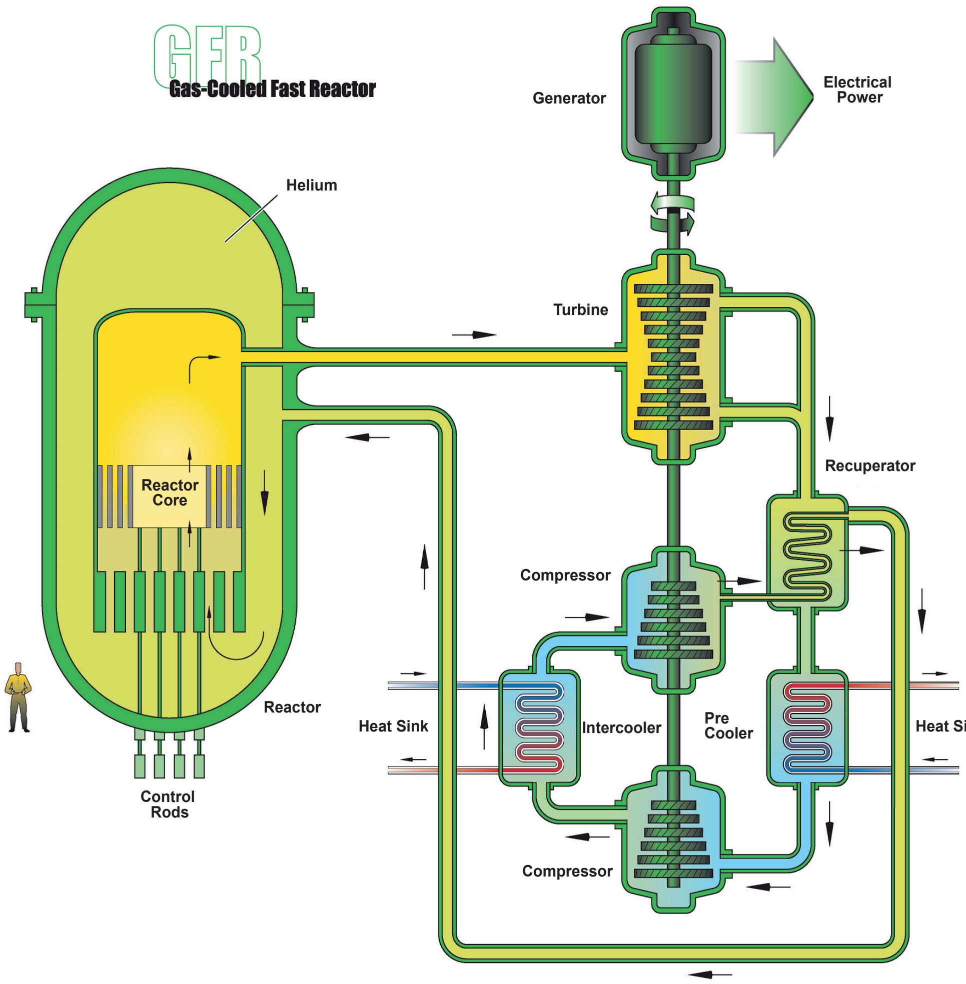

Figure 3: Gas-Cooled Fast Reactor - GEN-IV Roadmap's pictorial view [2].

ceramic fuel matrix inside the plate can sustain very high temperatures:

- Clad ~1,600 °C without FPs release
- Clad ~2,000 °C without loss of geometry.

6. Liquid-Metal-Cooled Reactor Systems

Thirty-three liquid metal concept descriptions, from eight different countries, were considered by the GIF's TWG3 [11]. Only four concepts of the 33 LMCRs were selected as having a proliferation resistance potential, and two (Sodium-cooled and Lead/Lead-Bismuth-cooled) eventually resulted as reference concepts [Figures 4 and 5]:

- L1-SFR Sodium Fast Reactor, MOX Fuel, 1,500 MWe
- L2-SFR Sodium Fast Reactor, Metal Fuel, 760 MWe
- L4-LFR Medium Pb/Pb-Bi-Cooled, US Design Systems, 300-400 MWe
- L5-LFR Large Medium Pb/Pb-Bi-Cooled, Russian Design Systems, 1,200 MWe
- L6-LFR Small Pb/Pb-Bi-Cooled, 50-150 MWe

All but L5 are retained as GEN-IV reference designs.

It is also worthwhile to remember the Russian RBEC-M, an example of an innovative SMR design cooled by lead flowing by natural convection and gas lift, sized between 120-400 MWth.

6.1. Fuels and Fuel Re-processing

All design concepts foresee a fast neutron spectrum for efficient conversion of fertile uranium and a closed fuel cycle, with full actinide recycle fuel cycle at either central or regional fuel cycle facilities.

The options for fuel type are still under consideration with the various designs.

An important common issue, requiring further R&D, is fuel loading with minor actinides and its behaviour at increasing burn-up.

6.1.1. Mixed-Oxide Fuel

MOX (PuO_2, UO_2) has been the reference fast reactor fuel for over 20 years, with the demonstration of high burn-up mixed oxide fuel achieved in the FFTF (USA), PHENIX (France), MONJU (Japan), and PFR (UK). MOX recycling options include the advanced aqueous process, with uranium and plutonium co-extraction, along with most of the minor actinides.

6.1.2. Metal Fuels

Metal fuels were reconsidered in the 1980's, and have so far been tested for shorter periods compared to MOX, reaching lower burn-ups, i.e., up to 10% FIMA compared to more than 20% FIMA for MOX. Examples of metal fuels are ternary metallic alloys U-Pu-Zr or U-TRU-Zr. Characteristics of metal bonded fuel are its higher density, yielding a faster neutron spectrum than oxide fuel and a smaller core volume, and high thermal conductivity, reducing the operating temperature of the fuel.

Re-processing is done by pyro-metallurgical processes, which could also be applied to MOX fuels (see par. 9.2.3).

6.1.3. Nitride Fuels

The state of development of nitride fuels is modest compared to either the mixed oxide or the metal alloy. They are attractive for their high heavy metal density, good thermal conductivity and excellent compatibility with sodium and lead. But the amount of testing is still very small.

Carbide fuels are also under consideration.

6.2. GEN-IV Sodium-Cooled Fast Reactor System (SFR)

For its density, heat transfer characteristics, and compatibility with the stainless steel materials of construction, sodium remains the coolant chosen in most fast reactor design developments. Sodium-cooled fast reactors are primary candidates for nearest term development in countries like France, within a European collaborative framework, USA (as Advanced Burner Reactor foreseen by the Global Nuclear Energy Partnership (GNEP) by 2025 [18]), as well as the Republic of Korea, India, and People's Republic of China.

Sodium has, however, two major disadvantages: its chemical reactivity, which has caused problems in the past in French and Japanese reactors, and its positive void coefficient of reactivity in most plutonium-fuelled applications, which needs to be investigated further to eliminate possible safety issues.

Three design options are under consideration by GIF, with different power levels.

Small (50-150 MWe) Modular type:

- Metal uranium-plutonium-minor actinides-10% zirconium alloy fuel
- 15% plutonium enrichment (Pu/HM)
- Burn-up 87 GWd/tHM

- Pyro-metallurgical processing in co-located facilities.

Intermediate power (300-600 MWe) Pool type:

- Metal uranium-plutonium-minor actinides-10% zirconium alloy fuel
- 24.9% plutonium enrichment (Pu/HM)
- Burn-up 79 GWd/tHM
- Pyro-metallurgical processing in co-located facilities.
- An example is the Korean KALIMER design.

Medium-to-large power (600-1,500 MWe), Loop-type:

- MOX uranium-plutonium fuel
- 13.8% plutonium enrichment (Pu/HM)
- Burn-up 150 GWd/tHM
- Advanced aqueous processing, possibly at a central location supporting several reactors (see par. 9.2).
- An example is the Japanese JSFR design.

The recently concluded EISOFAR Specific Support Action *(Roadmap for a European Innovative Sodium-Cooled Fast Reactor, EURATOM 6th Framework programme)* has contributed to identify various R&D needs for the development of the SFR [19].

6.3. GEN-IV Lead-Cooled Fast Reactor

The LFR systems feature a fast neutron spectrum and a closed fuel cycle for efficient conversion of fertile uranium and management of actinides. A full actinide recycle fuel cycle with central or regional fuel cycle facilities is envisioned. The LFR can also be used as a burner of actinides from spent fuel. A burner/breeder could use thorium matrices.

Lead or lead-bismuth eutectic can be used as liquid metal coolants, exhibiting features particularly important for a fast reactor, even if their technology is less developed than the sodium technology. They are neutronically superior to other liquid metal coolants (i.e., capture less than sodium and are less activated), they are chemically inert with air and water, and they have very high boiling temperatures (1,737 °C and 1,670 °C, respectively, compared to 883 °C for sodium) and low vapour pressures. The resulting total core void reactivity coefficient is negative, unlike for sodium.

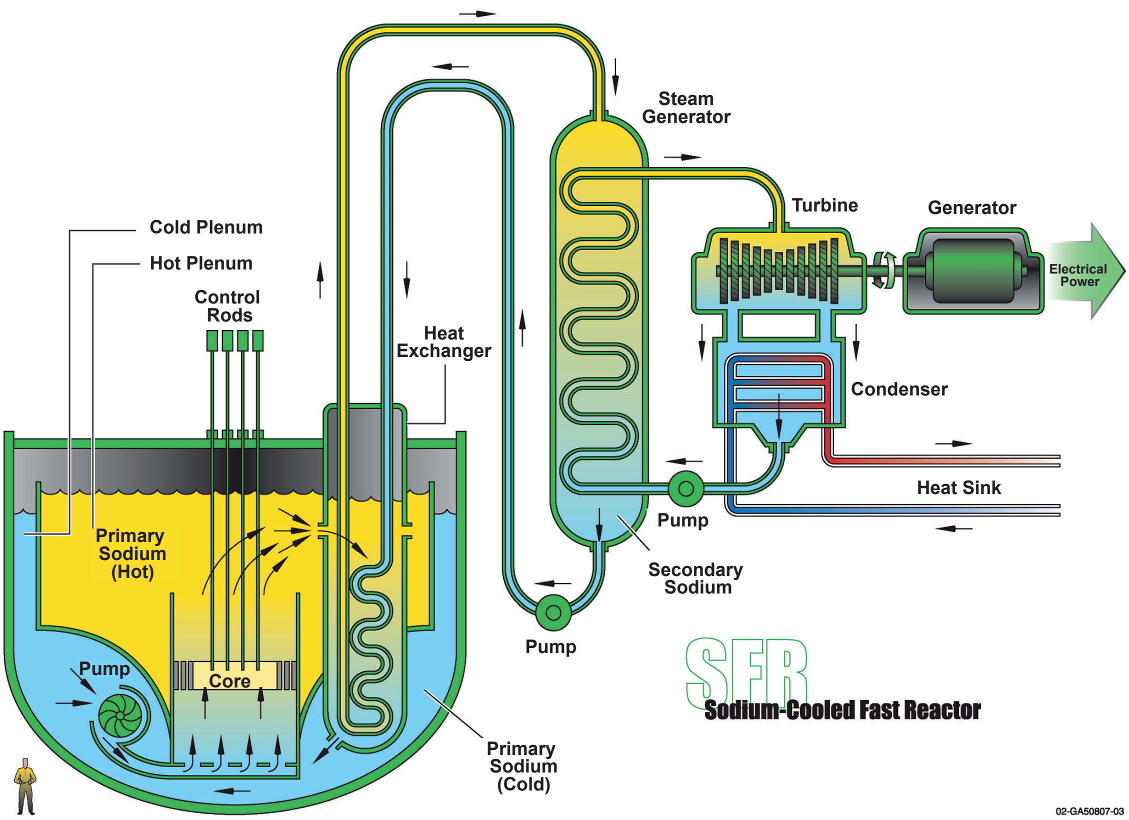

Figure 4: Sodium-Cooled Fast Reactor - GEN-IV Roadmap's pictorial view [2].

The use of lead or lead-bismuth coolants, however, raises some safety and reliability concerns, such as the corrosion of the structural materials and the production of volatile and radioactive ^{210}Po.

However, experience gained with lead-bismuth eutectic-cooled reactors in Russian nuclear submarines indicates that many of the technical problems can be overcome by adequate design and manufacture.

In currently envisioned GIF reference designs, lead is the coolant. A range of plant ratings are considered. The two options that comprise the dual-track approach currently being pursued under the GIF are the small transportable system of 10 - 100 MWe size that features a very long core life, and the larger system rated at about 600 MWe, intended for central station power generation, hereafter briefly described.

Small transportable “battery” system of 10-100 MWe unit size:

The reference is the US designed SSTAR (Small Secure Transportable Autonomous Reactor), a small natural circulation fast reactor, cooled by pure lead, with very long refuelling interval of up to 30 years. The compact active core is removed by the supplier as a single cassette and replaced by a fresh one, factory made. The system is rated at about 20 MWe, which can be scaled up to 180 MWe.

Central station reactor module of 300 - 600 MWe :

The reference is the European Lead-Cooled System (ELSY), a concept intended for power generation and waste transmutation. ELSY employs pure lead coolant, with forced cooling, and substantial design simplification in contrast with other liquid-metal-cooled reactors (e.g., integral steam generation and pumps). The system is rated at about 600 MWe, for power generation and waste transmutation.

The fuel options considered are oxide (ELSY) or nitride (SSTAR) based, containing fertile uranium and transuranics (minor and mixed actinides). In both cases the fissile enrichment is up to about 20%, and the burn-up up to 100 GWd/tHM.

6.4. LMCR PR and PP Intrinsic Features

The intrinsic proliferation resistance features of LMCR can be summarized as:

- High burn-up and hence high spent fuel radiological barrier (up to 150 GWd/tHM)

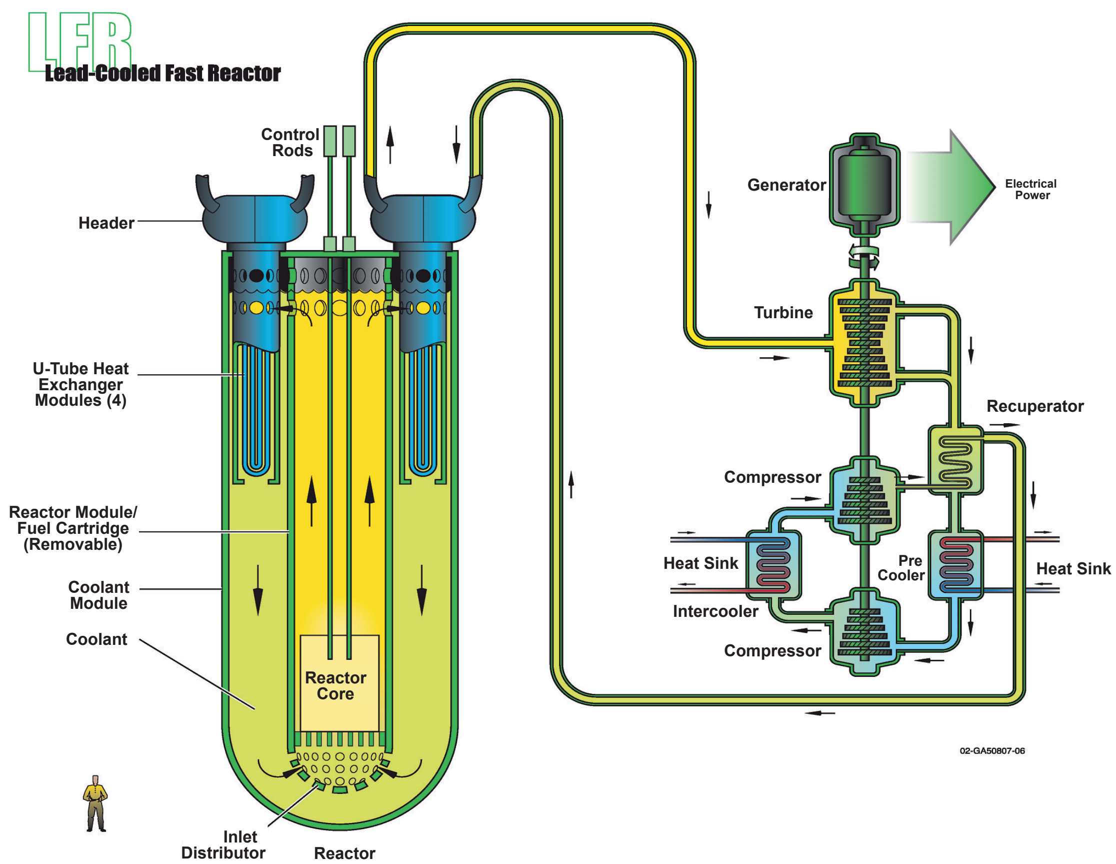

Figure 5: Lead-Cooled Fast Reactor - GEN-IV Roadmap’s pictorial view [2].

- Pyro-processing or advanced aqueous fuel reprocessing methods, with incomplete removal of fission products and minor actinides (see par. 9.2)
- No separation of uranium and plutonium at any fuel cycle stage, with either reprocessing method
- Inherently low decontamination factor of fuel, with need for remote handling, which complicates operations and enhances proliferation resistance.

The Pb-cooled "battery" (SSTAR) design, with a small size core with a very long core life attaining up to 30 years (typically 10 years) has good intrinsic proliferation resistance characteristics. The reactor module is factory made, and shipped to the plant site. It would require little action from the operators, and access to the fuel is only possible in the factory. In addition, other features of the LFR (shared by both the SSTAR and ELSY concepts) include [15]:

- Simple, compact core
- Low pressure operation
- Integral power conversion equipment
- No intermediate cooling system
- Lead coolant is non-reactive and has a high margin to boiling.
- Fast spectrum offers fuel cycle and materials management flexibility.
- Minor actinide fuel
- Natural circulation Decay Heat Removal (DHR).

Significant physical protection features of the LFR systems include:

- System simplification
- Non-reactive coolants
- Low pressure operation
- Passive decay heat removal
- Compact security footprint.

As for SFR, proliferation resistance issues are mainly linked to the presence of a radial or axial fertile blanket [15]. In the case of employing blankets, blanket assemblies are considered to be processed together with driver assemblies. Some ideas to enhance proliferation resistance in this context are investigated, e.g., a concept that generates low-grade plutonium in discharged blankets, reducing its attractiveness. PR&PP can be enhanced by innovative design, e.g., fuel handling machines considering issues of physical protection and of safeguards application.

7. Non-Conventional Reactor Systems

32 non-conventional innovative energy systems were considered by GIF's TWG4 [12], encompassing a wide variety of coolants and fuel designs.

The largest groups of NCRs are Liquid Core, Molten Salts and Gas Core reactors, besides other concepts like Direct Energy Conversion and Waste Minimization.

Of the 32 initial systems, only one Liquid Core Reactor System and two types of Molten Salt Reactors, also considered in the IAEA TECDOC's list, were selected for GIF's second screening, i.e., molten salt core and molten-salt-cooled reactors.

- N1-MSR Molten Salt Core Reactor (MSR), Thermal & Epi-thermal spectrum, 1,000 MWe
- N2-Liquid Core Reactor Systems
- N3-Molten-Salt-Cooled Reactor Systems (AHTR), 1,000 MWe.

N1 and N3 were eventually selected as MSR GEN-IV references.

7.1. GEN-IV Molten Salt Reactor

The two reference concepts under consideration have the main difference of having, respectively, fluid (i.e., fluoride salts) and solid (i.e., graphite-matrix) fuels [17].

In both cases, the heat generated in the molten salts is transferred to a secondary coolant system through an intermediate heat exchanger and then to the power conversion system, which could be a high-temperature steam cycle or possibly a helium gas turbine cycle.

Thanks to the very good cooling properties and higher thermal capacity of molten salts, compared to helium, MTRs can be built in larger sizes, at lower pressure, with smaller equipment.

Initially developed in the early 1950s in the USA, when a small test reactor was successfully operated, the MSR design has the potential to be a thermal ^{232}Th-^{233}U breeder power reactor with high thermal efficiency [12]. More recent concepts use a fast neutron spectrum core, with large negative temperature and void reactivity coefficients and a closed fuel cycle for the efficient utilization of plutonium and minor actinides, with full actinide recycle fuel. Therefore, besides operating as a thermal breeder reactor on a ^{232}Th-^{233}U fuel cycle, with very low resource demands, the MSR could be loaded with both Th and ^{238}U [20].

7.1.1. Fluid-fuelled MSR

This concept [Figure 6] uses a circulating mixture of fluorides (or nitrates) of sodium, lithium, beryllium and fissile materials, i.e., ^{233}U – Th; U-Pu-MA; Pu and Th.

The MSBR is the reference point for innovative fluid-fuelled MSR designs characterized by thermal-epithermal neutron spectrum, and a core with graphite moderator.

The Thorium Molten Salt Reactor (TMSR) is a recent example of fast MSR liquid core design in development, with a reference power level of 1,000 MWe, operating at low pressure (5 bar) and a coolant outlet temperature of 700-900 °C, allowing high thermal efficiency and potential for hydrogen production [20].

7.1.2. Molten-Salt-Cooled MSR

Molten-salt-cooled reactor designs are being investigated in the USA and Europe. The Advanced High-Temperature Reactor (AHTR) [refs. 15 and 21], is a thermal reactor using liquid salt coolants (lithium-beryllium and sodium-zirconium fluoride salts) and solid fuel, with the same graphite core structures and coated fuel particles of the modular high-temperature gas-cooled reactor concept (GT-MHR, see 5.1).

The AHTR concept still requires substantial development, but has potentially superior economical advantages, including potential for hydrogen production.

7.1.3. MSR Proliferation Resistance Intrinsic Features

Due to the scarce studies to date, in the GIF Roadmap the non-proliferation characteristics of a MSR were conservatively defined as being equivalent to an LWR. However, given the radically different characteristics of the MSR, there is a basis for an effective and significantly higher proliferation resistance, which needs to be better investigated. Some facts and comments follow.

Molten salt core reactors do not require fuel fabrication, which is a very expensive and difficult process for fuel including the minor actinides (americium and curium). Actinides and most fission products are directly formed, or added, in the liquid molten salt coolant and completely burnt. There are no burn-up constraints due to fuel integrity and handling concerns, however, there are limits to trifluoride solubility [22].

There is no spatial segregation of low burn-up material in the core, and added fuel (also coming from LWR spent fuel) is immediately diluted. The dilute concentration of actinides in molten salts also eliminates the handling of concentrated higher actinides, with their very high decay-heat generation rates. This would seem to lower the radiological barrier to proliferation. However, no subsequent recycling of actinides is foreseen, once they have been added to the molten salt.

The combined reactor and fuel cycle fissile inventory is low (~ 2 kg-fissile/MWe), because MSR is a reactor with a small critical mass and medium power density in the liquid. Moreover, the MSR fissile material isotopic composition would be unfavourable for use in weapons fabrication, i.e. *“Deep-burn”* plutonium.

The thorium fuel cycle is foreseen to reduce greatly the generation of higher actinides, compared to an LWR. It is claimed that the production of 1 TWh would require 100 kg of natural thorium for an MSR, instead of 20 tons of natural uranium with a PWR, with lower fissile inventories. The MSR would hence minimize the waste output [12].

Uranium isotopes dilution, or denaturing, in molten salts (^{238}U in ^{233}U) enhances proliferation resistance. However, on-site reprocessing could create an issue of possible “access to ^{233}U”: if thorium fuels were reprocessed on site soon after discharge, appreciable quantities of undecayed ^{233}Pa could still be present, and its relatively simple chemical separation could subsequently lead to ^{233}U by decay [15].

7.1.4 MSR Physical Protection Intrinsic Features

As said above, fissile material is diluted in massive volumes of highly radioactive salt, and can be easily denatured, also isotopically [15].

On-site reprocessing could provide opportunity for theft of concentrated ^{233}Pa-^{233}U. This is seen as a primary PR&PP threat.

As for resistance to sabotage, the MSRs have many passive safety features, besides the high thermal capacity and medium specific power (~22 kWth / l).

“Fluid fuel” fast MSRs would exhibit large negative temperature and void reactivity coefficients, which is a characteristic not found in solid-fuel fast reactors [17].

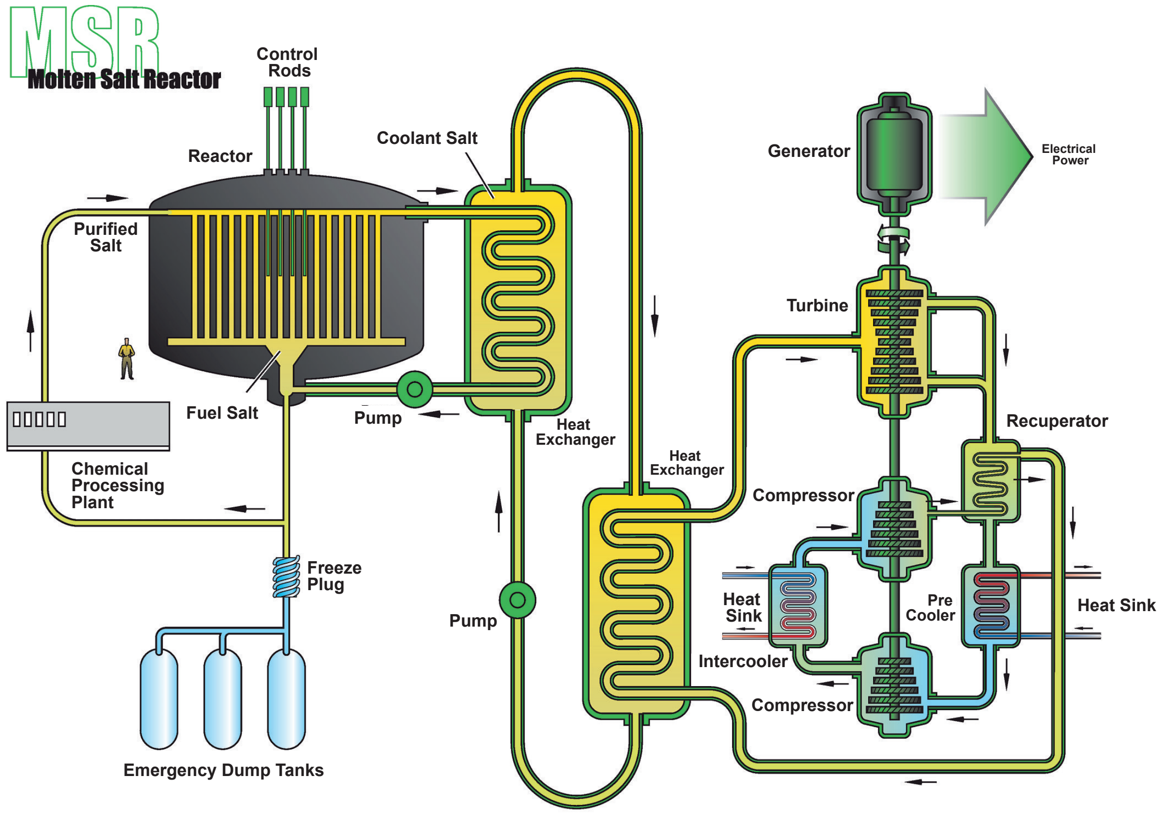

Figure 6: Molten Salt Reactor - GEN-IV Roadmap's pictorial view [2].

8. The Six GEN-IV Reference Concepts Summary

Table 2 summarises the main technical characteristics of the six reference concepts, with their options [17].

9. Discussion

9.1. Considerations on Intrinsic Physical Protection Features of Innovative Nuclear Systems

As recalled in Table 1 of par. 3.1, the GEN-IV Roadmap assessment considered physical protection according to the Vulnerability of Installations criterion, and Passive Safety Features metric.

The main objective of intrinsic physical protection is to limit the consequences of attacks aimed at theft of nuclear material or sabotage.

Extrinsic security measures can be adopted to protect the installation against both threats, whereas the design can rely on intrinsic physical protection features to improve the response to damage caused by sabotage.

In response to sabotage, a nuclear energy system should exhibit a high capability to withstand lack of coolant and, consequently, very severe temperature rise without core meltdown. To this end, refractory materials with high heat capacity and temperature resistance are a useful first passive structural feature, largely foreseen as we saw in gas-cooled reactor systems.

Decay heat removal should be possible without intervention of active systems, i.e., as much as possible with passive functions, without requirements of power. Cooling by natural convection as in LFR is a suitable option. Low-pressure fluids are preferable, to reduce safety issues in case of circuit break. These features are typical of liquid-metal-cooled reactor systems.

The location of nuclear system components should be in structures able to withstand attacks by means of energetic projectiles.

As much as possible, implementation of highly automated control systems is considered beneficial to safety, and provides resistance to sabotage, reducing the need for human intervention, with proper cross-checking of alarm parameters.

Based on the above, most of the 19 innovative nuclear energy system concepts selected for physical protection potential by the GEN-IV Roadmap were found to be similar or slightly better than the reference Advanced LWR. In particular, PBMR, GT-MHR, and SFR with metallic fuel were considered superior to ALWR for their high thermal inertia, heat

	Size [MWe]	Neutron Spectrum	Fuel / Fuel cycle	T out [°C]	Uses	Due by
GFR	~1,200	Fast	Fertile U, Actinide-carbides or nitrides, ceramic clad, or ceramic composite / Closed	850	Electricity, hydrogen	2025
LFR	• ~10-100 • ~300-600	Fast	Fertile U, TRU, nitride or metallic / Closed	480-567	Electricity, hydrogen	2025
MSR	1,000	Thermal/ Epitherm	Liquid mixture MA, U, FP, Na, Zr / Closed	700-850	Electricity, hydrogen	2025
SFR	1) 600-1500 2) 300-600 & 50-150	Fast	1) MOX (and MA) 2) U-Pu-MA-Zr metal Closed	550	Electricity	2015
SCWR	1,500	Thermal; Fast	U oxides / Open ; Closed	550	Electricity	2025
VHTR	600 MWth	Thermal	ZrC coated particles in blocks, pins or pebbles / Open	1,000	Hydrogen, heat, (electricity)	2020

Table 2: Summary of GEN-IV systems' main technical characteristics.

removal means and level of passive safety features.

9.2. Considerations on Intrinsic Proliferation Resistance Features of Innovative Nuclear Systems

According to the proliferation resistance metrics considered by the GIF Roadmap, the innovative nuclear energy system designs could roughly be grouped according to the following intrinsic proliferation resistance characteristics:

- Long-life inaccessible core
- Once-through high burn-up fuel not recycled
- High Burn-up spent fuel with pyro-processing,
- Low Burn-up spent fuel with aqueous reprocessing.

In particular, as we saw, we can identify some intrinsic features adopted by GEN-IV innovative nuclear energy system reference designs to increase proliferation resistance:

- Open cycle and high burn-up spent fuel, with no industrially available technology for reprocessing ceramic composite fuel element (VHTR)
- Modular factory-made cassette core inaccessible by the operator (LFR battery option)
- Complete burning and no recycling of actinides in molten salts (MSR)
- MOX or metal-bonded fuel charged with minor actinides (SFR).

The design choices made to pursue the sustainability, economics, safety and reliability, and PR&PP goals are all interdependent. In some cases, apparent benefits also have unexpected drawbacks. This is true also for proliferation resistance and physical protection, as we summarise hereinafter.

9.2.1. Fuel and Actinide Management

Most of the innovative nuclear energy system designs have a fast neutron spectrum primarily for burning minor actinides (Am, Np, Cm) and for breeding fissile material to improve the system sustainability.

The goal of this type of design is that of ensuring a self-sustainable operation or breed fuel to feed other reactors, and possibly reduce the need of fuel enrichment and related enrichment facilities, thus enhancing proliferation resistance at the global fuel cycle level.

To be intrinsically more proliferation resistant, and avoid potential diversion of separated plutonium, nuclear energy system designs should ideally foresee core concepts without fertile blankets. The created plutonium should be within the fissile fuel, and recycled by co-extraction with other actinides, without separation. Alternatively, fertile blankets charged also with minor actinides for the purpose of transmutation (and reduction of attractiveness) could be considered. The joint processing of driver and blanket assemblies would also add proliferation resistance features to the fuel cycle.

9.2.2. Burn-up

Intrinsic radiological barriers are certainly enhanced by longer irradiation and higher burn-up. High burn-up of fuel not only increases the specific energy production, but also degrades the plutonium isotopic composition and creates a higher radiological barrier to its handling, hence a higher technological difficulty for the proliferator. However, fuel handling becomes more complicated also for the operator, increasing costs and affecting economics.

In this respect, we found in the literature plutonium generation calculations as a function of burn-up showing that ^{239}Pu build-up with irradiation tends to slow down after 33 GWd/tHM, whereas the production of higher isotopes (and ^{238}Pu) continues to increase nearly linearly. For burn-up around 60 GWd/tHM, the isotope 239 percentage decreases from 60% to 50% of the total plutonium content (see [23], [24]). A further increase in burn-up would only slightly degrade the plutonium isotopics.

Higher burn-up reduces the frequency of fresh and spent-fuel handling and spent-fuel transportation requirements. By reducing spent-fuel inventories, it also reduces the total mass of associated plutonium, although this still remains substantial, and the need for on-site storage of spent fuel [25].

9.2.3. Fuel Reprocessing

The choice of fuel and fuel reprocessing have strong implications for proliferation resistance. Advanced techniques aim at improving proliferation resistance by avoiding separation of uranium and plutonium from minor actinides.

Evolving from the traditional PUREX method, where uranium and plutonium are separated with an industrial yield close to 99.9%, and minor actinides and fission products are conditioned in a glass matrix for interim storage and final disposal, the Advanced Aqueous process is characterized by uranium and plutonium co-extraction, along with most of the minor actinides, and no separation of plutonium at any stage of the process. This increases proliferation resistance of the system, because the processed material is still significantly contaminated through the presence of minor actinides.

This results in too high a radiation activity and radiological barrier to use a simple glove-box facility for fuel fabrication, hence complicating the operations. Moreover, due to the shorter half-life of fission products compared to the heavier actinides that are recycled and burned, waste radio-toxicity is reduced in time and inventory by up to a factor of 100.

Pyro-processing, developed since the 1980's, has only reached the pilot-scale stage, and it is claimed to be more compact, less complex, less costly and generating less waste streams than the conventional aqueous (PUREX) process used for oxide fuel [11].

Pyro-processing of metal fuel is based on a few process steps by electrochemical dissolution (electro-refining) in a molten salt eutectic. Uranium, plutonium and other actinides are co-extracted, and there is no recovery of pure fissile material at any stage of the process. The method could also be applied to MOX fuels [26].

9.2.4. Facility Locations

As for the locations of recycle facilities, various options with PR&PP advantages and drawbacks are under consideration. Integration of the fuel cycle in the nuclear site would minimize both transport of nuclear materials (restricted to make-up fuel) and the total amount of nuclear materials needed for the lifetime operation of the system, as the fissile fuel needed is bred in situ from fertile fuel. However, application of safeguards to many reprocessing plants would become more complicated.

Regional or centralized reprocessing facilities would instead facilitate centralized application of safeguards, but require nuclear material transportation at various distances, hence raising physical protection issues.

The attractiveness of the type of feed fuel is another factor to be considered, both from proliferation resistance and physical protection points of view. Besides the isotopic composition of spent fuel, with its radiological barrier, fresh material could also be a possible proliferation target, especially MOX fuel, which is categorized direct use by the IAEA.

Acknowledgements

For this update, some use has been made of the information released in the workshop held at Brookhaven National Laboratory, with the participation of GIF PR&PP WG members and GIF Systems Steering Committees (SSC) representatives [15]. The authors wish to thank the presenters in the workshops and particularly H. Khalil of ANL and C. Smith of LLNL for their useful comments.

Acronyms

ABWR	Advanced Boiling Water Reactor
AHTR	Advanced High Temperature Reactor
AHWR	Advanced Heavy Water Reactor
ALWR	Advanced Light Water Reactor
BWR	Boiling Water Reactor
ELSY	European Lead-cooled System
FIMA	Fissions of Initial Metal Atoms
FFTF	Fast Flux Test Facility (Hanford site, U.S.)
FP	Fission Product
GCR	Gas-Cooled Reactor
GFR	Gas-cooled Fast Reactor
GNEP	Global Nuclear Energy Partnership
GT-MHR	Gas Turbine Modular Helium Reactor
HEU	High-Enriched Uranium
HTTR	High Temperature engineering Test Reactor
IMR	Integrated Modular water Reactor
IRIS	International Reactor Innovative and Secure
JSFR	Japan Sodium Fast Reactor
KALIMER	Korea Advanced Liquid Metal Reactor
LEU	Low-Enriched Uranium
LMFBR	Liquid Metal Fast Breeder Reactor (an archaic term for fast reactors)
LMR	Liquid Metal Reactor
LOCA	Loss Of Coolant Accident
LWR	Light Water Reactor
MA	Minor Actinides
MARS	Multipurpose Advanced Reactor, inherently Safe
MOX	Mixed uranium-plutonium OXide (fuel)
MSR	Molten Salt Reactor
PBMR	Pebble Bed Modular Reactor (in South Africa)
PBR	Pebble Bed Reactor
PFR	Prototype Fast Reactor
PMR	Prismatic fuel Modular Reactor
PRISM	Power Reactor, Inherently Safe Module
PWR	Pressurized Water Reactor
RBEC-M	Lead-Bismuth Cooled Reactor with high level of natural circulation
RMWR	Reduced Moderation Water Reactor
SMR	Small and Medium Reactors
SCWR	Super Critical Water cooled Reactor
SSTAR	Small Secure Transportable Autonomous Reactor
SMART	System-integrated Modular Advanced Reactor
TMSR	Thorium Molten Salt Reactor
TRISO	Refractory (coated particle fuel)
TRU	Transuranic (isotopes)
TWG	Technical Working Group
VBER-300	Water Cooled Modular Power Reactor
VHTR	Very-High-Temperature Reactor

References

[1] International Nuclear Fuel Cycle Examination (INFCE), *Report of the First Plenary Conference of the International Nuclear Fuel Cycle Examination (INFCE)*, International Atomic Energy Agency, Vienna (1980).

[2] US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, A Technology *Roadmap for Generation IV Nuclear Energy Systems*, GIF002-00, December 2002. http://gif.inel.gov/roadmap/

[3] International Atomic Energy Agency (IAEA), *Proliferation Resistance Fundamentals for Future Nuclear Energy Systems*, IAEA Department of Safeguards, STR-332, IAEA, Vienna (2002).

[4] GIF PR&PP Expert Group et al., *Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems* - Revision 5, 2006; GIF/PRPP-WG/2006/005. http://www.gen-4.org/Technology/horizontal/PRPPEM.pdf

[5] International Atomic Energy Agency, *Methodology for the Assessment of innovative nuclear reactors and fuel cycles*, IAEA-TECDOC-1434, IAEA, Vienna, 2002.

[6] US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, *Report of the International Workshop on Technology Opportunities for Increasing the Proliferation Resistance of Global Civilian Nuclear Power Systems (TOPS)*, NERAC March 29–30, 2000, Washington.

[7] GIF PR&PP Expert Group et al., *Addendum to the Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems*, Technical Addendum to Revision 5, 2007, GIF/PRPPWG/2006/005-A, Pre-release draft.

[8] Cojazzi, G.G.M., Renda, G., Sevini, F., Proliferation Resistance Characteristics of Advanced Nuclear Energy Systems: a Safeguardability point of view, *Proceedings of 29th ESARDA Annual Meeting Symposium on "Safeguards and Nuclear Material Management"*, Aix en Provence, France, May 22-24, 2007. See also the updated paper in this special issue.

[9] US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, *Generation IV Roadmap, Description of Candidate Water-Cooled Reactor Systems Report*, GIF-015-00, December 2002. http://gif.inel.gov/roadmap/pdfs/015_description_of_candidate_water-cooled_reactor_systems_report.pdf

[10] US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, *Generation IV Roadmap, Description of Candidate Gas-Cooled Reactor Systems Report, GIF-016-00, December 2002.* http://gif.inel.gov/roadmap/pdfs/016_description_of_canddate_gas-cooled_reactor_systems_report.pdf

[11] US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum*, Generation IV Roadmap, Description of Candidate Liquid-Metal-Cooled Reactor Systems Report, GIF-017-00, December 2002.* http://gif.inel.gov/roadmap/pdfs/017_descrp_of_liquid_metal_final_report_d1.pdf

[12] US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, *Generation IV Roadmap, Description of Candidate Nonclassical Reactor Systems Report*, GIF-018-00, December 2002. http://gif.inel.gov/roadmap/pdfs/018_description_of_candidate_nonclassical_reactor_systems_report.pdf

[13] US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, *Generation IV Roadmap Final System Screening Evaluation Methodology R&D Report* Issued by the Nuclear Energy Research Advisory Committee and the Generation IV International Forum December 2002, GIF-012-00. http://gif.inel.gov/roadmap/pdfs/000_contents.pdf

[14] International Atomic Energy Agency, *Status of Innovative small and medium Sized Reactor designs 2005, Reactors with Conventional refuelling Schemes*, IAEA-TECDOC-1485, March 2006.

[15] GIF SSC Representatives, *Presentations at the GIF PR&PP-SSC Joint Meeting*, Brookhaven National Laboratory, USA, May, 19-20, 2008.

[16] Rodriguez, C., et al., Deep-Burn: making nuclear waste transmutation practical, *Nuclear Engineering and Design* 222 (2003) 299–317.

[17] *GIF, Generation IV International Forum, 2007 Annual Report*

[18] US DOE, *Global Nuclear Energy Partnership Strategic Plan*, GNEP-167312, Rev. 0, Office of Nuclear Energy, Office of Fuel Cycle Management, January 2007.

[19] *Roadmap for a European Innovative Sodium cooled Fast Reactor – EISOFAR*, EURATOM SIXTH FRAMEWORK PROGRAMME, Specific Support Action Contract Number 044824.

[20] Charles W. Forsberg, Thermal- and Fast-Spectrum Molten Salt Reactors for Actinide Burning and Fuel Production, *Proceedings of Global 07 conference: Advanced Nuclear Fuel Cycles and System*, ANS Manuscript Number: 175768, American Nuclear Society, September 9–13, 2007, Boise, Idaho, USA

[21] Charles W. Forsberg, Per F. Peterson, Larry Ott, The Advanced High-Temperature Reactor (AHTR for Producing Hydrogen to Manufacture Liquid Fuels, 2004 *Americas Nuclear Energy Symposium*, American Nuclear Society, October 3-6, 2004, Miami Beach, Florida, USA

[22] Y. Hirose and Y. Takashima, The Concept of Fuel Cycle Integrated Molten Salt Reactor for Transmuting Pu+MA from Spent LWR Fuels, *Proceedings of Global '01 conference*, September, 2001, Paris, France,

[23] Kankeleit, E., *Report on the Usability of Reactor Plutonium in Weapons*, Institute for Kernphysik, Technische Hochschule Darmstadt, 1989, translated as LLNL reference 04191.

[24] Carson Mark, J., Explosive properties of Reactor-Grade Plutonium, *Science & Global Security*, 1993, Vol. 4, pp.111-128.

[25] Hassberger, J. A., *Application of Proliferation Resistance Barriers to Various Existing and Proposed Nuclear Fuel Cycles*, U.S. Department of Energy, LLNL, UCRL-ID-147001, Oct. 2001.

[26] Madic, C., Overview of the Hydrometallurgical and Pyro-Met allurgical Processes Studied Worldwide for the Partitioning of High Active Nuclear Wastes, *Proceedings of 6th Informa tion Exchange Meeting on Actinide and Fission Product Parti tioning and Transmutation*, 11-13 December 2000, Madrid (Spain).

Non-proliferation Aspects of Advanced Fuels under Light Water Reactor Conditions

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Abstract

In the paper, different nuclear fuel cycles of advanced fuel types and their non-proliferation aspects are examined and compared. The investigated fuels include mixed oxide fuel, thorium-based fuel and zirconium inert matrix fuel. All of them are used to carry and burn or transmute plutonium created in the classical UOX cycle. The computing cycles are based on reprocessing of spent UOX fuel, separation of plutonium, fabrication of an advanced fuel type and its reuse in a light water reactor. Minor actinides are separated along with plutonium only in the case of the inert matrix fuel. The calculated and compared values include plutonium and minor actinides transmutation rates, mass of reprocessed fuel and mass of fuel sent to the repository. All fuel cycles were calculated using the HELIOS 1.9 spectral code.

Keywords: plutonium transmutation, MOX fuel, inert matrix, thorium-based fuel.

1. Introduction

Implementation of advanced nuclear fuel cycles under real operating conditions requires consistent studies of fuel material composition changes under neutron irradiation. This work examines plutonium and minor actinides changes using advanced nuclear fuel in light water reactors.

The analysed cycles are based on reprocessing of spent uranium-oxide (UOX) fuel burned in the VVER-440 type reactor under normal operating conditions, separation of plutonium, fabrication of an advanced fuel type and its reuse in the same reactor type, i.e., VVER-440. Minor actinides are separated along with plutonium only in the case of the inert matrix fuel. Detailed information about the cycles is given in the following sections.

The calculated and compared values include plutonium and minor actinides transmutation, the mass of reprocessed fuel and the mass of fuel sent to the repository. All fuel cycles were calculated using the HELIOS 1.9 spectral code.

2. Advanced Fuel Types and their Cycles

Advanced fuel types have several advantages over classical uranium fuel (UO_2 , UOX) used nowadays worldwide. The advantages are the capability to transmute plutonium and minor actinides to non-radioactive nuclei or to nuclei with shorter decay times and proliferation resistance [1]. This work is focused on the transmutation potential of mixed oxide fuel, inert matrix and thorium-based fuel. All of them should be applicable in power reactors under conditions similar to those of UOX fuel.

2.1. Mixed Oxide Fuel Cycle

Mixed oxide fuel (MOX) is a well-known type of fuel prepared by mixing separated plutonium oxide with uranium U-238 including a small amount of uranium U-235. The analysed MOX fuel cycle is characterized as follows: Natural uranium is enriched and burnt in a light water reactor in the same way as in the case of the open fuel cycle (OFC), up to the target burn-up of 50,000 MWd/tHM. After a cooling time of 5 years the spent fuel is reprocessed and the plutonium is separated. The reprocessing calculations take into account 0.1% plutonium losses. Separated plutonium is then mixed with depleted uranium U-238 (with a tails essay of 0.25% U-235). To reach a similar multiplication capability as with UOX fuel, the content of plutonium is set to 8.5%. The plutonium isotopic vector in spent UOX fuel is summarized in Table 1, the MOX fuel cycle is shown in Figure 1.

Pu isotope	% composition
Pu-238	2.78
Pu-239	55.46
Pu-240	23.20
Pu-241	12.16
Pu-242	6.40

Table 1: Plutonium isotopes content in the plutonium vector of the spent UOX fuel.

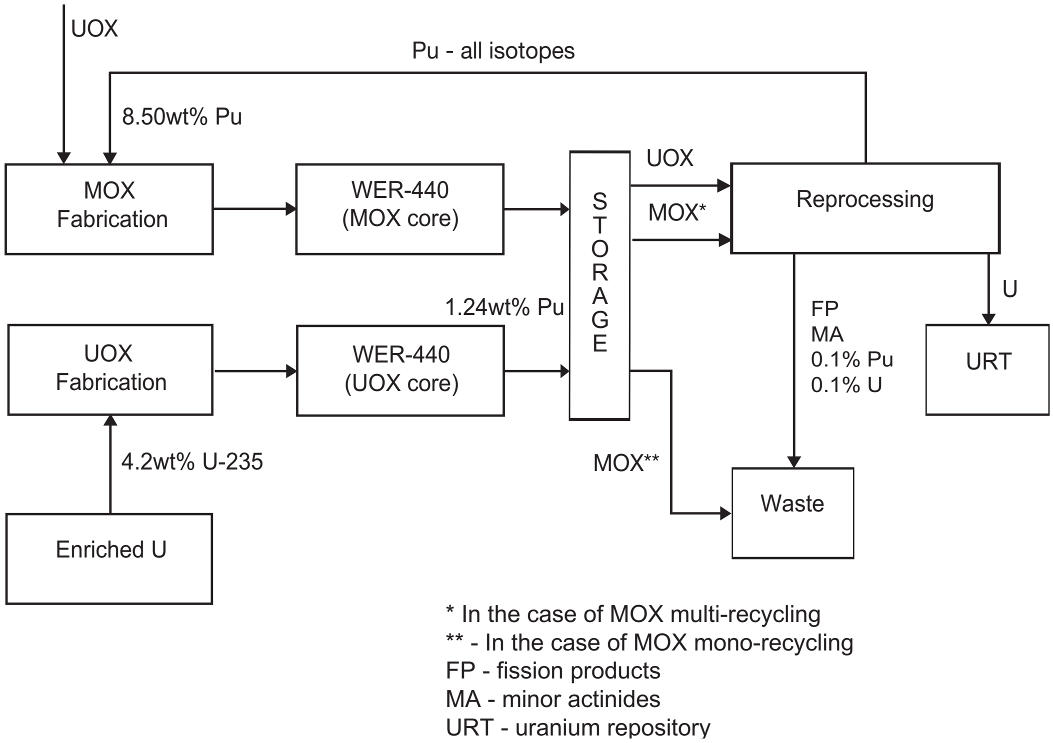

Figure 1: MOX fuel cycle.

The fuel cycle is analysed in two different variants. The first one is mono-recycling of plutonium, which means that plutonium is separated only once from UOX spent fuel and burnt in MOX fuel.

Burnt MOX fuel is then considered as waste and stored. The second case is multi-recycling of plutonium to the equilibrium state. Burnt MOX fuel is reprocessed and plutonium is separated and burnt again in fresh MOX fuel, until there is no material difference between burnt MOX fuel from two subsequent cycles.

Multi-recycling of plutonium in MOX fuel causes changes of the plutonium vector from one cycle to the next. The plutonium vector in the MOX equilibrium state is shown in Table 2. The plutonium content in MOX fuel for mono-recycling and also for multi-recycling of plutonium was set, as mentioned above, to 8.5%, while the plutonium isotopic vector was set according to Table 1 for mono-recycling of plutonium, and according to Table 2 for multi-recycling of plutonium.

Pu isotope	% composition
Pu-238	0.68
Pu-239	11.41
Pu-240	7.12
Pu-241	1.13
Pu-242	79.65

Table 2: Plutonium vector in MOX equlibrium cycle.

2.2. Thorium-based Fuel

The thorium-based fuel cycle with plutonium content (ThPu) is similar to the one of MOX fuel, the only difference being in mixing the separated plutonium oxide with thorium oxide. The case of mono-recycling has been analysed. The plutonium content in thorium fuel was estimated to reach values of the multiplication factor similar to that reached in the case of the UOX open fuel cycle. Several plutonium contents were calculated and, finally, a plutonium content with 5.40% Pu-239 was chosen for this fuel cycle. The other plutonium isotopes which are present in the spent UOX fuel are corresponding to Table 1. The total amount of plutonium in the fuel is 9.74%. The scheme of the fuel cycle is shown in Figure 2.

2.3. Inert Matrix Fuel Cycle

Inert matrix fuel (IMF) is a fuel prepared by mixing separated plutonium and minor actinides into an yttria-stabilized zirconium matrix. The advantage of the inert matrix fuel is the proliferation resistance against outside impacts. The fuel cycle is similar to the MOX fuel cycle: Natural uranium is enriched and burnt in a light water reactor in the same way as described before, up to a target burn-up of 50,000 MWd/tHM and, after a cooling time of 5 years, the spent fuel is reprocessed. Separated plutonium and minor actinides are then mixed with the zirconium inert matrix and inserted into a fresh assembly. To

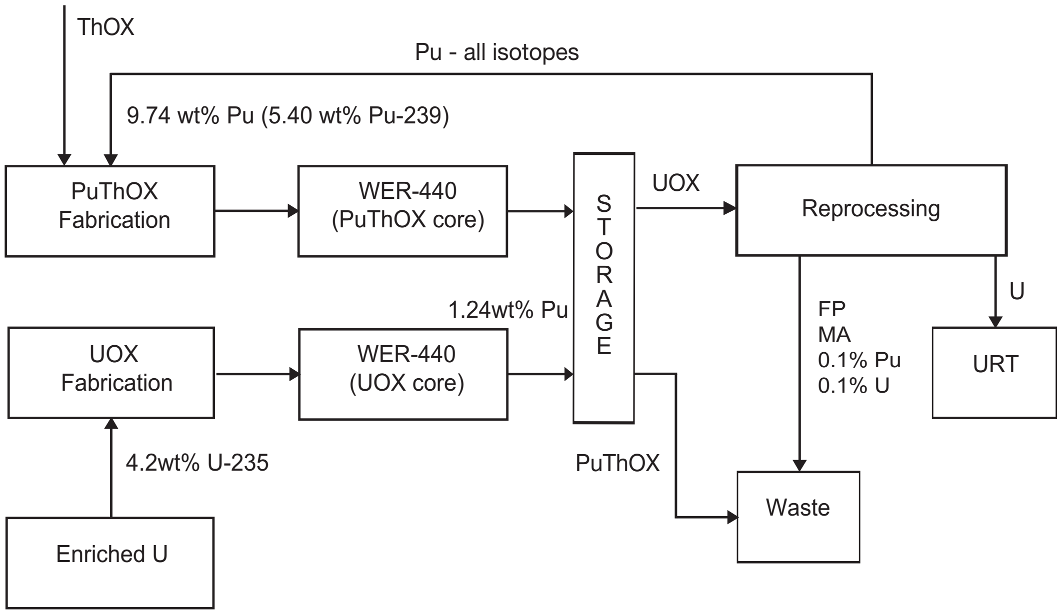

Figure 2: Thorium based fuel cycle.

ensure symmetrical distribution of power loading, a new type of fuel assembly was modelled. Detailed information about computation models is given in the next chapter.

The analysed fuel cycle with inert matrix fuel is operated in the self-cleaning manner. Separated plutonium and minor actinides from one burnt UOX assembly are loaded into one advanced assembly to selected pins. The fuel cycle calculations account for 0.1% Pu and MA losses during the separation process. There is no multi-recycling of the iner matrix fuel. A scheme of the advanced fuel cycle is shown in Figure 3.

More information about fuel cycles can be found in ref. [2], [3].

3. Computation Models

Two models of VVER-440 fuel assemblies were prepared. The first one, shown in Figure 4a, is the VVER-440 assembly with one type of fuel pins, only. This fuel assembly was used for calculations of the MOX as well as thorium-based fuel cycles. To perform calculations with inert matrix fuel, an advanced VVER-440 fuel assembly was prepared, shown in Figure 4b, with two different fuel types. The dark pin positions indicate the advanced fuel, all the rest are fresh UOX pins.

The fuel assemblies are computed in an infinite lattice – neutrons which escape from one surface of the fuel assembly and enter into the assembly at another surface. The models were prepared and fuel cycles were calculated using the HELIOS 1.9 spectral code [4].

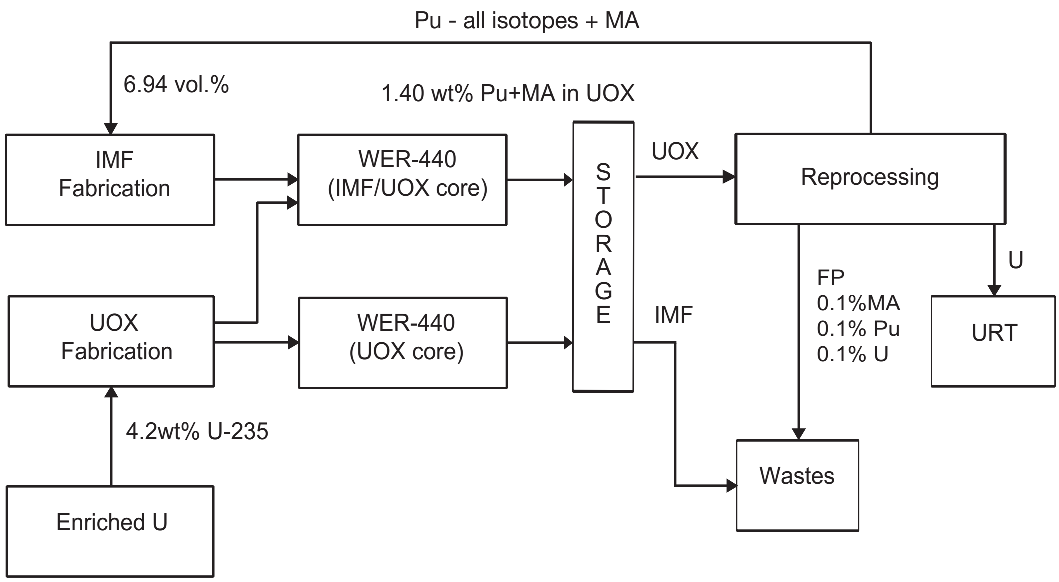

Figure 3: Inert matrix fuel cycle.

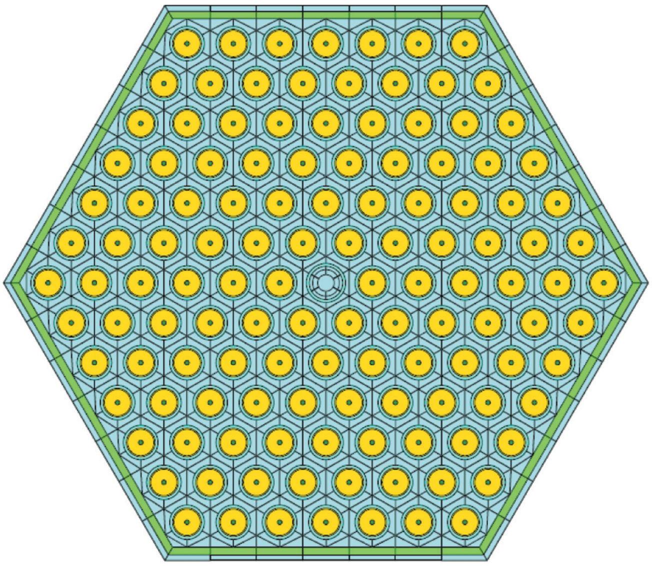

Figure 4a: Model of WER-440 assembly

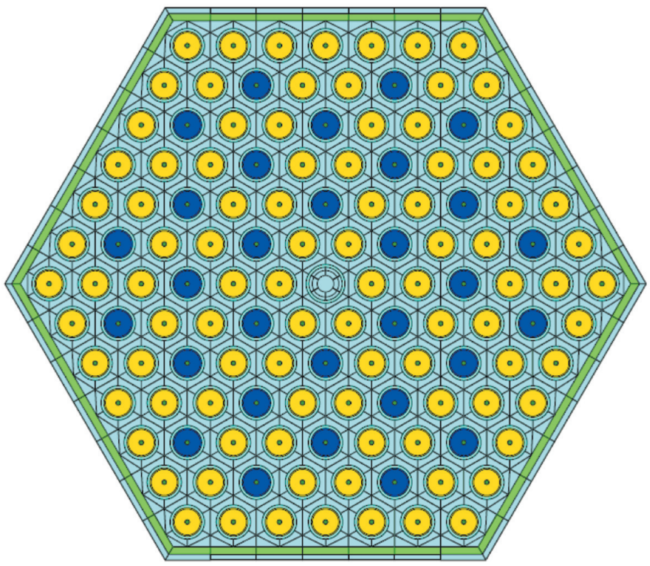

Figure 4b: Model of advanced WER-440 assembly

The target burn-up is the same for all types of fuels: 50,000 MWd/tHM in 5 cycles of 320 days each.

4. Results

Equilibrium advanced fuel cycle calculations were performed. Figure 5 shows the results gained for multiplication factors in different fuel cycles compared with the open fuel cycle.

From Figure 5 it can be seen that multi-recycling of MOX fuel cannot be applied in a light water reactor. The values of the multiplication factor for this equilibrium cycle are very low. It is recommended to perform no more than three recycling steps and then to consider the burned MOX fuel as waste. In order to manufacture advanced nuclear fuel for one reactor core, fuel assemblies from more than one UOX reactor are needed due to the required higher plutonium content. It is only the inert matrix fuel cycle that works in the self-cleaning cycle, which means that all plutonium created in one UOX/IMF core is separated and recycled into fresh IMF fuel

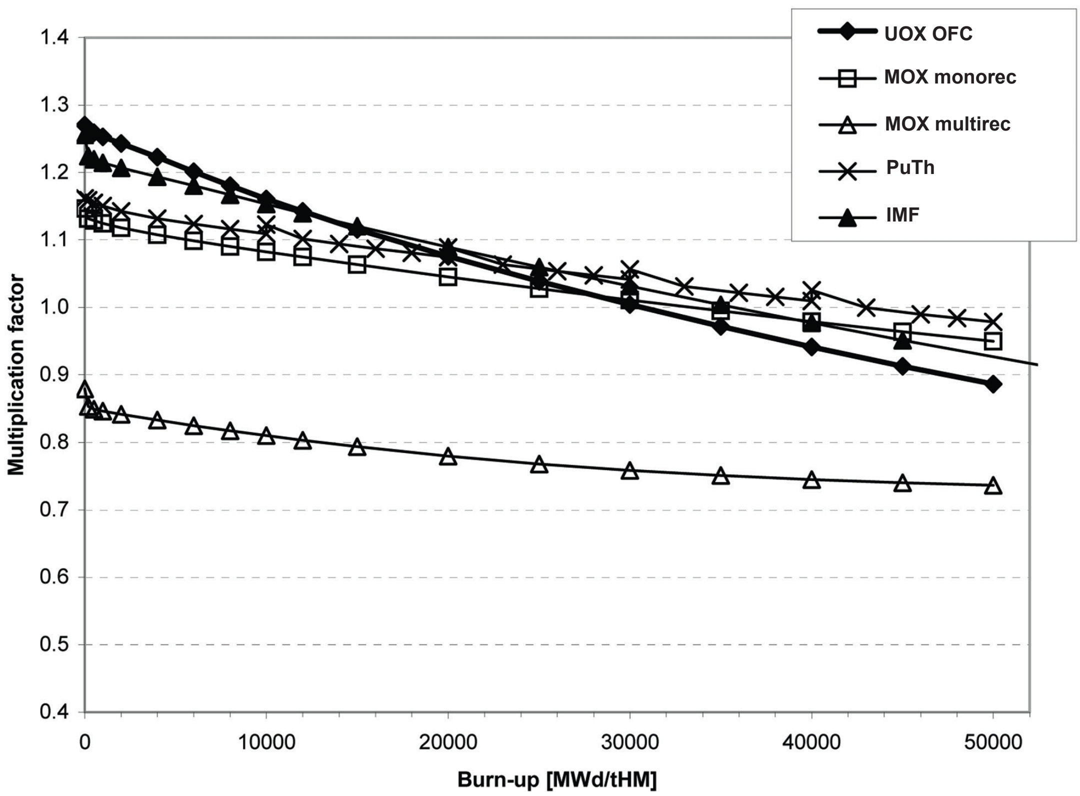

Figure 5: Multiplication factor for different fuel cycles in comparison with UOX open fuel cycle.

for one UOX/IMF core. The numbers of feeding reactors for other fuel cycles are not negligible.

Table 3 summarizes the total number of UOX reactors needed to operate the equilibrium fuel cycle. In the case of burning weapon plutonium or plutonium stored in spent fuel worldwide, no feeding reactor is needed to operate these fuel cycles. Table 3 also summarizes the initial content and content of plutonium and minor actinides in the spent fuel and transmutation rates for Pu and MA, masses of fuel entering reactors and masses of fuel sent to a repository with the total amount of finally disposed plutonium.

The total amount of finally disposed fuel is somewhat misleading. To operate one MOX reactor, almost 7 UOX reactors are needed (to produce enough plutonium to reach 8.5% plutonium content in the fresh MOX fuel) from which all plutonium, excluding 0.1% losses, is separated and used in MOX fuel. Hence, the higher values of the total amount of disposed plutonium include plutonium from these feeding reactors. In the case of burning plutonium from spent fuel storage or weapon plutonium these values should be lower in comparison with UOX open fuel cycle.

5. Conclusions

An overview of several advanced nuclear fuel cycles taken into account in sustainability evaluations is given. The total amount of disposed plutonium and high level waste can be reduced by introducing advanced fuel types into power reactors.

From the point of view of the total amount of finally disposed plutonium, the inert matrix fuel cycle seems to be the best choice for the equilibrium cycle.

Advanced fuel types have the potential to enhance proliferation resistance against impacts from outside. The problem of proliferation is related to the step of separation of plutonium and adding of plutonium to the fuel matrix. The highest volume of plutonium is separated in the thorium fuel cycle, the lowest one in the case of the inert matrix fuel. From this point of view the inert matrix fuel cycle is also the best choice [5].

Advanced fuel types can be operated under light water reactor conditions. They can participate in transmutation of cumulated plutonium and also in plutonium production reduction. Introduction of advanced cycles into power reactor practice requires detailed studies of operational and other characteristics.

References

[1] P. DAŘÍLEK, V. NEČAS, V. ŠEBIAN – LWR *Fuel Cycle with Reduced HLW Production*, AIP Conference Proceedings, International Conference on Nuclear Data for Science and Technology, 26th Sept – 1st Oct 2004, Santa Fe, New Mexico, pp. 1458-1461.

[2] L. BOUCHER, at al.: DRAFT - *Definition of Technical Data and Detailed hypotheses*, RED-IMPACT, November 2005.

[3] J. BREZA, R. ZAJAC, P. DAŘÍLEK, V. NEČAS - *PWR and VVER thorium cycle calculation,* 16th AER Symposium, 25-29 September 2006, Bratislava, Slovakia

[4] HELIOS documentation, Studvisk Scandpower, April 2000, Norway.

[5] V. NEČAS, V. ŠEBIAN, K. KOČÍŠKOVÁ, P. DAŘÍLEK – *Radiotoxicity and Risk Reduction of TRU Elements from Spent Fuel by Transmutation in the Light Water Reactor*, AIP Conference Proceedings, International Conference on Nuclear Dara for Science and Technology, 26th Sept – 1st Oct 2004, Santa Fe, New Mexico, pp. 1474-1477.

	UOX	MOX monorec	MOX multirec	PuTh	IMF
Number of feeding reactors	–	6.82	6.82	7.87	–
Pu and MA initial content [wt %]	0.00	8.50	8.50	9.74	1.41
Pu and MA content in burned fuel [wt %]	1.22	6.74	8.33	5.75	0.46
Pu transmutation rate [Kg/TWhe]	0.00	51.06	35.26	104.44	30.72
MA transmutation rate [Kg/TWhe]	0.00	0.00	0.00	0.00	0.47
Mass of reprocessed fuel [t/TWhe]	0.00	2.4	2.40	0.00	2.39*
Average quantity of separated Pu [Kg/TWhe]	0.00	118.63	141.38	221.17	41.19
Amount of finally disposed Pu [Kg/TWhe]	26.02	118.63	141.38	116.73	10.48

* - **UOX** from UOX/IMF core is reprocessed.

Table 3 . Comparison of selected parameters of advanced nuclear fuel cycles.

Safeguards Instrumentation for Future Nuclear Fuel Cycles

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Abstract

In the frame of the Generation IV International Forum (GIF), the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), and, more recently, the Global Nuclear Energy Partnership (GNEP), experts discuss the future use of nuclear power by addressing a variety of areas, ranging from new nuclear reactor technologies to international fuel cycle models. Aside from economic and inherent safety issues, considerations on proliferation resistance have gained increased international attention and importance for the feasibility of nuclear fuel supply and fuel cycle services models. Proliferation resistance is ruled by both intrinsic and extrinsic factors. Intrinsic factors are related to the quantities and quality of nuclear materials used in any given nuclear facility and the ease with which both materials and technologies could be withdrawn from the installation. Extrinsic features stem from institutional barriers against diversion or misuse and relate mainly to the application of International Atomic Energy Agency (IAEA) safeguards. Looking forward towards future nuclear technologies, the question arises how these will impact the future safeguards culture and supporting instrumentation. The paper addresses this question and discuss some future aspects of safeguards by extrapolating and expanding on the evolution of safeguards from a material and technology based control system to an information driven approach. Furthermore, factors will be outlined that may impact not only the development and implementation of future safeguards instrumentation, but also the design of future nuclear reactors. Features of safeguards instrumentation may be ranging from remote interrogation capabilities to multipurpose, synergy-enabling functions, i.e., the consequences of an expected increase in a global nuclear market within a 'nuclear renaissance' on future safeguards instrumentation will be highlighted. Also, the need for early involvement of all concerned parties, especially treaty verification authorities, will be discussed. Considerations on how the non-proliferation community can best become prepared for the technological needs of the future will conclude the paper.

Keywords: proliferation resistance, safeguards, technologies, nuclear renaissance.

1. Introduction

The high cost and limited availability of fossil fuel resources as well as climate change concerns have prompted government leaders world-wide to review their nuclear power generation programs or to investigate avenues to initiate such activities. Under the banner of a Nuclear Renaissance, industrial players support such tendencies by promoting the inherent security of modern fuel cycle facilities and nuclear reactors and by introducing new, advanced means of reducing both the amount and the danger of the generated nuclear waste.

At the same time, the international community along with non-proliferation authorities strives to guide the expanding use of nuclear energy within the spirit of Atoms for Peace and the Non-Proliferation Treaty (NPT) by enhancing the control and the safeguarding of sensitive technologies and materials. In this context, multi-national approaches including the Global Nuclear Energy Partnership (GNEP) as well as fourth generation, proliferation resistant nuclear reactors under GIF and INPRO have been much discussed in recent years.

Neither the establishment of new, multi-national fuel cycle models nor the design, development, and construction of new reactors is a short-term endeavor; the international community is planning for what is to come in two decades and beyond. This horizon naturally poses a broad range of challenges, especially when addressing the question of how it will impact non-proliferation policy and the verification of safeguards commitments compliance, but it also offers some opportunities worth exploring.

The NPT verification authority, the International Atomic Energy Agency (IAEA), has the mission to verify the correctness and completeness of signatory states' declarations about their peaceful nuclear programs. To this end, IAEA inspection personnel has access to declared facilities to check, applying the most effective and efficient combination of all safeguards measures. One such measure that supports inspectors is safeguards instrumentation that is either installed operating in unattended mode at nuclear facilities or is carried into the field for attended operation. Under the Additional Protocol, IAEA inspectors' access rights include searching for undeclared nuclear facilities and activities.

Looking ahead towards the implementation of new fuel cycle models and the accompanying increased safeguards responsibilities, the future cousins of today's instrumentation are likely to play an increasingly important role. This is an area where some appealing opportunities can be realized if the opposing challenges can be successfully addressed. The following paper will first investigate some of the factors that will impact the development of future non-proliferation policy to show the multi-dimensional nature of the related challenges. It will then explore what features need to be designed into instrumentation to support meeting such challenges. Some questions and concerns on the future instrumentation development path will be raised next. Recommendations, as far as it is possible at this early stage, as to how the international community can best approach the development task will conclude the paper.

2. Factors Potentially Impacting Safeguards

The term Nuclear Renaissance implies a significant increase in the use of nuclear power world-wide, and indeed more and more countries have identified atomic energy to be a viable addition to or expansion of their electricity portfolio to counter increased prices of fossil fuels and climate change concerns. Further, the globally progressing industrialization increases the need for baseline electricity production capacities. This future expansion of nuclear activities will likely lead to the globalization of the nuclear energy market which will also affect international non-proliferation policy.

One push is to concentrate sensitive nuclear technologies such as enrichment and reprocessing by introducing Multi-National fuel cycle Approaches (MNAs) where a few supplier countries offer nuclear services to recipient countries. Their success will depend heavily on how much assurance of supply of nuclear fuel to recipient countries such models can credibly guarantee; otherwise, the incentive to develop national fuel production capabilities will remain. The implementation of MNAs leads to safeguards being applied in Nuclear Weapons States (NWS) if they are a supplier country as they take on fuel cycle services for a Non Nuclear Weapon State (NNWS) that is under full scope safeguards obligations. Such a shift might support a different movement towards the implementation of comprehensive safeguards as a universal standard for all NPT signatory states, including NWS.

The implementation of safeguards is not a static approach, but rather of a very dynamic nature with the flexibility to adapt to changes within the non-proliferation regime and treaty compliance verification efforts. One such currently ongoing transition is driven by the implementation of the Additional Protocol and Integrated Safeguards. In the practical sense, this means that the safeguards system is shifting from a quantifiable declaration-and-verification regime to a more information-driven, qualitative approach with extended access rights on the part of the IAEA (Complementary Access (CA)). In the effort to verify both the correctness and the completeness of a NPT signatory state's declarations, traditional safeguards measures are re-evaluated and complemented by other information sources to detect undeclared materials and activities in addition to diversion and misuse of declared ones.

This is a significant shift in the safeguards culture that is likely to continue throughout the next two decades and will apply the state-level approach where countries are evaluated as a whole, allowing for the concentration of inspection resources in a more focused manner. As such policies develop, they have to be flexible enough to adapt to changes in regional political structures, as well. For example, as national barriers disappear within the European Union, Integrated Safeguards and the state-level approach have to be modified accordingly. How this will evolve over time is very difficult to judge as it depends not only on the political development of the EU as an entity, but on its future expansion, as well.

Lastly, the development of new nuclear technology will impact safeguards and non-proliferation policy, as well. Fourth generation nuclear reactors strive to offer higher Safeguardability and inherent proliferation resistance and such advantages will certainly impact the safeguards regime that controls such installations. But nuclear developments are not restricted to such technologies that facilitate easier implementation of safeguards; as new technologies become available for sensitive activities such as separation and enrichment (e.g., laser technologies), non-proliferation policy and safeguards have to be adapted to cover all technologies and materials of concern.

Policy decisions always have an impact on how safeguards are implemented influencing the decisions about the set of tools that supports them. When looking at the possible developments in safeguards policy, it is worthwhile to investigate what features safeguards instrumentation should have and how it can best facilitate the actual implementation of efficient and effective safeguards.

3. Future Safeguards Instrumentation

The implementation of the Additional Protocol charges the IAEA with the detection of undeclared nuclear materials and activities. Foremost stands the expanded use of information technology based resources such as open source information analysis and satellite imagery. Furthermore, this has also a direct impact on the instrumentation that safeguards inspectors deploy during their inspection visits, especially sampling and in-field analysis tools.

For traditional safeguards, instrumentation is designed for applications such as verification of declared material including isotopic compositions, monitoring of specific operational activities (e.g., open core operations), and keeping materials and access points under seal. During Additional Protocol, or Complementary Access inspections, the nature of instrumentation that is required is fundamentally different. The inspector has limited or no knowledge about what to expect; therefore, the instrumentation required to support him/her must be portable and much more versatile than installed monitoring systems or even the portable traditional systems that are designed to verify declared materials.

Furthermore, the location where measurements are made or samples are taken during Complementary Access inspections is of critical importance for later analysis and cross-matching with other information sources such as satellite imagery, wide area monitoring, or open sources. This implies the need for better data management and location tagging capabilities, if possible. Installed, unattended instrumentation will undergo changes as well, as new technological approaches become available and the shift towards an information driven, qualitative assessment allows for the drawing of state-level conclusions about the absence of undeclared nuclear materials and activities in addition to the correctness and completeness of declarations.

The development of advanced and fourth generation reactor models has interesting consequences for safeguards instrumentation, as well. The implementation of safeguards measures during the design of such installations can alleviate the impact of treaty verification efforts on the operation of a nuclear installation today. To mitigate the need to pull cabling, retrofit the facility to provide the infrastructure for instrumentation, easier implement remote monitoring, and reduce on-site inspection time are all factors that will be appreciated by the operator.

In domestic safeguards systems, instrumentation does not necessarily have to operate for safeguards purposes only. There is a broad range of possible synergies, especially when the application of equipment is evaluated prior to the completion of the design of a facility. Surveillance cameras, for example, produce image data that IAEA inspectors use to draw conclusions about the correctness of declared and the absence of undeclared activities. Such image data are of interest to other concerns at a nuclear installation. First of all, it could be used to support physical protection measures as it might give an indication on insider or collusion threats. Next, it could strengthen personnel safety measures if image data analysis capabilities that can detect smoke or indication for other hazardous situations are added. Also, image data can provide a management tool if the operator can use image data to see if personnel are properly trained, rules are obeyed, and procedures (e.g., two-person rule) are followed.

In international safeguards there are concerns that the IAEA cannot share its data with the operator. However, new instrumentation could have the capability to generate different datasets specifically for each interested party that only contain the data necessary for their specific purpose. Such data would have to be independently authenticated to ensure their integrity. But if such requirements can be fulfilled, the same instrumentation could be utilized by multiple parties for various purposes.

With the shift of safeguards towards Integrated Safeguards and state-level conclusions, the question arises as to whether or not there will be a need for surveillance in future safeguards applications. Such discussions are mainly driven by the resources needed to operate a surveillance infrastructure not only for the equipment, but also for the image data analysis, field maintenance and support, and the frequency with which their data need to be extracted and reviewed. If multiple parties shared the benefits of surveillance, however, it could advance to be a feature implemented easily during the design with its cost shared among the users, thus becoming a true Safeguardability benefit.

Also, other instrumentation can be envisioned for synergies with new safeguards approaches. New measurement techniques that might have the potential to replace swipe sample taking and allow for immediate analysis could be added to the safeguards portfolio. Following the shift towards information-driven safeguards, such technologies can be envisioned in a portable form, as well. As an example, laser spectroscopy measurement techniques can be deployed to immediately detect and analyze the presence and enrichment of uranium-235 in a given uranium-hexafluoride (UF_6) sample. Such a technology could be used in portable applications to detect undeclared enrichment programs at undeclared sites or enrichment higher than declared at declared facilities. But it could also be employed in a fixed installation for continuous, on-line measurement.

If the measurement accuracy of such an approach is comparable or better than the currently used mass spectrometry, safeguards authorities will not be the only parties interested in it. Facility operators will have a similar if not larger interest in using the same technology for their quality assurance and cost-effectiveness qualities. Again, synergies between multiple users can be realized, the implementation facilitated during the design of the instrumentation, and the cost shared among the beneficiaries; thus truly offering Safeguardability attributes.

Overall, if instrumentation can support a multi customer approach and can be implemented early in the design of nuclear facilities, it can enhance safeguards as well as the State System of Accounting and Control of the host country. It also offers the opportunity to conduct safeguards related process monitoring in closer cooperation with the operator of a facility, thus realizing development and operational synergies.

4. Future Instrumentation Development Paths

When looking at the potential developments that might impact both safeguards and non-proliferation policy and the development of instrumentation, one should also ask the question whether the existing infrastructure to support research, development, and manufacture is appropriate for the emerging challenges. Currently, safeguards instrumentation is produced for a niche market with high reliability and tamper indicating requirements that have little applicability for other markets. If the usage of instrumentation is expanded towards more joint use and multi customer approaches, the market might expand accordingly.

The IAEA's support structure in place today outsources the development and production of safeguards instrumentation, often sponsored by Member States Support Programs (MSSPs) with IAEA experts developing the user requirements and overseeing project progress. If multi-customer scenarios with larger equipment quantities installed but also multiple party inputs to the user requirements emerge, this infrastructure might have to change accordingly. The question is whether more development effort should be spent by the IAEA itself as opposed to the external outsourcing approach that is used today. Should the IAEA conduct research and development in accordance with both internal and external requirements input and then identify suitable partners for commercialization? A different approach might be to shift more development responsibilities to the nuclear plant operator as the primary owner of the instrumentation while ensuring that IAEA requirements are 100% implemented. Either scenario would certainly affect the way the IAEA works with MSSPs.

Moving from a quantitative to a qualitative safeguards approach will also enhance the importance of identifying new technological developments and existing fields of technical solutions towards safeguards. Through its Novel Technology program the IAEA already identifies new and creative ideas to address new and existing challenges, and this area will see a further increase in activity as new, cooperative approaches emerge. A development path that involves a network of partners, those that provide new solutions and those experienced in safeguards to implement them into instrumentation and solutions ready for fielding, will be needed. Here, the IAEA has a strong foundation of research and development institutions and private industry to build upon.

5. Conclusions

The changing safeguards culture and the shift towards information-driven safeguards is a complicated concept that bears both challenges and opportunities. Synergetic instrumentation installed in future nuclear fuel cycle facilities that support both the operator and safeguards authorities can be envisioned to realize such opportunities while addressing the challenges. But its benefits need to be carefully balanced against implementation difficulty and cost; only if a benefit exists for both sides, treaty compliance verification authorities and operators, the implementation will be possible. If joint use, data sharing, and synergies can be realized while all security and data integrity concerns are addressed, the instrumentation will be a valuable addition for all parties involved.

Decisions on how to best proceed towards the new safeguards regime cannot be made by safeguards authorities alone. Rather, the early involvement of all participants to jointly decide on a course of action will promise the greatest chances of a rewarding result. This also needs to be a continuous process. As quantitative elements decline and qualitative elements increase, careful discussion of all stakeholders is needed to adapt existing agreements to changes in the non-proliferation regime and to the availability of new technologies. Also, what might be identified as an approach with high synergies between operators and safeguards for new nuclear reactors might not be applicable for existing facilities if cost and effort of retrofitting exceed the advantages of new instrumentation.

In support of new, proliferation-resistant fuel cycles and multi-national approaches, the goal should be to set a new standard for future nuclear safeguards while carefully measuring the interdependencies with other critical factors such as physical protection, environmental concerns, personnel safety, quality assurance, and economic sustainability. Only a balanced approach with input from all stakeholders can facilitate a swift and synergetic implementation.

References

[1] US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, *A Technology Roadmap for Generation IV Nuclear Energy Systems*, GIF002-00, December 2002. http://gif.inel.gov/roadmap/

[2] International Atomic Energy Agency, *Methodology for the Assessment of innovative nuclear reactors and fuel cycles*, IAEA-TECDOC-1434, IAEA, Vienna, 2002.

[3] US DOE, *Global Nuclear Energy Partnership Strategic Plan*, GNEP-167312, Rev. 0, Office of Nuclear Energy, Office of Fuel Cycle Management, January 2007.

Proliferation Resistance Characteristics of Advanced Nuclear Energy Systems: a Safeguardability Point of View

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Abstract

Among the international community there is a renewed interest in nuclear power systems as a major source for energy production in the near to mid future. This is mainly due to concerns connected with future availability of conventional energy resources, and with the environmental impact of fossil fuels. International initiatives have been set up like the Generation IV International Forum (GIF), the International Project on Innovative Nuclear Reactors and Fuel Cycles (IAEA-INPRO), and, partially, the US driven Global Nuclear Energy Partnership (GNEP), aimed at defining and evaluating the characteristics, in which future innovative nuclear energy systems (INS) will have to excel. Among the identified characteristics, Proliferation Resistance plays an important role for being able to widely deploy nuclear technology worldwide in a secure way.

Studies having the objective to assess Proliferation Resistance of nuclear fuel cycles have been carried out since the nineteen seventies, e.g., the International Nuclear Fuel Cycle Evaluation (INFCE) and the Non-proliferation Alternative Systems Assessment Program (NASAP) initiatives, and all agree in stating that absolute intrinsic proliferation resistance, although desirable, is not achievable in the foreseeable future. The above finding is still valid; as a consequence, every INS will have to comply with agreements related to the Non Proliferation Treaty (NPT) and will require safeguards measures, implemented through extrinsic measures. This consideration led to a renewed interest in the "Safeguardability" concept which can be seen as a bridge between intrinsic features and extrinsic features and measures.

Keywords: Proliferation Resistance, Safeguardability, Innovative Nuclear Energy Systems, Holistic Approach, Evaluation/Assessment.

1. Introduction

While not a completely new concept, safeguardability has often been connected to nuclear material characteristics. The GIF Proliferation Resistance & Physical Protection Working Group (GIF PR&PP WG) recognised that this is a reductive view of the subject, and preliminarily began to address it in a more comprehensive way. The PR&PP WG has more broadly defined safeguardability as the degree of ease with which a system can be effectively and efficiently put under International Safeguards.

Although not required for the PR&PP evaluation framework, the safeguardability concept has been recommended for use in the PR&PP approach because it can be of support to designers to consider safeguards needs beginning with the earliest phases of design. A preliminary list of attributes affecting the safeguardability of an Innovative Nuclear Energy System, including attributes that affect material accounting, containment and surveillance, and design information verification, has been identified in the GIF PR&PP methodology report, revision 3, and has been already reported [1, 2]. The current list of the safeguardability attributes is reported in Appendix D of the PR&PP methodology report, revision 5 [3, 4]. The Joint Research Centre (JRC) is among the promoters of safeguardability in GIF PR&PP EG (expert group), and, starting from the work that is being carried out within the group, is interested in addressing the safeguardability concept from a holistic point of view.

In section 2, the paper will recall how the issue of safeguardability had been approached in the past, and then it will give a summary of the safeguardability concept so far emerged within the PR&PP expert group activity (see section 3). In section 4, a possible way forward for the safeguardability concept in the GIF PR&PP frame is proposed. In a second part of the paper, and section 5 highlights a possible redefinition of some key concepts in a holistic approach to safeguardability. Finally, in section 6, some preliminary conclusions are presented.

2. Safeguardability in Literature

The issue of how to tackle the problem of safeguardability of nuclear energy systems has existed since the foundation of the International Atomic Energy Agency, and during the last thirty years at least two different approaches emerged:

- Developing new safeguarding techniques and equipments to enhance safeguards effectiveness and efficiency.
- Providing guidelines for designers of new systems, in order to enhance systems safeguard ability during early design stages.

In the following paragraphs these two different approaches will be briefly described.

2.1. Developing new safeguarding techniques and equipments to enhance safeguards effectiveness and efficiency

Although nuclear proliferation concerns have always accompanied the development of civilian nuclear technologies, the standard approach to non-proliferation was not to consider safeguardability as part of the design requirements of nuclear energy systems. Typically, a nuclear energy system was initially designed and licensed, and, in a following phase, the IAEA and the State in the process of building the system negotiated the safeguards approach for it. This implied that safeguarding techniques and equipments were conceived and/or adapted for the design of the nuclear energy system after the design was already fixed, with very limited capabilities of changing those design aspects that might create safeguardability concerns. The main R&D activities connected to safeguards were therefore those aimed at enhancing the effectiveness and efficiency of detection equipments and inspection planning (see e.g. [5,6]), and the few assessments in the field of safeguards had the objective of evaluating the effectiveness of a given safeguards approach applied to a given system (an overview of some of such studies can be found in [7]).

This approach has been favoured by the fact that up to the nineteen nineties systems did not change much from a safeguarding point of view, and, therefore, efforts were put in advancing in the technologies connected to detection equipments. Recently, with the start-up of international initiatives whose objective is to design the next generation of nuclear energy systems, the safeguarding community faces a new scenario, involving the need of new safeguards techniques able to cope with the major changes in the nuclear processes considered for new systems. Examples of discussions about the safeguardability of innovative fuel cycles in relation to the current safeguarding practice and capability can be found in [8] and [9], where the safeguardability of a pyro-processing facility [8] and of an advanced spent fuel conditioning process [9] are discussed, or in [10], where the safeguardability of an innovative closed fuel cycle is under examination. In [8], the Los Alamos National Laboratory carried out an analysis of the safeguardability of a pyro-processing facility under various assumptions of both design and safeguards approaches [11]. This exercise is particularly interesting, because it tackles the issue of safeguardability from a double point of view: not only by varying safeguards measures and techniques, but also by varying the original process design in order to make it more safeguardable.

In [9], the safeguardability of an advanced spent fuel conditioning process developed in the Republic of Korea is analysed in a study jointly carried out by the Los Alamos National Laboratory and the Korean Atomic Energy Research Institute (KAERI). After giving some hints of the innovative characteristics of the process leading to the absence of separated Pu (i.e., discussing material attractiveness, an aspect related to proliferation resistance), a hypothetical safeguards approach is assumed, in order to highlight whether IAEA detection goals are achievable or not.

In [10], European Commission JRC ITU analysed a hypothetical closed fuel cycle (double strata) from a double point of view: proliferation resistance and safeguardability. Proliferation resistance is mainly analysed from a material quality point of view, following the concept that proliferation resistance of a nuclear system may be increased by reducing the attractiveness of the involved nuclear material [12]. Safeguardability is then analysed by reasoning on the possibility to set up effective safeguards measures with current detection equipments and analytical techniques, and eventually hints are provided on the R&D activities to be carried out to cope with actual limitations. It is interesting to note, how the two aspects of proliferation resistance and of safeguardability are analysed separately and with different criteria. The relationship between these two aspects is briefly discussed in the next section of this paper.

These studies put in evidence how assessing safeguardability of systems whose designs are not yet finalised can open a double front: on the one side there is the possibility to understand how current safeguarding techniques and practices are suited for new concepts, and on the other side there is the possibility to propose design changes to enhance systems safeguardability.

2.2. Providing guidelines for designers of new systems in order to enhance systems Safeguardability during early design stages

Currently, national and international initiatives aimed at developing the next generation of nuclear energy systems have been launched. Among them, the two major international initiatives are the Generation IV International Forum [13] and the IAEA International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) [14] initiatives. Both efforts set up a number of goals that future systems will have to reach, and among them there is a call for increased proliferation resistance.

Proliferation resistance has been defined by [15] as:

"That characteristic of an NES that impedes the diversion or undeclared production of nuclear material or misuse of technology by the Host State seeking to acquire nuclear weapons or other nuclear explosive devices.

The degree of proliferation resistance results from a combination of, inter alia, technical design features, operational modalities, institutional arrangements and safeguards measures."

Proliferation resistance is, therefore, considered to be made of intrinsic features and institutional measures concurring in making the nuclear energy system unattractive for its use in a military programme. Innovative nuclear energy systems will, therefore, still rely on international nuclear safeguards, and this should be considered a design requirement. It has to be noted that, even if the INPRO initiative does not talk openly about safeguardability, this concept is de facto present behind indicators they identified as relevant to detectability [16].

Although no one is entitled to impose proper design requirements related to systems safeguardability, systems designers repeatedly asked the PR&PP WG for guidelines able to help them to take this aspect into account in their activity. Scanning past literature for studies providing guidelines for designing more safeguardable nuclear energy systems highlights that this topic has been investigated since the late nineteen seventies, but relatively few studies have been carried out, mainly in the US [17] and by IAEA [18, 19]. Their main focus is on nuclear reactors, and they provide a good basis for preparing guidelines for designing future systems.

3. Safeguardability within the GIF PR&PP Evaluation Methodology

Although not required for the PR&PP evaluation framework, the safeguardability concept has been recommended for use in the PR&PP WG, because it can be of support to designers to consider safeguards needs beginning with the earliest phases of design, and is currently described in Appendix D [4] of the PR&PP Evaluation Methodology report, revision 5 [3].

3.1 Safeguardability as in GIF PR&PP Evaluation Methodology report, Rev. 5, App. D

In [4], safeguardability is defined as the degree of ease with which a system can be effectively and efficiently put under international safeguards. Its analysis is strongly influenced by a number of aspects connected with a system's design, such as system's layout, the type of process chosen and its actual implementation, the foreseen operating profile. In addition, another relevant source of influence is the international non-proliferation legal framework under which the system will be operated.

From its definition, safeguardability will be related to the potential easiness of setting up a safeguards approach which would effectively and efficiently provide credible assurance that no undeclared illicit activities have been carried out in the nuclear energy system. This implies that a safeguardability analysis is based on the knowledge of the nuclear energy system, of the legal framework in which the nuclear energy system is operated and on the verification activities and techniques at disposal within the considered legal framework. Any change not only to the first aspect, but also to the other two will lead to a different result of the safeguardability analysis, making it evolutionary in nature and context dependent.

Two possible objectives were considered for safeguardability:

The first one was to have a possible substitute for two pathway measures - Detection Probability and Detection Resources Efficiency - where not enough information for their estimation is available. Typically, this is the case of very early design stages.

The second one was to answer to a system designers' request to have some kind of guidance for designing systems able to ease activities connected with the implementation of international safeguards.

For analysing safeguardability, a set of attributes have been identified by the Expert Group, grouped in three broad categories: 1) attributes capturing the potential ease of performing Design Information Verification (DIV), 2) attributes capturing the potential ease of performing Nuclear Material Accounting (NMA) and, finally, 3) attributes capturing the potential ease of implementing Containment and Surveillance (C/S). For each category, a table has been prepared, containing the relevant attributes and a short description of their meaning.

Attributes have been identified as the system's intrinsic features upon which current safeguards techniques rely, independently of the fact that such technique will be actually implemented. In analysing safeguardability, no assumption of a particular safeguards approach is made. Reference [4] states that, *whereas safeguardability is an index of the potential ease of implementing an effective and efficient safeguards approach based on current practice, detection probability is an index of the effectiveness of an implemented safeguards approach.*

The current tables are published in [4] and represent a first step towards the identification of the systems' design attributes that may affect safeguardability. The tables reported there, have been developed considering two different scenarios:

In the first scenario, a team of experts is to perform a safeguardability assessment of the system's design, in order to provide relevant feedback to the system's design team. Designers would, hence, have the possibility to improve the design according to the recommendations resulting from the assessment. This close and iterated interaction between designers and safeguardability experts would eventually lead to a highly safeguardable nuclear energy system.

In the second scenario, the safeguardability assessment is performed for supporting policy makers in decisions where safeguardability is an important characteristic.

This double approach led to a set of attributes that are general in nature, partly characterised having a feedback to designers in mind, and partly characterised having a feedback to policy makers.

In the following paragraph, the first scenario has been selected for illustrating a possible way forward for advancing in the work.

3.2. Advancing in the work: an improved characterisation of the identified attributes

The first step for progressing is to validate the current tables and investigating their completeness. These tables have undergone a validation exercise within the PR&PP WG. Since they are not to be considered as finalised but only a first milestone in a work in progress, the tables are currently undergoing an external validation process via interviewing relevant domain experts not directly involved in the GIF PR&PP activities. Both the internal and the ongoing "external" review process put in evidence that the first step in the scheduled advancement in the work is to better characterise the identified attributes. Ideally, each attribute could be characterised according to six aspects, identified by the six keywords What, Why, Who, How, When, and Where (this paradigm is inspired by and adapted from [20]). For each attribute, the objective is to fill Table 1, adapted from [21], where the What aspect captures

Name				
What				
Why	*Rationale:* *Example(s):*			
Who	*Relevant to:* *Assessed by*:			
How	*Performance Indicator*	*Description*	*Scale*	*Comment*
When				
Where				

Table 1: Attributes Characterisation table in terms of What, Why, Who, How, When/Where. Adapted from [20].

the objective of the attribute, the Why captures the rationale, and eventually some examples where this attribute is fulfilled and not fulfilled, the Who captures all the players for whom the attribute is relevant and the experts needed for assessing it, the How captures the performance indicators that could be used for assessing the attribute and its related description and scale, the When and Where are grouped for capturing the stage at which the attribute comes into the game.

In this paper an attempt to characterise the safeguardability attributes identified in [4] in terms of objective (What) and rationale (Why) is presented in Tables 2, 3, and 4.

Name	Objective (What) and Rationale (Why)
Comprehensiveness of facility documentation and data.	**Objective:** Making sure that the facility documentation is exact and complete in all the aspects relevant to design verification activities. **Rationale:** Every facility that has to be put under international safeguards will have to be described in a documentation set requested by the safeguards inspectorate. For Design Information Examination (DIE) and DIV, a Design Information Questionnaire (DIQ) has to be compiled. Exact and complete documentation about the facility, with detailed layout in both hardcopy and electronic form would greatly facilitate a comprehensive compilation of DIQ, and this will in turn provide benefit to the inspectorate for the foreseen activities.
Transparency of layout	**Objective:** To make sure that the system layout is conceived in such a way that process lines are easily identifiable and could be checked for consistency with the declarations at different design stages. **Rationale**: An important aspect of Design Information Verification (DIV) is verification of the process equipments and layout. Often process equipments layout is not conceived for easy layout verification, and is therefore difficult to check. Although this might not be an issue in item facilities, it might create big difficulties when bulk facilities are involved, especially when the process is continuous.
Possibility to use computerised reconstruction models	**Objective:** To make sure that the system is conceived in such a way that technologies aimed at making the DIV activity easier could be used. **Rationale**: Modern techniques such as the ones based on 3D laser rangefinders allow performing a 3D mapping of the area to be verified, thus, allowing computer assisted verification of the equipments and facility layout.
Possibility to have visual/ instrumented access to facility equipments while operational.	**Objective:** To make sure that the system is conceived in such a way that every relevant process equipment can be visually or instrumentally checked for DIV purposes during normal operation of the facility. **Rationale**: DIV is normally performed during planned shut downs of facilities, but this is not always possible. In addition, on some facilities, accessibility to all relevant equipments is not possible even during planned shut downs, e.g., due to radiological hazards. Compliance with this attribute would ease the work of the inspectors and avoid any loss of time/resources to the operator.

Table 2: Nuclear System attributes faciliting design Inventory Verification (DIV).

Name	Objective (What) and Rationale (Why)
Uniqueness of material signature	**Objective:** Making sure that the nuclear material available in the facility has intrinsic characteristics that contribute to easily recognise it in terms of type and composition. **Rationale:** Having unambiguous material signature positively affects measures aimed at discriminating the nuclear material composition. Moreover, it makes any concealment activity aimed at substituting the declared nuclear material with dummies more difficult. In origin inspired by [22]
Hardness of material signature.	**Objective:** To make sure that the system layout is conceived in such a way that process lines are easily Objective: Making sure that the nuclear material available in the facility has a radiation signature which is easy to be measured. **Rationale:** During PIV, inspectors typically perform attribute verification measurements on a sample of the available nuclear material. Having hard nuclear material signature would facilitate the inspectorate to detect and record it and, therefore, conclude that the measured material is compliant with the operator's declarations. In origin inspired by [22].

Possibility of applying passive measurement methods	**Objective:** Making sure that the nuclear material available in the facility can be characterized via passive measurement methods instead of requiring active measurement methods. **Rationale:** During PIV inspectors typically perform attribute verification measurements on a sample of the available nuclear material. If the nuclear material characteristics allow passive NDA techniques to be used to reach the intended objective, the inspectorate would have the opportunity to get the job done in an easier and cheaper way. In origin inspired by [22]
Item/bulk	**Objective:** Making sure that the nuclear material form inside the system is compatible with accurate and efficient NMA. **Rationale:** In terms of nuclear material accounting, item facilities are easier and cheaper to verify than bulk facilities, since the material balance is easier to be closed (in principle no MUF is expected) and verifications typically rely on NDA measurements rather than DA measurements. In principle, NMA in item facilities is less resources intensive than in bulk facilities. In origin inspired by [22]
Uncertainties of detection equipments	**Objective:** Making sure that the system is designed in such a way that the nuclear material treated can be verified and inventoried with current verification techniques and equipments. **Rationale:** New systems might take advantage of new processes and possibly innovative nuclear material forms. Designing the system keeping in mind that those processes will have to cope with NMA, carried out using specific verification techniques, with known performance target values, means to facilitate the safeguards designers' work and reduce the safeguards resources needed to be allocated. In origin inspired by [22]
Annual throughput	**Objective:** Making sure that the amount of nuclear material produced by the system's processes is compatible with the international safeguards inspection goals. **Rationale:** Large bulk facilities handling big quantities of nuclear material per year might generate throughputs challenging the possibility to reach the safeguards detection goals. Annual throughput might affect the accuracy of physical inventories in absolute terms, and, therefore, detection limits.
Batch/ continuous process	**Objective:** Making sure that the type of material handling, within the process chosen for the facility, is compatible with an accurate closing of a nuclear material balance. **Rationale:** How nuclear material is handled in a bulk facility might have a strong impact on the overall difficulty of closing the material balance. In principle, batch processes, possibly with constant nuclear material composition between batches, could lead to easier and more accurate closing of material balances when compared with continuous processes.
Radiation Field	**Objective:** Making sure that the system is designed in such a way that the nuclear material radiation field does not affect the inspection activities. **Rationale:** The presence of a radiation field generated by the system's processes and involved nuclear material is unavoidable. A safeguardable facility should be designed in such a way that the inspectorate's activities are not jeopardised by radiological hazards. This affects both the inspection activities and the eventual servicing of fixed equipment (accessibility etc.).
Amount of hidden (unverifiable) inventory	**Objective:** Making sure that the system's design is optimised for minimizing the amount of nuclear material that might not be accessible to inspectors during safeguarding activities. **Rationale:** Each process has a physical amount of nuclear material that is not accessible during inspection activities, because, e.g., the material may be inside process pipelines. A system optimised to minimise this amount of material would facilitate the closing of the material balance by the inspectorate.
Possibility to implement near real time accountancy	**Objective:** Making sure that the system's design is compatible with the implementation of near real-time accounting techniques. **Rationale:** For some processes the closure of material inventory for timeliness purposes is particularly challenging. Having the possibility to implement near real time accounting would greatly help the inspectorate in closing the material balance frequently and without interrupting the process, in order to achieve timeliness objectives while not intruding in the system's operation.

Table 3: Nuclear System Attributes facilitating Nuclear Material Accounting (NMA).

Name	Objective (What) and Rationale (Why)
Operational practice	**Objective:** Making sure that the operational profile and procedures of systems facilitate the applicability of C/S measures. **Rationale:** The aim of C/S measures is to maintain continuity of knowledge on the systems' nuclear material inventory between two inventory verifications. The way in which the system is operated might increase the ease of applying containment and surveillance measures.
Extent of automation and remote handling	**Objective:** Making sure that procedures are carried out with a low need of human intervention and that all operator's equipments can be instrumented by the inspectorate. **Rationale:** Having highly automated processes would facilitate the application of C/S measures in several ways: collection of data coming from operation equipments can be used for continuity of knowledge purposes, and the possibility to have few personnel in the operations area would facilitate the review of surveillance cameras recordings.
Standardisation of items in transfer	**Objective:** Making sure that items in transfer inside the system are as standardised as possible. Rationale: Having standardised items (e.g., flasks) in transfer would facilitate the application of C/S measures in several ways. Examples are: easier interpretation of recorded images in the review phase, easier surveillance of standard items in transit as they would probably result in moving at the same speed.
Possibility to apply optical surveillance	**Objective:** Making sure that the system's design is optimised for the use of optical surveillance devices. **Rationale:** In current practice, surveillance relies heavily on the use of images recorded by surveillance cameras. In order to maximise the benefits that this technique embeds, the system can be designed keeping this in mind.
Number of possible transfer routes for items in transit	**Objective:** Making sure that the system's design foresees a limited amount of possible transfer routes for the nuclear material in transit. **Rationale:** Having only one or very limited possible transfer routes for items in transit would greatly improve easiness of performing surveillance and interpreting surveillance records during the review phase. This attribute is also particularly important for facilitating the application of containment measures, given that operational rules allow their use (e.g., no transfers for long periods of time).
Possibility to apply remote surveillance	**Objective:** Making sure that the system's design allows the possibility to transfer C/S data offsite. **Rationale:** Remote surveillance helps to achieve timeliness and saves onsite inspection efforts, concurring in lowering the resources needed by a safeguards approach.

Table 4: Nuclear System Attributes facilitating application of Containment and Surveillance (C/S) and other monitoring systems

These tables build on the official ones in [4], and, although a brainstorming for identifying new missing attributes to be added to the current ones is in course, this effort has not been included here. As a consequence, no additional attributes are introduced here, and the focus is uniquely on advancing with the available material.

A safeguardability analysis intended at providing feedback to systems designer teams should be capable of analysing the systems design at the earliest possible design stages. Clearly, not all the attributes identified in Tables 2-4 can be relevant at very early design stages, some of them might require a fairly detailed description of the system. Since every attribute is considered to be important for a sound safeguardability analysis, this aspect calls for a method able to deal with incompleteness of information.

4. The way forward within the PR&PP WG framework

In addition to characterising the current attributes by filling in Table 1 for each of them, the ongoing review of the current safeguardability tables put in evidence a number of aspects that are going to be deepened and implemented in the next steps. Among the aspects that will be deepened in the near future are the following:

- The current tables are mainly focused on the activities to be carried out during routine inspections. It is foreseen to give more coverage to those aspects that are relevant for providing the safeguards inspectorate with the information needed for designing the system's safeguards approach, e.g., the information needed for compiling the system's Design Inventory Questionnaire (DIQ) and the facility attachments.

- No particular attention has been paid yet to the aspects influencing the easiness of collecting the information needed by the operator for the reporting activities foreseen by the inspectorate. It is desirable to take explicitly into account those aspects facilitating the setting up of an accurate and efficient accounting and reporting system by the operator.
- Aspects critical for the safeguardability of a nuclear energy system but not considered yet are those that influence the easiness of, e.g., recovering from interruption of continuity of knowledge, or of performing typical follow-up actions. It is foreseen to enhance the coverage of these aspects.
- During the preliminary investigation of the safeguardability concept, the PR&PP WG deliberately focused only on the traditional safeguards measures and activities. It is foreseen to begin the investigation of the aspects related to the measures introduced by the Additional Protocol and the subsequent Strengthened Safeguards regime.

Once the attributes are finalised, there is the need of means to capture the evidence for and/or against their fulfilment, in order to be able to draw a conclusion on every single attribute and, eventually, on the basis of the conclusions on the single attributes, to be able to draw a conclusion on the overall safeguardability of the system under investigation. Any candidate technique for this task will have as a minimum requirement to be able to cope with conflicting and incomplete evidence, and to be able to capture the involved uncertainty in all its forms, i.e., fuzziness, incompleteness, and randomness.

5. Some reflections on the Safeguardability concept: setting up the case for a holistic approach

5.1. Safeguardability analysis on what? Setting the scale

Since the safeguardability concept is aimed at providing feedback to system designers, it is worthwhile to spend some time investigating the meaning of nuclear energy system. The actual definition of nuclear energy system within the GIF PR&PP evaluation methodology is the following:

A Generation IV Nuclear Power Producing Plant and the facilities necessary to implement its related fuel cycle.[1]

Actually, this definition implies that the nuclear energy system includes the whole nuclear fuel cycle involving an innovative nuclear power reactor. It is very unlikely that a single design team will address the whole nuclear fuel cycle, and, therefore, it would be reasonable to allow the safeguardability analysis to be performed also on a single facility or even to a single process of a particular facility. In principle, the scope of the analysis will define, if the analysis will be performed on a process line, on a facility or on a complete fuel cycle. If this is accepted, it would be possible to notice how the safeguardability concept could be defined in a holistic way, and suited to all needed levels.

The term *holistic* is connected with the concept of *holon*: a holon is at the same time a part and a whole. For example, the human being is a whole, for it is made of different sub-systems such as the skeleton, the cardiovascular system, and the pulmonary system. At the same time, it is a part, since it is part of a social structure (family, city, nation, etc.)[2]. Whether a human being is considered to be a part or a whole is a matter of the scope of the analysis and of the chosen level of detail. The same consideration applies to our problem: depending on the scope, a facility can be considered as a part of a system or of a fuel cycle or as a whole for the processes carried out in it. In principle, a safeguardability analysis could be performed regardless of the chosen holon. It has to be noticed that passing from, e.g., a facility level to a fuel cycle level, the safeguardability of the latter generally emerges from the integrated behaviour of the former ones, i.e., the safeguardability of the fuel cycle might be different from a simple aggregation of the safeguardability of its individual facilities.

5.2. Safeguardability: a concept connected to intrinsic features or to extrinsic measures?

Both in literature and within the PR&PP WG the issue of characterising safeguardability as an intrinsic or an extrinsic characteristic of a nuclear energy system emerged. In the proliferation resistance do-

1 [3], p.66.
2 For a more detailed description of the holon concept in technical areas, see [20].

main these two concepts (intrinsic and extrinsic) were defined in [15]. Intrinsic proliferation resistance features are those features that result from the technical design of nuclear energy systems, including those that facilitate the implementation of extrinsic measures[3], and extrinsic proliferation resistance measures are those measures that result from States' decisions and undertakings related to nuclear energy systems[4].

When reasoning on the safeguardability concept, one possibility might be to define it as the collection of those features resulting from the technical design of the system, that facilitate the implementation of extrinsic proliferation resistance measures. This definition would classify safeguardability as an intrinsic proliferation resistance feature.

Although the above point of view might well be adopted, it is worthwhile to take a broader and holistic point of view, and (following [20]) begin to acknowledge that any technical (hard) system is embedded (integrated) in a human, social (soft) system in which it is operated and with which it interacts in numerous and often complex ways. Indeed, failures leading to catastrophic accidents often occur due to failures or misunderstandings or understatements of these interactions between hard and soft systems. In our case, the nuclear energy system is deployed and operated by an operator, and subject to international non-proliferation agreements leading to nuclear safeguards verification activities. Once this scenario is accepted, safeguardability could be defined as a property emerging from the interaction of a nuclear energy system (hard) with the activities connected with the non-proliferation legal framework in force (soft). In particular, safeguardability could be seen as an index of the potential quality of this interaction.

5.3. Safeguardability and Proliferation Resistance: a tight and complex relationship

The relationship between the safeguardability concept and proliferation resistance is certainly a tight one, but its characterisation strongly depends on how safeguardability and proliferation resistance are defined.

If the definition of proliferation resistance given in [15] is accepted, and the first possibility of defining safeguardability given in 4.1 is assumed, then safeguardability would be seen as a subset of the PR intrinsic features.

If we adopt a holistic point of view, and accept that hard systems are integrated in soft systems with which they interact, proliferation resistance and safeguardability could be seen as indexes of the potential quality of different interactions of the hard system with the soft system: proliferation resistance could be seen as measure of the potential quality of the interaction of a nuclear energy system with activities connected with a proliferation effort, and safeguardability, as stated in the previous paragraph, could be seen as a measure of the potential quality of the interaction of the nuclear energy system with the activities connected with the non-proliferation legal framework in force.

It is worth noticing that some of the attributes identified for analysing safeguardability are relevant also for analysing proliferation resistance, but their contribution might be a positive one in one case and a negative one in the other. For example, limited accessibility to nuclear material due to the radiological hazards connected to the radiation field is negative for safeguardability (inspectors' activities are negatively affected), but very positive for proliferation resistance (a good radiological barrier increases the technical difficulty associated with a diversion scenario). This aspect puts in evidence that designing a nuclear energy system excelling in ensuring non-proliferation is a challenging task, where trade-offs on a number of important aspects will have to be achieved, and optimisation of these trade-offs will not be always straightforward.

6. Conclusions

Due to various reasons, there is a renewed interest in nuclear energy as an important player in the near to mid future. This led to various international efforts aimed at shaping and designing future nuclear power plants and their related fuel cycles. Among other goals, future nuclear energy systems will have to be proliferation resistant and will have to operate under an international nuclear safeguards regime.

Designers of nuclear energy systems repeatedly asked the GIF PR&PP WG to provide guidelines to ensure that their teams take the issue of safeguardability into account at early design stages, and the group is developing this concept for answering to this need. As a first step of the work, a list of relevant attributes has been developed and published in the latest revisions of the GIF PR&PP Evaluation Methodology Report.

3 [15], p.1.
4 15], p.2.

Work on the subject is still ongoing and currently an external validation process and a further characterisation of the identified attributes is being performed at JRC, taking advantage of JRC experts in the relevant domains. The proposed way forward for this activity has been presented in section 4.

While continuing to contribute to the main stream of the PR&PP WG activities, JRC is exploring the possibility to couple the experience on the concepts of proliferation resistance and safeguardability gained in the GIF PR&PP frame with the internal know-how on safeguards and non-proliferation to try to develop a holistic approach for analysing the evidence that a nuclear energy system design will be highly safeguardable. This activity is being carried out adopting a holistic systems thinking approach developed at the University of Bristol (UK).

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References

[1] PR&PP expert group and other contributors, Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems, *Proceedings ESReDA 29th Seminar, Systems Analysis for a More Secure World. Application of System Analysis and RAMS to Security of Complex Systems*. October 25-26, 2005, JRC, IPSC, Ispra.

[2] Bari, R., Nishimura, R., Peterson, P., Roglans, J., Bjornard, T., Cazalet, J., Cojazzi, G.G.M.*; Delaune, P., Golay, M., Haas, E., Rochau, G., Renda, G., Senzaki, M., Therios, I., Zentner, M., Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems: an Overview. *Eight International Conference on Probabilistic Safety Assessment and Management* (PSAM 8),New Orleans, USA, May 14-19, 2006.

[3] *PR&PP Expert Group,* E*valuation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems*, Revision 5, November 30, 2006 GIF/PRPPWG/2006/005. Available http://www.gen-4.org/Technology/horizontal/PRPPEM.pdf

[4] *PR&PP Expert Group, Addendum to the Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems*, Technical Addendum to Revision 5, January 31, 2007 GIF/PRPPWG/2006/005-A. To be made available at http://www.gen-4.org/Technology/horizontal/

[5] Mayer, K., Poucet, A., Schenkel, R. and Wellum, R. Focused research contributing to more effective and efficient safeguards, *ESARDA 24th annual meeting, Workshop on R&D responses to the new safeguards environment*, Luxembourg, 28-30 May, 2002.

[6] Canty, M., and Avenhaus, R. Timely Inspection and Deterrence, *Proceedings of the 21st ESARDA annual meeting, symposium on Safeguards and nuclear material management*, Sevilla, 4-6 May,1999.

[7] Cojazzi, G.G.M., and Renda, G. *Safeguards assessment methodologies: Review and perspectives from a systems analysis point of view*. EUR 21307 EN, September 2004.

[8] Budlong Sylvester, K.W., and Pilat, J.F. Safeguards and advanced nuclear energy systems: enhancing Safeguardability, *ESARDA/INMM joint workshop on Safeguards perspectives for a future nuclear environment*, Como, Italy, October 14-16, 2003.

[9] Li, T.K., Lee, S.Y., Burr, T., Thomas, K., Russo, P., Menlove, H., Kim, H.D., Ko, W.I., Park, S.W., and Park, H.S. *Safeguardability of advanced spent fuel conditioning process, 45th INMM Annual Meeting*, Orlando, USA, July 18-22, 2004.

[10] Mayer, K., and Schenkel, R. Safeguardability of advanced fuel cycles. In 24th *ESARDA Annual Meeting, Workshop on R&D responses to the new safeguards environment*, Luxembourg. EC DG-JRC, May, 28-30 2002.

[11] Budlong Sylvester, K., and Eller, G.P. *Safeguards Evaluation for a Proposed Pyroprocessing Facility*, LA-UR-05-4364, June 28, 2003.

[12] Bragin, V., Carlson, L., Leslie, R., Schenkel, R., Magill, J. and Mayer, K. *IAEA Conference on International Safguards*, 29 October to 2 November 2001, Vienna, Austria.

[13] US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum. *A technology roadmap for generation IV nuclear energy systems*. GIF002-00, December 2002. http://gif.inel.gov/roadmap/

[14] International Atomic Energy Agency. *Methodology for the assessment of innovative nuclear reactors and fuel cycles*. IAEA-TECDOC-1434, 2004.

[15] International Atomic Energy Agency. *Proliferation resistance fundamentals for future nuclear energy systems*. IAEA-STR-332, December 2002.

[16] International Atomic Energy Agency. *Guidance for the Application of an Assessment Methodology for Innovative Nuclear Energy Systems*, INPRO Manual — *Proliferation Resistance Volume 5 of the Final Report of Phase 1 of the International Project on Innovative Nuclear Reactors and Fuel Cycles* (INPRO), IAEA-TECDOC-1575, October 2007.

[17 Goldman, L., Pasternak, T., and Swartz, J. *Design features of nuclear reactors which would make international safeguards more efficient*. Report ISPO-123, Science Applications International, 1980.

[18] International Atomic Energy Agency. Nuclear facility design: *guidelines assisting the implementation of IAEA safeguards*. Safeguards technical report series n.144, 1983.

[19] International Atomic Energy Agency. *Design measures to facilitate implementation of safeguards at future water cooled nuclear power plants*. Technical Report 392, 1999.

[20] Blockley, D., and Godfrey, P. Doing it Differently: *Systems for Rethinking Construction*. Thomas Telford Ltd, 2000.

[21] van Wijk, L.G.A., *A Process Model for Nuclear Safeguarding*, PhD thesis, University of Bristol, Bristol, United Kingdom, 2005.

[22] *Nuclear Energy Research Advisory Committee. Annex: Attributes of Proliferation Resistance for Civilian Nuclear Power Systems*, October 2000.

Methodologies for Proliferation Resistance Evaluation

A Practical Tool to Assess the Proliferation Resistance of Nuclear Systems : the SAPRA Methodology

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Abstract

This article is aiming at presenting a simple approach which may be easily implemented to assess and make use of provisions for enhancing Proliferation Resistance (PR) of nuclear systems (that is nuclear reactors and their associated fuel cycles). In the initiatives to develop innovative nuclear energy systems (Generation 4, INPRO), PR is one of the key elements, along with economics, safety, sustainability and environment which has to be addressed. Assessment of proliferation resistance is therefore of timely importance.

This method is called "SAPRA" (which stands for "Simplified Approach for Proliferation Resistance Assessment") and it uses the classical concept of barriers. Four categories of barriers are distinguished: material, technical, institutional and specific barriers for the weapon making phase. In the proliferation process (or route), four steps are considered: diversion of materials, transformation, transport and making of the nuclear weapon using either high-enriched uranium or plutonium. All steps of the fuel cycle from uranium mining to final disposal of spent fuel or nuclear waste are examined. A scale of value is then defined in order to quantify each of these elementary steps and figures are aggregated to obtain global performance indices for PR and to identify weak points. Various nuclear systems have been analyzed and general conclusions drawn from these results are presented.

This paper offers a first developed framework to derive a practical and effective use of a proven tool to the needs and specificities of proliferation resistance assessment.

1. Introduction

Proliferation Resistance (PR) has become one of the primary topics to be addressed in the frame of the development of nuclear energy systems. While PR analysis and assessment have already been implemented for a long time, the need arises to structure the methodology and to develop simple approaches which can be directly used for assessing PR of various nuclear fuel cycle options. The method presented here, called SAPRA, is based on a quantitative assessment of the effectiveness of technical and institutional barriers which may impede a proliferation attempt.

2. The Framework of the Analysis

First of all, when speaking of proliferation, we must define precisely what is exactly the threat we are dealing with. To this end, we start from basic definitions derived from the US "NPAM" project [1] and a recent IAEA-sponsored work [2] :

- <u>Proliferation:</u> Acquisition of one or more nuclear weapons by a nation that does not have them.
- <u>Proliferation Resistance:</u> All of those characteristics of a nuclear energy system that impede the diversion or undeclared production of nuclear material, or misuse of technology, **by States** in order to acquire nuclear weapons or other nuclear explosive devices.
- <u>Proliferation Risk:</u> The likelihood of a nation acquiring one or more nuclear weapons within a given period of time.

Within the frame of this terminology, it appears that "Proliferation Risk" is a combination of PR as defined above and of proliferator characteristics, which measures the willingness and capacity of a given state to acquire a nuclear weapon (these elements are related to political issues as well as to economical and technical capabilities of the proliferators). Furthermore, the analysis presented here is limited to PR only, that is to say that other threats such as sabotage or fabrication of "dirty bombs" are not "proliferation" as defined above and are, therefore, not addressed in this paper.

3. A Pragmatic Approach for Assessing Proliferation Resistance

3.1. The General Framework

The proliferation concern has been particularly addressed for a long time in France through its international commitments and actions as well as its own provisions aimed at preventing proliferation of nuclear weapons. In this context, French nuclear institutional and industrial actors, in particular **AREVA**, have established a working group on "Proliferation Resistance and Physical Protection" (PRPP). In the terms of reference of this group, it is stated that one of the mandates of the group is to select and develop a methodology for assessing proliferation resistance of nuclear systems. A subgroup of experts was created for that purpose and a report was issued in 2006 on this work [4]. This paper is aimed at providing a short description of this work.

3.2. The Methodology

From the very definition of proliferation resistance (PR) given in section II., we have considered that a pragmatic approach for assessing PR must take into account two types of factors :

1 – Material-related, technical or organizational and institutional factors, for a given nuclear system, that may impede the diversion of sensitive nuclear material and / or the misuse of sensitive technologies. These are the "barriers" (or "lines of defence") of the well known "defence in depth" fundamental principle considered in nuclear safety analysis.

2 – "Country profile", related to the international commitments in the nuclear field and the status of the nuclear activities of a given country. nuclear activities of a given country.

On this basis, we have implemented an approach which is similar to the one initially proposed by the "TOPS" special task force [3], initiated by the US-DOE in 1999, and in which French representatives participated actively. As a matter of fact, we have reviewed several other methodologies developed for assessing PR, and we judged that the TOPS approach is among the most practicable ones, at least at the present time (while awaiting the availability of the more sophisticated methodology which is being developed in the frame of the Gen-IV initiative, by the "PR&PP" working group).

The main difference between our approach and the TOPS approach is that we have explicitly distinguished between four phases in the proliferation process, in order to better follow potential proliferation routes and assess the efficiency of the different barriers at each step of these routes.

Following this approach, we have considered a **matrix** made up of two types of parameters:

1 – A set of all fuel cycle steps associated with the nuclear system (one step is one line of the matrix);

2 – A set of various phases in the proliferation process, and, for each of these phases, a set of relevant "barriers" (one stage of the process with a given barrier is one column of the matrix).

Then, the analysis consisted in grading each box of the matrix with the help of a number which was allocated by an expert group to assess the efficiency of each barrier with regard to the proliferation route envisaged. In the final step of the methodology, various aggregations of these numbers were carried out, in order to obtain global figures liable to characterize the degree of resistance to proliferation of a system for a given type of proliferator.

3.3. Hypotheses and Data Structure for the Assessment

3.3.1. Material-related, Technical and Institutional Barriers

A comprehensive list of material properties, technical characteristics and institutional measures likely to constitute an obstacle to a proliferation route was drawn up. This list is similar to the one worked out by the TOPS study, with only a few modifications or additions. For example, we considered the physical form of a material as a potential barrier (which is not considered in TOPS), because it may make diversion or a transformation process more or less difficult. Another example that we have taken into account is the fact that a clandestine facility may be detected by its impact on environment. It is a technical barrier that we call "signature". With this respect, it must be noted that a reprocessing plant (aimed at separating plutonium) is most likely more detectable than an enrichment plant (aimed at producing high-enriched uranium). With regard to institutional barriers, we did not consider "location" as a barrier because, as it is pointed out in the final TOPS report, its effectiveness would require careful evaluation of the threat and location implications to determine the "net" value of this barrier. On the other hand, the distance to cover for transportation of a nuclear material quantity was considered as a barrier in our method.

3.3.2. The Different Phases of the Proliferation Process

As already mentioned, we have explicitly distinguished between four phases in the proliferation process which are the following:

1. Diversion, that is covert theft of nuclear material (fissile material or spent fuel) from a facility which is supposed to be under international safeguards and which is supposed to have physical protection measures (in this case, a certain level of internal or external “collusion” or “complicity” is generally necessary)

2. Transport of the nuclear material from the facility to the location where it must be further processed

3. Transformation, of the nuclear material in order to make “ready-to-use” weapon grade material

4. Manufacturing of a nuclear weapon, once the weapon grade material (“ready-to-use”) is available.

Of course, all barriers identified in the first step are not necessarily relevant for each of these four phases, and only the one which may have an effect on a given phase has been considered in each of these phases. This is very clear, for example, for the ultimate phase of nuclear weapon production, where institutional barriers no longer exist by definition.

3.3.3. Country Profile

Actually, in such an analysis, we must take into account what we call the “country profile”, because the proliferation risk clearly depends on the institutional situation and technical capabilities peculiar to each country. Incidentally, we underline the fact that, according to the very definition of PR, we did consider **only** the case of proliferation by a **state**, and **not** by a sub-national group. From this point of view, the following elements were considered:

- Status of the country concerning nuclear weapon possession as well as position of the country with regard to the international non-proliferation regime (NPT, Additional Protocol or any other international agreements in the nuclear domain)
- level of development (high, average, low)
- type of reactor(s) operated (if any) : LWR, CANDU, HTR
- fuel cycle options (open or closed cycle) and associated domestic facilities.

Of course, one cannot carry out a specific study for all possible combinations of these factors and, up to now, we have studied only some of the most representative cases according to the present status and near prospects of nuclear activities in the world.

3.3.4. Fuel Cycle Steps

In a classical manner, we have considered the three major stages of the nuclear fuel cycle and for each of them, only their main steps (if relevant), that is:

- Front-end: mines, conversion, enrichment, fuel fabrication, and transportation of nuclear materials and fuel assemblies between each step (specific steps are added for the CANDU-DUPIC fuel cycle as well as for MOX fuel fabrication).
- Reactor operation: fresh fuel storage and handling, irradiation in the reactor core, spent fuel handling and storage at the reactor.
- Back-end: spent fuel transportation and storage, handling, reprocessing, plutonium storage, spent MOX fuel storage and, for final disposal, handling, conditioning, and emplacement of spent fuel or conditioned waste packages from reprocessing, and, finally, long-term period after closure of the repository.

It is to be noted that **this approach is based on a diversion scenario and does not take into consideration the proliferation risk of a country when no extrinsic barriers are in place** (e.g., withdrawal from NPT).

It does not allow a detailed analysis of proliferation risk associated with each step of the civilian nuclear fuel cycle industry, in particular, at enrichment and reprocessing plants. Such a work is not within the scope of a general study of this kind and requires a specific in-depth study of each facility, taking into account the particular safeguards and physical protection measures implemented in the facility.

3.4. Quantitative Assessment

3.4.1. The Grading Process

Having defined the framework and attributes for PR, the question was to find means to evaluate **the degree of proliferation resistance** of a nuclear system a proliferator would face, taking into account the various barriers. For this purpose, it was decided to adopt the following scale of values allowing a scoring of each barrier for each individual fuel cycle step:

Very weak barrier..0
Low resistant barrier...1
Moderately resistant barrier...................................2
Highly resistant barrier...3
Very highly resistant barrier....................................4

On the basis of this scale of values, a score was given by a panel of experts for each square of the matrix defined in paragraph 3.2. Then, for each individual fuel cycle step, scores were summed up and the average value normalized to one to obtain a PR index for each set of barriers (material, institutional…) and then for each phase (diversion, transportation…).

3.4.2. Specific Assumptions for Attribute Rating

It is important to explain hypothetical or technical inputs which have been taken into account by experts for score attribution.

First, we have considered that a country trying to "make" a nuclear weapon has the choice between two (and only two) devices, as far as is presently known[1], and according to unclassified information widely available in the open literature:

- A **"gun type" nuclear weapon**, the operating principle of which is to create a critical mass by driving together very rapidly (with the help of a chemical explosive) two or more sub-critical masses of fissionable material.
- An **"implosion type" nuclear weapon"**, the operating principle of which is to create a critical mass by "squeezing" (imploding), as fast as possible, a sub-critical mass (which may have the shape of a shell) of fissile material (the "pit") into a critical sphere with the help of a powerful inwardly-focused implosive shock wave created by means of "lenses" of classical explosives surrounding the fissile mass.

In this study we have clearly distinguished between these two options **and in the hierarchy of scores assigned in each case, we have taken into account, at least partly, the following points:**

- The **gun type** device is **relatively simple to make** and its **reliability is very high**. However, for kinetic reasons, **it is very hard to use plutonium** having significant portions of Pu-240 and Pu-238 (such as civilian plutonium from power reactor spent fuel), as these isotopes are intense sources of spontaneous neutrons and even more civilian plutonium. **Only high-enriched uranium can be “easily” used in a gun type device.**
- Conversely, an **implosion type device** is a much more sophisticated system and the making of a nuclear weapon of this type requires a fairly high level of scientific and technological background. Nevertheless, this option **allows the use of plutonium** and preferably for **reliability** and **efficiency** reasons, the use of plutonium having **as low as possible concentrations of Pu-240 and Pu-238** (typically, about 6% for Pu-240 and 0.01% for Pu-238, which is then "weapon grade" plutonium used in modern nuclear weapon arsenals).

The two other reasons for limiting the concentrations of Pu-240 and Pu-238 are:

- Heat generated by these isotopes; if the heat generated by the plutonium is too high, the classical explosive surrounding the fissionable mass can reach too high a temperature which may prevent it from exploding.
- Radiation emitted by these isotopes; strong radiation can be troublesome for the manufacturing process of the nuclear weapon, as the staff making the bomb can be dangerously irradiated (nevertheless, this kind of risk may be accepted by a determined proliferator).

Having said this, a proliferating state may choose the plutonium route (and, thus, the implosion device) in spite of the difficulties described above, because, once it has obtained the appropriate quantity of spent fuel, it is relatively easy to separate the plutonium from the spent fuel by mechanical and chemical processes which are well known today (the more so that the proliferator does not have to comply with the severe constraints of **industrial** reprocessing plants). Developing a **completely domestic** enrichment technology is a more difficult task, but a proliferating state may try to divert part of this technology. Another discriminating factor which is considered in this study is the chemical form of the material to be processed for making the nuclear weapon: metallic (uranium or plutonium), uranium-hexafluoride or oxide. In this study we have tried to take into account all these technical factors:

- the fact that it is easier to reprocess spent fuel than to enrich uranium
- the chemical form of the material
- the easiness of the weapon making process
- and, when "civilian plutonium" is used in an implosion device, the technical barriers resulting from heat generation (score is 3), neutron source (score is also 3), and radiation emission (score is 1).

From the above considerations, it is clear that, in fact, **it would be very difficult to manufacture a**

1 We do not speak here of other devices invlvng both fission and fusion processes in the same weapon (such as bosted fission systems) wich are much more sophsticated weapons.

reliable nuclear weapon with an implosion device using **"civilian plutonium"** recovered from spent fuel having a **high burn-up**, such as it is the case for all PWRs today (more than 50 GWd/tHM in most cases) and as it should be the case for HTRs with prismatic bloc core design (more than 100 GWd/tHM). For pebble bed HTRs, it would be easier a priori, because part of the pebbles have a low burn-up (e.g., 10 GWd/tHM or so) when they are unloaded after their first passing through the core (pebbles are recycled several times in the core, in order to reach a high burn-up of more than 100 GWd/tHM). However, this would be compensated by the fact that it would be necessary to divert (and reprocess) several hundreds of thousands of pebbles to obtain enough plutonium for the manufacturing of a nuclear weapon. For CANDU reactors using natural uranium, the situation is the same since the burn-up of their fuel is even less than 10 GWd/tHM. We have taken into account this technical feature by lowering the scores described above, that is 1 each for heat generation and neutron emission, and 0 for radiation.

3.4.3. Case Studies

In the frame of this paper, we will present in only one case (at the end of this paper) a figure where PR indexes (numbers between 0 and 1) are attributed to each phase of the fuel cycle. We consider a Non Nuclear Weapon State, with a high level of technological development, having a fleet of light water reactors and a reprocessing plant (thus, implementing a closed fuel cycle with MOX fuel) but having no enrichment plant (this is Case C in Table 1). **The country is supposed to comply with international safeguards measures and regulations but decides to manufacture nuclear weapons in a concealed way using its civilian nuclear facilities.**

First, one can immediately see in the figure that the PR index is equal to 1, when the fuel is in the reactor core, while the reactor is in operation. In fact, this index is set to 1, because we consider that it would be impossible to covertly divert an irradiated fuel assembly under such conditions without being detected, since a PWR must be shut down (and the lid of the vessel removed) to “steal” one or several fuel assemblies. The scoring process and the elementary aggregation method that we use in the methodology do not allow to take into account directly this kind of circumstances, where one of the barriers is insurmountable.

Now, the principal trend emerging from this particular case is that the **PR indexes of front end and back end are comparable, with or without reprocessing.** This results mainly from the consider-

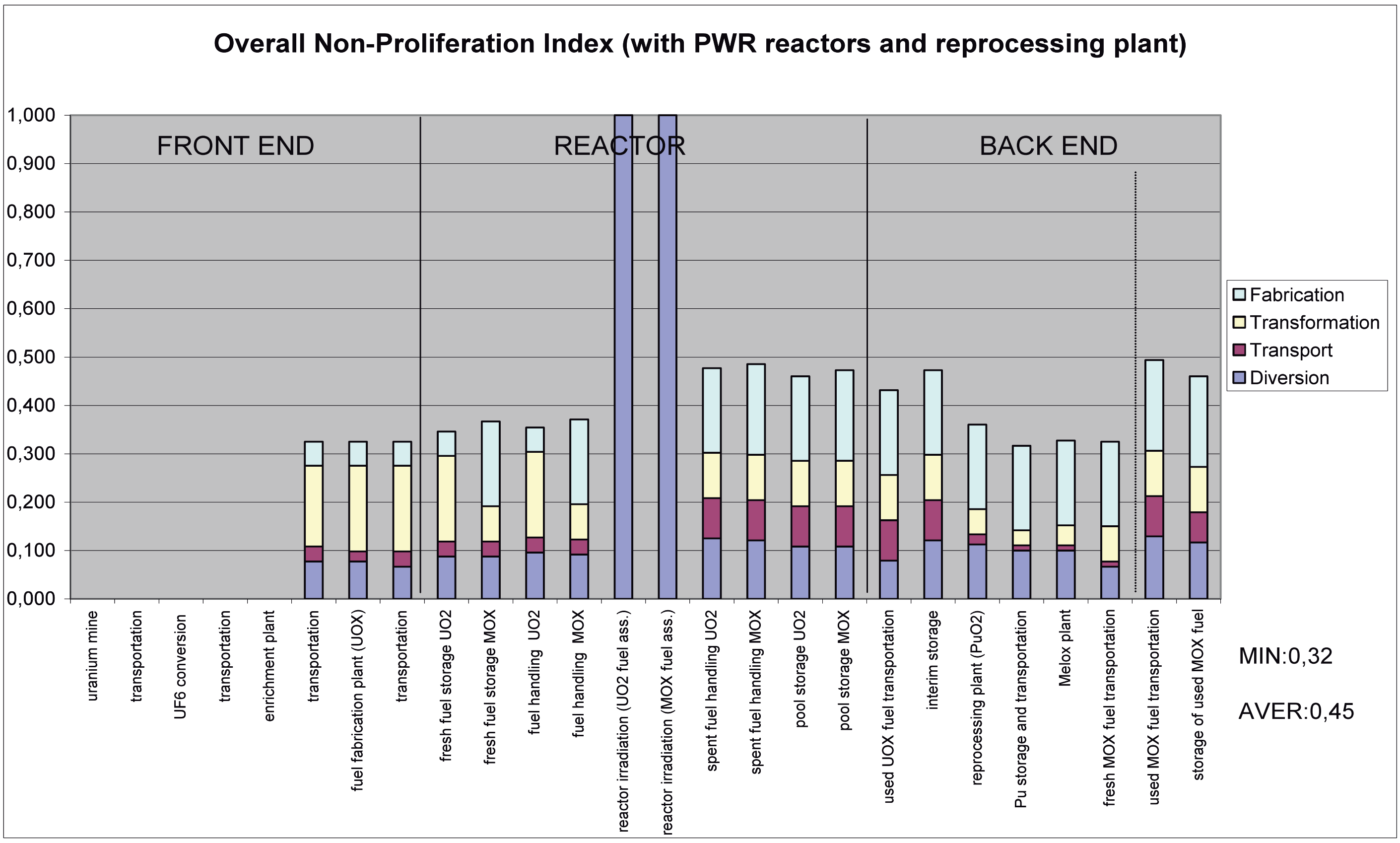

Figure 1: Overall non-proliferation index (with PWR reactors and reprocessing plant).

ations evoked in section 3.3.2. on the manufacturing of a nuclear weapon. In fact, this conclusion remains valid for a country which has no reprocessing or enrichment plant (Case A) or an enrichment plant but no reprocessing plant (Case B). This is illustrated by Table 1 which shows PR indexes per phase and the average overall index, within the same type of threat.

	Case A (Reactor alone : no fuel cycle facilities)	Case B (Reactor + enrich.)	Case C (Reactor + reprocess.)
Diversion	0,43	0,37	0,45
Transport	0,25	0,21	0,19
Transformation	0,61	0,53	0,41
Weapon fabrication	0,53	0,40	0,58
Total	**0,49**	**0,40**	**0,45**

Table 1: Summary of results for Cases A, B and C.

As expected, Case A has a higher overall non-proliferation characteristic than Cases B and C.

Beyond this result, we can derive from this analysis some more general outcomes.

3.5. Main Outcomes of the Study

3.5.1 On the Methodology

➜ The aggregation process of scores attributed to the various barriers to obtain a global “Proliferation Resistance Index” (PRI) for a given fuel cycle step, is a simple arithmetic sum, without any attempt to weight various barriers. This process is, of course, questionable for at least 2 reasons:

1 - A specific barrier can be regarded objectively as being more important than others and, nevertheless, get the same weight as others with this process. It may even have a reduced influence on the global PRI, if the number of other barriers is too high (a barrier can be sub-divided into “sub-barriers”, which, mathematically, reduces the weight of other barriers).

2 - The scale of values (from 0 for a “very low resistance” to 4 for a “very high resistance”) may not have the same meaning according to the type of barrier which is evaluated (for example, a score of 4 for the radiological barrier is not necessarily equivalent to a score of 4 for the isotopic barrier).

To cope with the first difficulty, we have tried to select a set of barriers as “homogeneous” and as independent as possible. It must be underlined that, in this kind of approach, the list of barriers must be carefully set up and each of them must be very well defined to avoid a possible misunderstanding between experts and imbalance of scores.

To cope with the second difficulty, we have paid much attention to the scoring of each of the barriers. One may think that expert subjectivity may impair the scoring process; but this exercise showed that, in fact, there has been, in most cases, a quick convergence (agreement) between experts regarding their allocation of scores.

However, the selection of barriers carried out in SAPRA and the annotation process are certainly not perfect and could be improved with further detailed discussion in an enlarged panel of experts. Anyhow, this shows simply that global **PRI are only rough indicators**, just allowing useful comparisons between different stages of the fuel cycle for different nuclear systems. They are not, nor intended to represent an absolute proliferation resistance measurement of a nuclear system in its entirety. In any case, they must be interpreted with care, and a precise or direct comparison between systems using these PRI may be meaningless. On the other hand, they can be used first and foremost to **identify weak points** and **relative merits** of nuclear systems, and this can **help to develop or implement appropriate and optimized measures to reduce their potential vulnerability** with regard to various threats.

➜ One of the improvements which could be implemented in SAPRA would be to develop an analytical approach for assessing the robustness of each barrier and to assemble these barriers by allocating weights to them. To do so, it could be interesting, for example, to take certain elements of the multi-attribute method (using in particular “utility functions”) developed by the Texas A&M University, and already utilized in the so-called “Blue Ribbon Report” (ref [5]). With such developments, the SAPRA method could be applied to study in more detail a particular stage of the fuel cycle (for example, reprocessing) and, then, different processes could be compared (which is not possible at this preliminary stage of development).

➜ Some of the barriers are clearly insurmountable (for example, theft of a fuel assembly inside a PWR reactor vessel, while the reactor is operating). Results cannot visibly reflect this kind of situation, since scores are not greater than 4, and since scores are simply added up (and normalized to one). Fortunately, such a case is exceptional (in fact, there is only the one we have just quoted), but the method could be probably adapted to deal with this particular case. In the present study, the value of the global PRI corresponding to this fuel cycle step (reactor irradiation) for a PWR has been set to 1 (of course, this is not the case of a reactor allowing on-load refuelling).

➜ The SAPRA method takes into account the fact that it is more or less difficult to manufacture nuclear weapons, depending on the nuclear material used (HEU or civilian plutonium of more or less “good” isotopic quality). We consider that this is an important feature that must be included explicitly in any proliferation resistance assessment study.

➜ The scale of values used to allocate a score to each of the barriers and the aggregation process to obtain global PRIs is such that one can never reach values close to one for PRIs. As a matter of fact, most of PRIs range from 0.25 to 0.75. This means that with this kind of approach, a PRI around 0.5 is a rather good score. This “severity” of annotations can be illustrated by the following example: the fuel cycle step “storage of spent fuel in reactor pools” gets an overall PRI of only 0.63 in case study N° 1 (theft in a foreign country). Now, it is perfectly clear that it would be very difficult (not to say impossible) to steal a spent fuel assembly from a pool of a PWR submitted to international safeguards, then to transport it to the country wanting to proliferate, then to reprocess it to recover the plutonium and, finally, to manufacture a nuclear weapon with this (degraded) plutonium, all of this without being detected. For better appreciating the more or less pessimistic character of the system of annotation, it would be interesting to be able to carry out a case study starting from a real example of attempt at proliferation from civilian nuclear facilities placed under international safeguards. However, such a case has never happened in the past, because all nations that have acquired nuclear weapons have done so using dedicated facilities and, therefore, this kind of evaluation cannot be based on experience.

➜ The SAPRA method, as it is developed today, cannot be applied to the particular threat resulting from a country which could denounce its engagement with respect to the NPT and, thus, which could be no longer subjected to any internationalsafeguards. In such a case, a similar approach may be developed, but the very structure of matrices must be completely revised.

➜ Generally speaking, this is a preliminary development phase of SAPRA and, in particular, additional effort will help to refine many of the discussions and ratings of the specific barriers to proliferation. Also, threats considered in this work may require further elaboration to ensure that we have adequately defined an overall assessment of the resistance to proliferation of nuclear systems.

3.5.2 On the Results

Until now, we have implemented the method by studying a selection of a few cases representing a combination of most of the current and forthcoming situations in nuclear countries worldwide. A lot of findings and comments may be drawn from these studies, but in the frame of this article we limit ourselves and give only the following general conclusions.

➜ Generally speaking, it is clear that there are no proliferation-proof nuclear systems, but this study demonstrates (or rather, confirms) that all nuclear systems feature a relatively **high resistance to proliferation, provided that comprehensive and efficient international safeguards can be implemented.** In other words, institutional (or “extrinsic”) measures to address proliferation resistance are of key, if not dominant, importance.

➜ Nevertheless, in most cases, it appears that the front end of the fuel cycle is less resistant to proliferation than the rest of the fuel cycle, and in this front end uranium mines appear to be the weakest part. This is largely so. because it would be much less difficult for a potential proliferant state to steal or to divert or simply to acquire, covertly, natural uranium rather than low enriched uranium (< 20 %) from a facility under international safeguards, and that, in both cases, the proliferant state needs anyway an enrichment step (which is the major obstacle to overcome) to obtain weapon usable material. Under these conditions, to start from more or less enriched uranium does not make much difference.

➜ This conclusion remains valid even for a closed fuel cycle involving the recycling of plutonium using MOX fuel. The main reason is that potential weaknesses linked to handling operations of the plutonium (including transport) are compensated by extremely rigorous and specific protection measures. Furthermore, plutonium which could be recovered from spent MOX fuel for the manufacturing of nuclear weapons is a far from attractive target for potential proliferant states because of the very poor isotopic composition of the plutonium contained in this spent MOX fuel.

➜ Nevertheless, back-end operations in the open fuel cycle appear more proliferation resistant than in the closed fuel cycle, with the important exception of the final disposal of spent fuel. As a matter of fact, it is clear that large amounts of spent fuel definitively stored in a geological disposal facility could become a potentially attractive target for a proliferant state, as it could contain huge quantities of plutonium (“plutonium mines” ranging from a few tons to several hundred tons) which will be less and less protected over time by a radiological barrier and very hard to monitor. This is the reason why this step appears to be the weakest part in the back-end of an open fuel cycle.

➜ In the DUPIC fuel cycle, all steps taking place after the first irradiation of the fuel in PWRs are clearly highly resistant to proliferation. This results from the very nature of this particular fuel cycle, in which uranium and plutonium remain always mixed with highly radioactive materials. The high level of radioactivity provides an efficient impediment to potentially use this fuel or these reprocessing facilities for proliferation purposes.

➜ HTR-based nuclear systems appear to be slightly less resistant to proliferation than PWR systems because of the front end part of their fuel cycle. HTR fuel requires a significantly higher enrichment than PWR fuel (15 % to 20 % for bloc fuel type HTRs instead of less than 5 % for current PWRs). However, this variation is not so important for the reasons already discussed above. For the back end of the fuel cycle, there is also a difference between HTRs and PWRs but in favour of HTRs. This is particularly due to the plutonium isotopic quality (even more degraded in HTRs than in PWRs), the number of fuel elements to be reprocessed (many more in HTRs than in PWRs), and the absence of a reprocessing technology for HTR fuel. It is to be noted that for the case of pebble bed HTRs (which has not been studied here) these conclusions could be modified because of very specific characteristics of their fuel and because of the loading / unloading mode of this fuel. However, one can say that the apparent drawback of this on-load refuelling mode (from a proliferation resistance point of view), could be compensated by the fact that it would be necessary to divert or steal (and reprocess) several hundreds of thousands of pebbles to obtain enough weapon-grade plutonium for the manufacturing of a nuclear explosive device. This case would deserve a more detailed study to enable a better assessment.

➜ Lastly, it is important to mention that SAPRA cannot be used (at least, as it is now) to deal with the threat resulting from a country having nuclear facilities, if it ABROGATES its adhesion to the NPT, and, thus, if it no longer authorizes any international safeguards in any of its facilities. One may suppose then, that this country intends to use these facilities to acquire nuclear weapons, and, in that case, the country has immediate access to the equipments containing nuclear material without any constraint. Then, with a minimum of skill and knowledge it appears that for a determined state, it is only a question of time and human resources to fabricate a nuclear weapon. The only intrinsic technical barriers in that case are those which may increase the difficulty and / or time delay associated with modifying or reconfiguring a facility or process to produce weapon usable materials, and this depends on the type of nuclear system existing in this country. From this point of view, the brief qualitative analysis of 3 cases presented in this report (PWR, CANDU, HTR), shows that HTRs seem to present a little bit more technical impediments than PWRs, and that CANDU reactors present less technical obstacles to such proliferation pathway. However, this acknowledgement is simply based on a partial analysis which would deserve to be more thorough. In any case, a pure technical analysis of this kind of threat using adapted methods would not be of a great help, because, in such a situation, it is clear that only measures of a diplomatic or political nature could be able to reduce this threat.

4. Conclusion

The proliferation resistance assessment methodology presented here is an example of implementation of the method of “barriers” which is, in a certain way, analogous to the approach used in nuclear safety analysis. It is based on an evaluation of the efficiency of material-related, technical, or institutional barriers against diversion or misuse by a country possessing civilian nuclear material or having developed technologies on its own territory or abroad. It is not a sophisticated method but rather a crude quantitative attempt to index or “measure” the proliferation resistance of a civilian nuclear fuel cycle at each of its steps. One of its main advantages is to offer the possibility to identify weak parts of a nuclear system with regard to proliferation risk. Preliminary results confirm that no “technical fixes” exist against nuclear proliferation but, nevertheless, prominent differences may occur between various fuel cycles depending on the context of their deployment. In any case, there is no substantial gap between global PR indexes for the front end and back end of the fuel cycle, and in both cases it appears that institutional barriers are of a paramount importance to prevent proliferation of nuclear weapons.

References

1. Non-Proliferation Assessment Methodology - US NNSA - ANS winter meeting 2002.
2. IAEA - Depart. of Safeguards - STR-232 - "PR Principles for Future Nuclear Energy Systems"
3. “Report of the TOPS task force of the Nuclear Energy Research Advisory Committee report”, USA, January 2001 and “Report of the international workshop on technology opportunities for increasing the PR of global civilian nuclear power systems”, March 2000.
4. “SAPRA : a Simplified Approach for Proliferation Resistance Assessments” – D. Greneche (AREVA), J.L. Rouyer (EDF), J.C. Yazidjian (Framatome). Report published in April 2006 (available to the author of this article).
5. “An evaluation of the proliferation characteristics of light water reactor fuel with the potential recycle in the United States” – Pacific Northwest Laboratory – November, 2004
6. “Proliferation Vulnerability Red Team Report” – Sandia National Laboratory – SAND97 – 8203 (1996)

Revised INPRO Methodology in the Area of Proliferation Resistance

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Abstract

The official INPRO User Manual in the area of proliferation resistance is being processed for the evaluation of innovative nuclear energy systems. Proliferation resistance is one of the goals to be satisfied for future nuclear energy systems in INPRO. The features of currently updated and released INPRO methodology were introduced on basic principles, user requirements and indicators. The criteria for an acceptance limit were specified. The DUPIC fuel cycle was evaluated based on the updated INPRO methodology for the applicability of the INPRO User Manual. However, the INPRO methodology has some difficulty in quantifying the multiplicity and robustness as well as the total cost to improve proliferation resistance. Moreover, the integration method for the evaluation results still needs to be improved.

Keywords: proliferation resistance; basic principle; user requirements; indicators; acceptance limit.

1. Introduction

Nuclear fission reactors are still expected to offer the possibility of meeting the world's energy needs for the next generation, because nuclear energy is seen to be a sustainable source of energy. Furthermore, nuclear energy sources do not emit green house gases or gases that lead to the production of acid rain. The IAEA initiated the INPRO (International Project on Innovative Nuclear Reactors and Fuel Cycles) program in 2000. INPRO proposed proliferation resistance (PR) as a key component of a future Innovative Nuclear System (INS) for fulfilling the energy needs in the 21st century along with sustainability, economics, environment, safety of nuclear installations and waste management. A set of Basic Principles (BP), User Requirements (UR), and Criteria (CR) including Indicators and Acceptance Limits (AL) has been developed.

INPRO is focused on the possible contribution of an INS to a weapons programme in a given State. INPRO assesses the whole INS in a specific State or region throughout a full life cycle, not only separate elements of innovative nuclear systems. The INPRO evaluation methodology for PR is to confirm that an adequate level of PR has been achieved in an INS, and also gives some guidance to a developer on how to improve PR. In this study, the features of the updated INPRO methodology in the INPRO User Manual were introduced. It is noted that the indicators and their subsequent variables are better classified and rearranged for applying the INPRO methodology to a realistic evaluation of an INS.

Finally, an application of the INPRO methodology was carried out on the DUPIC (Direct use of PWR spent fuel in CANDU reactors) fuel cycle to assess the adequacy of the revised INPRO methodology with new indicators as a practice.

2. INPRO methodology for Proliferation Resistance

2.1. Structure of the INPRO User Manual

The degree of proliferation resistance results from a combination of technical design features, operational modalities, institutional arrangements, and safeguards measures. In the user manual [1], the previous BP1 and BP2 [2] were combined to make one BP, because BP2 was regarded as complementary to BP1. The URs corresponding to BP1 and BP2 were rearranged as well. Five URs were categorized for the basic principle of PR to provide guidance to a government, sponsors, designers, regulators, investors and other users of nuclear power and fuel cycle facilities, which incorporate the PR of a future nuclear energy system. The criteria were set up based on the objectives of a UR, and each UR has different and independent indicators. Therefore, all the indicators under a UR involve all the essential elements to present a UR. Each indicator specifies the evaluation parameters with the acceptance limits to decide and provide guidance on the actual evaluation results of an INS. Table 1 shows the structure for the evaluation scheme.

2.2. Basic Principle, User Requirements and Criteria

INPRO has defined one basic principle in the area of PR. The BP emphasizes the importance of both intrinsic features and extrinsic measures for achieving proliferation resistance. The basic principle implies that intrinsic features and extrinsic measures for PR be implemented throughout a full life cycle of an INS. The intrinsic features and extrinsic measures are the most important barriers for proliferation resistance, and each indicator itself is also a good metric for a barrier against a proliferation.

Requirement UR1 is to be fulfilled by the State. Commitments, obligations and policies of a State regarding non-proliferation have a considerable impact on the proliferation resistance of an INS. There are two criteria specified for UR1. UR 2 refers to material attractiveness, four indicators have been defined, and each indicator has various evaluation parameters. The attractiveness of nuclear material (NM) could be illustrated by two intrinsic features; the conversion time and the significant quantity. Regarding the detectability of a diversion (UR3), six indicators are proposed for describing those features of an INS that may facilitate or impede the implementation of the IAEA safeguards. UR3 requires that a diversion should be reasonably difficult and detectable. UR4 is related to Fundamental Principle I of physical protection [3] that asks for reflection of a concept of several layers and methods of protection in the design of a nuclear energy system. A prerequisite for the assessment of this user requirement is the existence of an acquisition/diversion path analysis to be performed by PR experts. UR5 recognizes that there are cost trade-offs between intrinsic features and extrinsic measures, and encourages their optimization for cost effectiveness. The specific purpose and contents for the BP, URs and Indicators are summarized in Table 1.

2.3. Evaluation Method

To assess the PR of an INS in terms of evaluation parameters, evaluation scales are required. Some barriers can be quantified but other barriers, such as extrinsic measures or safeguardability, may be expressed only in a logical value such as "Yes" or "No". The present study suggests a five stage scale such as VW(Very Weak), W(Weak), M(Moderate), S(Strong), and VS(Very Strong) regarding the quantifiable evaluation parameters. For a logical scale, U(Unacceptable) and A(Acceptable) for extrinsic measures and W(Weak) and S(Strong) for some intrinsic features related to safeguardability are suggested. Most quantified scales of the evaluation parameters are referenced in [4].

The key to the bottom-up approach for an evaluation is to determine if a nuclear energy system can meet the acceptance limits suggested in INPRO and then to judge the higher level requirements. The starting point for the analysis should be indicator 1 of UR1, because it will be a common indicator for all the identified nuclear system components.

2.3.1. Indicators of UR1

The indicators are on a State's commitments, obligations and policies, and institutional structural arrangements in PR. Nine evaluation parameters [1] are listed in the first indicator, and the second indicator is evaluated through the review of facility/enterprise undertakings to provide PR, and it contains three evaluation parameters for the extrinsic measures.

2.3.2. Indicators of UR2

There are four indicators in UR2: Material quality, material quantity, material form, and nuclear technology [1]. The features related to the quality of nuclear material are: 1) Isotopic composition, 2) radiation field, 3) heat generation rate, 4) material type, 5) spontaneous neutron generation rate. The intrinsic features related to the quantity of nuclear material are: 1) Mass of an item, 2) mass of NM in bulk material (dilution), 3) number of items to obtain a significant quantity (SQ) of NM, 4) number of SQ in throughput of a facility. The intrinsic features related to the form of nuclear material are the chemical and/or physical form defining the difficulty and effort necessary to produce weapon usable material. The evaluation parameters for the attractiveness of nuclear technology include enrichment, extraction of fissile material and irradiation capability of undeclared nuclear material.

2.3.3. Indicators of UR3

There are six indicators in UR3: Accountability, amenability, detectability of NM, difficulty to modify process, difficulty to modify a facility design and detectability to misuse technology or facilities. The criterion for accountability deals with the detectability of diversion and defines the necessary quality of the measurement system used in a nuclear facility to control the flow and inventory of NM. The objective is to achieve an amount of "material unaccounted for" (MUF) [5] equal or less in comparison to ex-

Basic Principle BP: *Proliferation resistance intrinsic features and extrinsic measures shall be implemented throughout the full life cycle for innovative nuclear energy systems to help ensure that INS will continue to be an unattractive means to acquire fissile material for a nuclear weapons program. Both intrinsic features and extrinsic measures are essential, and neither shall be considered sufficient by itself.*

User Requirements (UR)	Criteria (CR)	
	Indicator (IN)	Acceptance Limits (AL)
UR1 State commitments: States' commitments, obligations and policies regarding non-proliferation and its implementation should be adequate to fulfill international standards in the non-proliferation regime.	1.1: States' commitments, obligations and policies regarding non-proliferation established?	1.1: Yes, in accordance with international standards
	1.2: Institutional structural arrangements in support of PR have been considered?	1.2: Yes
UR2 Attractiveness of NM and technology: The attractiveness of nuclear material (NM) and nuclear technology in an INS for a nuclear weapons program should be low. This includes the attractiveness of undeclared nuclear material that could credibly be produced or processed in the INS.	2.1: NM quality.	2.1: Attractiveness based on NM characteristics considered in design of INS and found acceptably low based on expert judgment
	2.2: NM quantity	2.2=2.1
	2.3: NM form	2.3=2.1
	2.4: Nuclear technology	2.4: Attractiveness of technology considered in design of INS and found acceptably low based on expert judgment.
UR3 Difficulty and detectability of diversion: The diversion of nuclear material should be reasonably difficult and detectable. Diversion includes the use of an INS facility for the production or processing of undeclared material	3.1: Accountability.	3.1: Based on expert judgment equal or better than existing designs, meeting international state of practice.
	3.2: Amenability for C/S measures and monitoring	3.2: Based on expert judgment equal or better than existing designs, meeting international best practice.
	3.3: Detectabilty of NM	3.3: Based on expert judgment equal or better than existing facilities
	3.4: Difficulty to modify process.	3.4: Based on expert judgment equal or better than existing designs, meeting international best practice.
	3.5: Difficulty to modify facility design	3.5=3.4
	3.6: Detectability to misuse technology or facilities	3.6=3.4
UR4 Multiple barriers: Innovative nuclear energy systems should incorporate multiple proliferation resistance features and measures.	4.1: The extent to which the INS is covered by multiple intrinsic features and extrinsic measures.	4.1: All plausible acquisition paths are covered by extrinsic measures on the facility or State level and by intrinsic features which are compatible with other design requirements.
	4.2: Robustness of barriers covering each acquisition path.	4.2: Robustness is sufficient based on expert judgment.
UR5 Optimization of design: The combination of intrinsic features and extrinsic measures, compatible with other design considerations, should be optimized to provide cost-efficient proliferation resistance.	5.1: PR has been taken into account as early as possible in the design and development of the INS.	5.1: Yes.
	5.2: Cost of incorporating into an INS those intrinsic features and extrinsic measures, which are required to provide or improve proliferation resistance.	5.2: Minimal total cost of the intrinsic features and extrinsic measures over the life cycle of the INS implemented to increase PR.
	5.3: Verification approach with a level of extrinsic measures agreed upon between the State and verification authority.	5.3: Yes

Table 1: Structure of the URs and Criteria for the BP of a proliferation resistance.

isting facilities in accordance with international standards. The detectability of diversion of NM can be enhanced by the installation of C/S measures and monitoring systems. Three evaluation parameters are involved: Amenability of containment measures, amenability of surveillance measures, and amenability of monitoring systems. The detectability of NM is related to the easiness of identifying/recognizing type and composition of nuclear material. The possibility to identify NM by NDA and detectability of the radiation signature are the evaluation parameters. The difficulty of modifying a process depends on the complexity of a modification, the cost for a process modification, the safety implication of such modification, and the time required to perform the relevant modification. The modification of a facility design might be detected by design information verification measures. The probability of detecting the misuse of technology or facilities is linked to the transparency of the facility design and process and to the availability of data.

2.3.4. Indicators of UR4

The criterion, the extent to which the INS is covered by multiple intrinsic features and extrinsic measures, asks for the results of an acquisition/diversion path analysis confirming that all plausible acquisition paths of the INS have been covered by multiple intrinsic features and extrinsic measures. The second indicator is the robustness (strength) of the barriers covering each acquisition path.

2.3.5. Indicators of UR5

The developer should consider proliferation resistance as soon as sufficient technical information is available in the development of a new INS. Governments should consider proliferation resistance as

Indicator (IN)	Evaluation Parameter, EP		Evaluation scale: VW	W	M	S	VS
Material quality	Material type		UDU[1]	*IDU*[2]	LEU	NU	DU
	Isotopic composition	^{239}Pu/Pu(wt%)	W			S	
			>50			*<50*	
		^{232}U contam. for ^{233}U(ppm)	<400	400~ 1,000	1,000~ 2,500	2,500~ 25,000	>25,000
	Radiation field	Dose (mSv/hr) at 1 meter	<150	*150~ 350*	350~ 1,000	1,000- 10,000	>10,000
	Heat generation	^{238}Pu/Pu(wt%)	*<20*			>20	
	Spontaneous neutron generation rate	(^{240}Pu+^{242}Pu)	>50			*<50*	
Material quantity	Mass of an item (kg)		10	*10~100*	100~500	500~ 1,000	>1,000
	Mass of bulk material for SQ (dilution) (kg)		10	10~100	100~500	*500~ 1,000*	>1,000
	No. of items for SQ		1	1~10	*10~50*	50~ 100	>100
	No. of SQ (material stock or flow)		>100	50~100	10~50	10~1	*<1*
Material form	Chemical/ physical form	U	Metal	Oxide/ Solution	U compound	*Spent fuel*	Waste
		Pu	Metal	Oxide/ Solution	Pu compound	*Spent fuel*	Waste
		Thorium	Metal	Oxide/ Solution	Th compound	Spent fuel	Waste

[1]Un-irradiated Direct Use Material, [2] Irradiated Direct Use Material.

Table 2: Evaluation of UR2.

soon as there is a firm plan for deployment of an INS. The second indicator for this criterion is the cost of incorporating intrinsic features and extrinsic measures, which are required to provide or improve proliferation resistance. The last criterion for this UR demands that an INS must have a verification approach with a level of extrinsic measures agreed upon between the verification authorities and the State.

3. Application to DUPIC Fuel Cycle

The basic concept of the DUPIC fuel cycle [6] is to fabricate CANDU nuclear fuel from PWR spent fuel by use of dry thermal/mechanical processes. Since no separation of the fission products and transuranic materials occurs in the process, the process materials are very radioactive throughout the whole manufacturing process. Therefore, access to the nuclear materials is extremely difficult which is strongly in favor of proliferation resistance. The material type is characterized as an irradiated direct use material. The isotopic composition, $^{239}Pu/Pu$, is ~60 wt%. Regarding the radiation field, the dose rate of a DUPIC fuel bundle is ~0.15 Sv/hr. The heat generation rate is related mostly to $^{238}Pu/Pu$ which is 1.7 wt%. A spontaneous neutron generation comes from (^{240}Pu+^{242}Pu)/Pu and it is ~30 wt%.

Table 2 shows the result of the evaluation of UR2. From the table it can be seen, that it is impossible to extract fissile materials and to modify the DUPIC facility.

4. Discussion and Conclusion

The updated evaluation methodology for the proliferation resistance area presented in the INPRO User Manual, using a basic principle, user requirements and indicators, is very informative and a big step forward to assess the degree of proliferation resistance of a nuclear energy system quantitatively. It was very clear to understand the PR evaluation methodology and to draw conclusions on a nuclear system by trying to apply the methodology to the DUPIC case.

The INPRO evaluation methodology provides a strategic structure, but it does not provide a deterministic method of proliferation resistance. An integrated decision method on the evaluation results is required. It was also noted that a quantitative evaluation method should be developed, both for policy decision makers and for system designers, in order to help them identify which innovative nuclear energy system is more resistant against proliferation, and what technical options are needed to enhance PR. Therefore, specific evaluation parameters need to be developed to assess the robustness and cost effectiveness for the real application of the INPRO User Manual on innovative nuclear systems.

Acknowledgements

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References

[1] International Atomic Energy Agency. Guidance for the Application of an Assessment Methodology for Innovative Nuclear Energy Systems, INPRO Manual - Proliferation Resistance Volume 5 of the Final Report of Phase 1 of the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), IAEA-TECDOC-1575, October 2007.

[2] International Atomic Energy Agency; Methodology for the assessment of innovative nuclear reactors and fuel cycles; Report of Phase 1B (first part) of the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO); IAEA TECDOC-1434; IAEA; Vienna; 2004.

[3] International Atomic Energy Agency; Nuclear Security–Measures to Protect against Nuclear Terrorism, Amendment to the Convention on the Physical Protection of NM; Fundamental Principles of PP of Nuclear Material and Nuclear Facilities; GOV/INF/2005/10-GC(49)/INF/6; Vienna; 2005.

[4] International Atomic Energy Agency; Design Measures to Facilitate Implementation of Safeguards at Future Water Cooled Nuclear Power Plants; IAEA Technical Report Series TRS-392; IAEA; Vienna; 1998.

[5] International Atomic Energy Agency; IAEA Safeguards Glossary 2001 Edition; International Nuclear Verification Series No. 3; Vienna; 2001.

[6] Korea Atomic Energy Research Institute; Korean Assessment of the Proliferation Resistance on the Whole Fuel Cycle of DUPIC; 2006.

GIF Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems: Overview and Perspectives

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Abstract

The Generation IV International Forum (GIF) was initiated in 2000 and formally chartered in mid 2001. It was set-up as an international collective representing the governments of ten Countries (Argentina, Brazil, Canada, France, Japan, the Republic of Korea, the Republic of South Africa, Switzerland, the United Kingdom and the United States) strongly involved in the deployment and development of nuclear technology for energy production. The European Atomic Energy Community (EURATOM), represented by the European Commission, signed the GIF charter on July 30, 2003. The People's Republic of China and the Russian Federation signed the GIF charter in November 2006.

The Technology Goals for Generation IV nuclear energy systems, developed during the Roadmap project [1], highlight Proliferation Resistance and Physical Protection (PR&PP) as one of the four goal areas for these technologies, along with Sustainability, Safety & Reliability, and Economics. On the basis of these four goal areas an evaluation methodology was developed which contributed to identify the six nuclear energy systems (NES) options currently under consideration by GIF.

The Generation IV Roadmap recommended the development of a comprehensive evaluation methodology to assess PR&PP of Generation IV nuclear energy systems. Accordingly the PR&PP Expert Group was formed and tasked by the GIF in December 2002 to develop an improved evaluation methodology on the basis of the Roadmap's recommendation. The group includes members of the GIF and representatives from the IAEA.

The methodology is organised as a progressive approach applying alternative methods at different levels of thoroughness as more design information becomes available and research improves the depth of technical knowledge. To date, the overall framework of the methodology is considered rather accepted and stable; the methodology was advanced with a development case study and has been tested though a demonstration case study.

This paper provides an updated overview of the methodology approach developed by the PR&PP Expert Group. The paper also highlights some of the achievements and the lessons learned during the demonstration case study carried out in 2005-2006, in which different techniques have been applied for the implementation of the PR&PP evaluation framework. Finally the paper presents some of the ongoing activities and future directions for the activity of the group.

Keywords: Proliferation Resistance, Physical Protection Robustness, Evaluation/Assessment.

1. Introduction

The technology goals for Generation IV Nuclear Energy Systems (NES) highlight Proliferation Resistance and Physical Protection (PR&PP) as one of the four goal areas along with Sustainability, Safety and Reliability, and Economics [1].

In particular, Generation IV Nuclear Energy Systems will increase the assurance that they are a very unattractive and the least desirable route for diversion or theft of weapons-usable materials, and provide increased physical protection against acts of terrorism. According to PR&PP Expert Group, the following definitions apply:

- Proliferation Resistance (PR) is that characteristic of a NES that impedes the diversion or undeclared production of nuclear material or misuse of technology by the Host State seeking to acquire nuclear weapons or other nuclear explosive devices.

- Physical Protection (robustness) (PP) is that characteristic of a NES that impedes the theft of materials suitable for nuclear explosives or radiation dispersal devices (RDD) and the sabotage of facilities and transportation by sub-national entities and other non-Host State adversaries.

The Generation IV Roadmap recommended the development of an evaluation methodology to assess NESs with respect to PR&PP. Accordingly, the Generation IV International Forum formed an Expert Group in December 2002 to develop a methodology. The development of the methodology has been documented in a number of progress reports and has been presented at several conferences [see, e.g., 2-7].

In section 2, this paper provides an updated overview of the methodology approach developed by the PR&PP Expert Group [8, 9]. Section 3, highlights some of the achievements and the lessons learned during the demonstration case study carried out in the period 2005-2006, in which different techniques have been applied for the implementation of the PR&PP evaluation framework [10].

Finally, in section 4, the paper presents some of the ongoing activities and of the future directions for the activity of the PR&PP Working Group[1].

2. Overview of PR&PP Methodology

In this section, a brief overview of the PR&PP methodology is given on the basis of the executive summary of revision 5 of the methodology report [8]: Figure 1 illustrates the PR&PP methodological paradigm. For a given system, analysts define a set of challenges, analyze system response to these challenges, and assess outcomes. The challenges to the NES are the threats posed by potential proliferant States and by sub-national adversaries. The technical and institutional characteristics of the Generation IV systems are used to evaluate the response of the system and determine its resistance to proliferation threats and robustness against sabotage and terrorism threats. The outcomes of the system response are expressed in terms of PR&PP measures and assessed.

The evaluation methodology assumes that a NES has been at least conceptualized or designed, including both the intrinsic and extrinsic protective features of the system. Intrinsic features include the physical and engineering aspects of the system; extrinsic features include institutional aspects such as safeguards and external barriers. A major thrust of the PR&PP evaluation is to elucidate the interactions between the intrinsic and the extrinsic features, study their interplay, and then guide the path toward an optimized design that identifies and minimizes vulnerabilities.

1 The PR&PP Expert Group was renamed PR&PP Working Group, for coherence with the other crosscuting groups established within GIF.

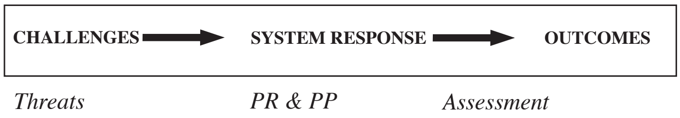

Figure 1: Paradigm for the PR&PP Evaluation Methodology.

The structure for the PR&PP evaluation can be applied to the entire fuel cycle or to portions of a NES. The methodology is organized as a progressive approach to allow evaluations to become more detailed and more representative as system design progresses. PR&PP evaluations should be performed at the earliest stages of design when flow diagrams are first developed in order to systematically integrate proliferation resistance and physical protection robustness into the designs of Generation IV NESs along with the other high-level technology goals of sustainability, safety and reliability, and economics. This approach provides early, useful feedback to designers, program policy makers, and external stakeholders from basic process selection (e.g., recycling process and type of fuel), to detailed layout of equipment and structures, to facility demonstration testing. Figure 2 provides an expanded outline of the methodological approach. The first step is threat definition. For both PR and PP, the threat definition describes the challenges that the system may face and includes characteristics of both the actor and the actor's strategy. For PR, the actor is the Host State for the NES, and the threat definition includes both the proliferation objectives and the capabilities and strategy of the Host State. For PP threats, the actor is a sub-national group or other non-Host State adversary. The PP actors' characteristics are defined by their objective, which may be either theft or sabotage, and their capabilities and strategies.

To facilitate the comparison of different evaluations, a standard Reference Threat Set (RTS) can be defined, covering the anticipated range of actors, capabilities, and strategies for the time period being considered. Reference Threat Sets should evolve through the design and development process of nuclear fuel cycle facilities, and for physical protection ultimately becoming Design Basis Threats (DBT) upon which regulatory action is based.

For PR, the threats include:

- Concealed diversion of declared materials;
- Concealed misuse of declared facilities;
- Overt misuse of facilities or diversion of declared materials;
- Clandestine dedicated facilities.

For PP the threats include:

- Radiological sabotage;
- Material theft;
- Information theft.

The PR&PP methodology does not determine the probability that a given threat might or might not occur. Therefore, the selection of what potential threats to include is performed at the beginning of a PR&PP evaluation, preferably with input from a peer review group organized in coordination with the evaluation sponsors. The uncertainty in the system response to a given threat is then evaluated independently of the probability that the system would ever actually be challenged by the threat. In other words, PR&PP evaluations are contingent on the challenge occurring.

The detail with which threats can and should be defined depends on the level of detail of information available about the NES design. In the earliest stages of conceptual design, where detailed information is likely limited, relatively stylized but reasonable threats must be selected. Conversely, when design has progressed to the point of actual construction, detailed and specific characterization of potential threats becomes possible.

When threats have been sufficiently detailed for the particular evaluation, analysts assess System Response, which has four components:

1. System Element Identification. The NES is decomposed into smaller elements or subsystems at a level amenable to further analysis. The elements can comprise a facility (in the systems engineering sense), part of a facility, a collection of facilities, or a transportation system within the identified NES where acquisition (diversion) or processing (PR) or theft/sabotage (PP) could take place.
2. Target Identification and Categorization. Target identification is conducted by systematically examining the NES for the role that materials, equipment, and processes in each element could play in each of the strategies identified in the

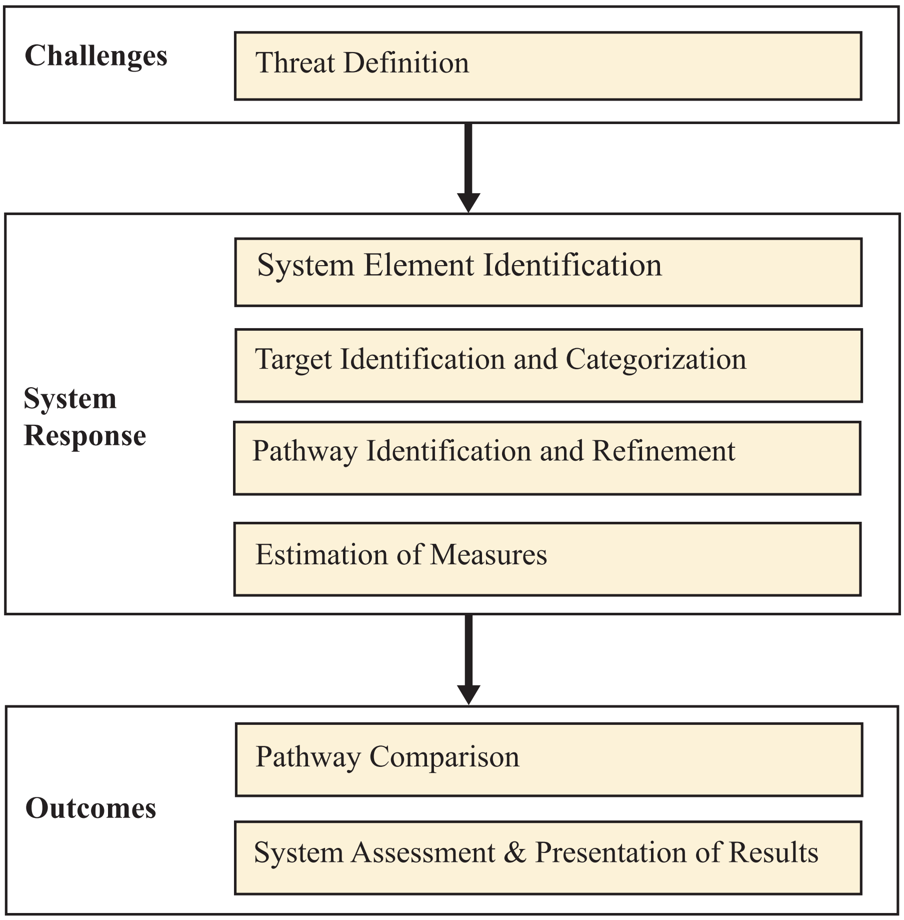

Figure 2: Detailed Framework for the PR&PP Evaluation Methodology.

threat definition. PR targets are nuclear material, equipment, and processes to be protected from threats of diversion and misuse. PP targets are nuclear material, equipment, or information to be protected from threats of theft and sabotage. Targets are categorized to create representative or bounding sets for further analysis.

3. Pathway Identification and Refinement. Pathways are potential sequences of events and actions followed by the actor to achieve objectives. For each target, individual pathways are divided into segments through a systematic process, and analyzed at a high level. Segments are then connected into full pathways and analyzed in detail. Selection of appropriate pathways will depend on the scenarios themselves, the state of design information, the quality and applicability of available information, and the analyst's preferences.
4. Estimation of Measures. The results of the system response are expressed in terms of PR&PP measures. Measures are the high-level characteristics of a pathway that affect the likely decisions and actions of an actor and, therefore, are used to evaluate the actor's likely behavior and the outcomes. For each measure, the results for each pathway segment are aggregated as appropriate to compare pathways and assess the system so that significant pathways can be identified and highlighted for further assessment and decision making.

For PR, the measures are:

- Proliferation Technical Difficulty (TD) – The inherent difficulty, arising from the need for technical sophistication and materials handling capabilities, required to overcome the multiple barriers to proliferation.
- Proliferation Cost (PC) – The economic and staffing investment required to overcome the multiple technical barriers to proliferation including the use of existing or new facilities.
- Proliferation Time (PT) – The minimum time required to overcome the multiple barriers to proliferation (i.e., the total time planned by the Host State for the project).
- Fissile Material Type (MT) – A categorization of material based on the degree to which its characteristics affect its utility for use in nuclear explosives.
- Detection Probability (DP) – The cumulative probability of detecting a proliferation segment or pathway.
- Detection Resource Efficiency (DE) – The efficiency in the use of staffing, equipment, and funding to apply international safeguards to the NES.

For PP, the measures are:

- Probability of Adversary Success (PS) – The probability that an adversary will successfully complete the actions described by a pathway and generate a consequence.
- Consequences (C) – The effects resulting from the successful completion of the adversary's action described by a pathway.
- Physical Protection Resources (PPR) – the staffing, capabilities, and costs required to provide PP, such as background screening, detection, interruption, and neutralization, and the sensitivity of these resources to changes in the threat sophistication and capability.

By considering these measures, system designers can identify design options that will improve system PR&PP performance. For example, designers can reduce or eliminate active safety equipment that requires frequent operator intervention.

The final steps in PR&PP evaluations are to integrate the findings of the analysis and to interpret the results. Evaluation results should include best estimates for numerical and linguistic descriptors that characterize the results, distributions reflecting the uncertainty associated with those estimates, and appropriate displays to communicate uncertainties.

The information is intended for three types of users: system designers, program policy makers, and external stakeholders. Thus, the analysis of the system response must furnish results easily displayed with different levels of detail. Program policy makers and external stakeholders are more likely to be interested in the high-level measures, while system designers will be interested in measures and metrics that more directly relate to the optimization of the system design.

3. The Demonstration Case Study: Achievements and Lessons Learned

For the development of the PR&PP methodology, Argonne National Laboratory (ANL) defined a notional Generation IV Sodium Fast Reactor (named Example Sodium Fast Reactor - ESFR -). It is a hypothetical Nuclear Energy System consisting of four, medium-sized, sodium-cooled fast reactors, rated at 300MWe each. The reactors are co-located with a shared dry fuel storage facility and with a fuel cycle facility operating a pyro-chemical processing of the ESFR spent fuel and providing also re-fabrication of new ESFR fuel elements. The reactors are connected to the dry storage facility by means of a staging washing machine operating in a fuel service facility. The site also foresees the presence of storages for all kinds of processing wastes (including excess uranium). The baseline design of the ESFR is that of an actinide burner, with a trans-uranics (TRU) conversion ratio of 0.64. Due to the burner core configuration, the system receives an external feed of material consisting of 56 light water reactor (LWR) spent fuel elements per year. The LWR spent fuel elements are processed on site together with the ESFR spent fuel elements for the re-fabrication of the ESFR fresh fuel elements. LWR spent fuel elements provide the needed feed of both fertile and fissile material.

This section summarizes some of the results achieved during the first demonstration case study which was carried out by the group in the period 2005-2006. The text is mainly based on the executive summary of the Demonstration Study Interim Report [10]. The demonstration study aimed at demonstrating the application of the PR&PP framework for proliferation resistance (PR) evaluation to elements of the ESFR nuclear energy system.

Three evaluation approaches, the qualitative evaluation approach, the logic trees approach (based on the application of Event Trees and Fault Trees), and the Markov approach, were each applied by a different task group of the PR&PP Working Group. Selected for the demonstration study was a portion, or “slice,” of the Fuel Cycle Facility (FCF) for the Example Sodium Fast Reactor (ESFR).

3.1 The Pyro-processing Fuel Cycle Facility

The FCF modeled is a pyro-chemical reprocessing (pyro-processing) facility designed to accept the spent sodium-bonded, metallic fuel from four advanced fast reactors and to convert it into three output streams (new fuel assemblies, metal waste ingots, and ceramic waste forms). Pyro-processing is a process that separates uranium, transuranics, and fission products using electrochemically driven transport between molten salt and metal phases. Under normal operation, the processes do not separate plutonium from the minor actinides, and therefore, all material handling occurs remotely in hot cells, where personnel access does not occur except under highly special circumstances. The pyro-processing technology as applied in the facility for the demonstration study has five main process steps.

1. Spent fuel assemblies are disassembled and the resulting fuel elements are mechanically chopped.
2. Chopped elements are electro-refined to partially separate the uranium from fission products and actinide elements. This step generates a uranium material, which is further processed to remove adhered salt and produce the uranium (U) product. This second step also generates metal waste resulting from undissolved cladding hull pieces.
3. This step consists of recovering the transuranic (TRU) material that is present in the salt used for uranium electro-refining. Similar to the uranium material, TRU/U material recovered is further processed to remove adhered salt and produce the TRU/U product.
4. The U product, TRU/U product, and makeup materials are melted together to produce fuel slugs. Fuel elements are then fabricated from these slugs and assembled into fuel assemblies to be returned to the co-located reactors. In this step, external material resulting from the processing of LWR spent fuel (uranium and external TRU metal) is added to the process.
5. The final step consists of conditioning the metal and salt wastes generated by the second and third steps, respectively, and producing ceramic/metal waste forms for disposal.

Figure 3 shows the process steps, the material flows and includes elements of the safeguards system, defined by the group for the purpose of the study.

Note that for the scope of the demonstration case study, the process part dealing with the treatment of the LWR spent fuel elements was not considered, hence only the resulting feeds of U and of TRU are reported in Figure 3.

3.2 The Safeguards System

Three Material Balance Areas (MBA) were defined for the safeguards of the demonstration slice.

- MBA-1: The spent fuel element disassembly process occurs in the Receiving/Shipping Cell. In this MBA, spent fuel assembly items are disassembled into spent fuel elements.
- MBA-2: Electrochemical processing occurs in the Process Cell. An MBA was defined to cover all the process area.

- MBA-3: New fuel manufacturing occurs in the Fresh Fuel Hall and the Assembly Fabrication that occurs in the Receiving/Shipping Cell.

It was assumed that the safeguards controls to be installed around the MBAs will use neutron counters, cameras, seals, and a material accountability system based on an initial evaluation of the Cm/Pu ratio performed on the spent fuel as it enters the facility.

3.3 The Threat and the Evaluations

The assumed proliferation scenario was that of a host-state diversion. For this study, the threat definition corresponded to the characteristics generally considered for a reactor state with technical know-how and industrial infrastructure. The host state was assumed to be a non-nuclear weapons state (NNWS), a signatory to the NTP and to have an Additional Protocol (AP) in force. The objective of the proliferators was to divert covertly 1 significant quantity (SQ) equivalent of nuclear material from the FCF within one year without detection by safeguards and process the diverted nuclear material in clandestine facilities.

The paradigm for the proliferation resistance methodology, developed by the PR&PP Expert Group, is composed of three elements (see Figure 1). In this paradigm, for a given system a set of challenges is identified; the system response to these challenges is analyzed, and outcomes are determined.

The system response involves:

- Subdividing the nuclear system into ‘elements’ (a facility, part of a facility, a collection of facilities, or a transportation system);
- Identifying potential targets within each element (targets are the nuclear material and processes to be protected from PR threats), and

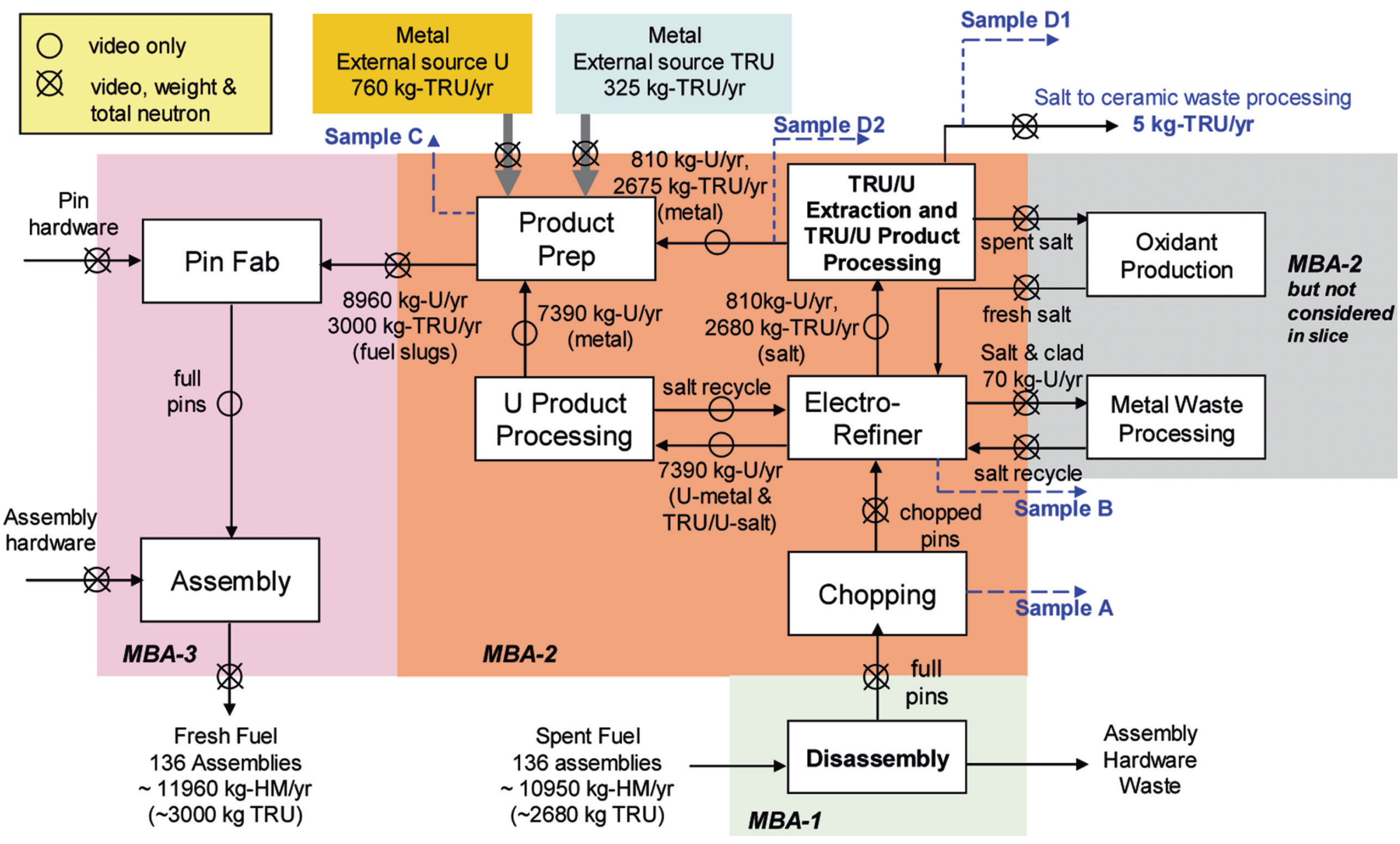

Figure 3: Process steps, material flows and elements of the safeguards approach.

2 The Safeguards Glossary of IAEA, 2001 Edition, defines an MBA, according to IAEA INFCIRC-153, as “an area in or outside of a facility such that: (a) The quantity of nuclear material in each transfer into or out of each ‘material balance area’ can be determined; and (b) The physical inventory of nuclear material in each ‘material balance area’ can be determined when necessary, in accordance with specified procedures, in order that the material balance for Agency safeguards purposes can be established”

- Identifying and evaluating all potential sequences of events (Pathway Analysis) that could result in the proliferant state succeeding in its objective of concealed diversion of 1 SQ.

The outcomes of the pathways are expressed in terms of PR measures:

- Proliferation Technical Difficulty;
- Proliferation Cost;
- Proliferation Time;
- Fissile Material Type;
- Detection Probability;
- Detection Resources Efficiency.

The major differences among the three evaluation approaches used in the demonstration study were in the implementation of the evaluation steps of the PR&PP framework. As a result the quantification of the PR measures undergoes a different process for the three evaluation approaches. The qualitative approach relies on expert judgment to assign values and uncertainty ranges to PR measures, using a set of metric scales for the PR measures[2]. Not all measures can be directly estimated with an event tree/fault tree approach (e.g., proliferation time). In that case auxiliary methods are needed to quantify the remaining measures. The Markov approach also cannot quantify all PR measures directly without resort to auxiliary calculations; one example is detection resources efficiency.

3.3.1 The Qualitative Evaluation

All quantitative PR&PP evaluations necessarily start with a qualitative one. Qualitative evaluation provides a necessary tool to structure the analysis problem and prioritize areas for detailed study using quantitative methods. In applying a qualitative evaluation approach to the demonstration slice, the emphasis is on the importance of using a structured approach in performing the evaluation steps; the study carried out also presents guidelines and examples for this process.

Steps involved in a qualitative evaluation follow closely the PR&PP framework and include:

1. Selection of system elements and of the threat for study. This step aims primarily:

   • To gather design information, this may range from conceptual to detailed;

   • To specify objectives, capabilities, and strategies (can be stylized descriptions) for the selected threat;

2. Use expert judgment to survey system elements and threat strategies to identify a small number of “representative” pathways for analysis;
3. Performing a qualitative analysis using expert judgment to estimate measure values for selected pathways:

   • Consider acquisition and processing separately, then aggregate measures;

   • Use check lists, as available, to assure that important system attributes have been considered;

   • Display results in tabular form, showing uncertainty intervals as ranges for the metrics for each measure;

4. Use insights from (3) to confirm initial selection in (2);
5. Discuss insights and conclusions from analysis.

In the qualitative approach, the PR measures are evaluated by expert judgment. The scales for the PR measures estimates range from “Very High” (VH) PR, making the pathway less attractive to a proliferant state, to “Very Low” (VL) PR, making the pathway more attractive to the state. Uncertainty bands are estimated for each measure. A qualitative uncertainty band is used to reflect the state of early phase of design and analysis when incomplete information is available. A narrower uncertainty band, the residual uncertainty band, is used to represent potential uncertainty range after detailed design and analysis are completed.

A total of four diversion scenarios were considered in the qualitative evaluation of the fuel cycle facility. The first three diversion scenarios assumed protracted diversion by the host state from different unit operations within MBA-2. The diversions were from:

- The spent fuel chopping operation;
- The TRU extraction operation;
- The product preparation operation.

The fourth diversion scenario was for a distributed diversion strategy, i.e., removing material clandestinely from many parts of the pyro-processing facility. This pathway involves protracted, concealed diversion of material from the facility, aimed at acquiring TRU without detection by safeguards. This TRU is subsequently processed in a separate, concealed facility to produce plutonium metal for fabrication into nuclear explosives.

3 Example scales for the estimates of each of the PR measures are reported in [8].

The PR measures for the four diversion scenarios were evaluated by two experts using the qualitative approach. The first three scenarios of concentrated diversion were evaluated by one expert (LLNL) and the distributed diversion scenario was evaluated independently by the other expert (UCB). In evaluating Technical Difficulty (TD), the experts considered both intrinsic and extrinsic barriers, workers' skills, and industrial capability of the host state. The TD measure ranged from "Low" to "Medium-High" for the four diversion scenarios. Proliferation time (PT) depends on the diversion rate and the need to have access to clandestine processing facilities. PT was judged to over around "Medium", roughly 1 to 5 years. Proliferation cost (PC) can vary from "Low to High" depending on the type and rate required to process the diverted material in clandestine facilities. Fissile Material Type (MT) is a measure to be estimated on the material available at the end of the processing stage and, therefore, is estimated once for the overall pathway. A qualitative measure for MT is the attractiveness or usability of the material for weapons. For different blends of TRU coming out of the pyro-processing facility MT was judged to range from Medium to Medium-High. Considering the various safeguards approaches available to detect diversion and the type of operation associated with each diversion point, Detection Probability (DP) for the four scenarios was ranged from "Low to High". Detection Resources Efficiency (DE) is evaluated against IAEA inspector efforts for similar facilities. Based on estimated effort for the pyro-processing facility (scaled from effort for PUREX plants) DE was judged to be "Low to Very Low".

3.3.2 The Logic Trees Based Evaluations

Two separate but complementary tree-based evaluations were taken by teams from PNNL and MIT in this methodology demonstration. The first (by PNNL) used fault trees to model potential failure of an attempt at diversion, the second (by MIT) used success trees to model the likelihood of success of such an attempt. The two teams worked closely together to ensure that the basis for the two separate analyses was comparable. Either approach can be used; the choice depends on the problem being studied and the preference of the analyst. The fault/success tree study carried out by PNNL only evaluated the material acquisition phase from the fuel cycle facility slice. Due to funding and time constraints, the study was not completed, but sufficient progress was made to demonstrate the utility of the methodology.

The application of the event tree/fault tree approach to the evaluation of proliferation resistance is a three step process. The first step is to do a threat analysis and identify potential diversion points or pathways by way of an event tree analysis. Event trees are inductive logic models used to identify sequences of events that lead to particular outcomes, both desirable and undesirable. The second step is to define a diversion strategy and identify corresponding safeguards detection methods to be overcome by the proliferators. A fault tree structure is constructed to model the failure of each safeguards detection method to detect potential diversion attempts within a specified time. Fault trees are deductive logic models constructed to define all possible failure combinations which lead to a particular event, for instance the failure of a specific system to function as required to perform a vital mission. The third and final step is to evaluate the likelihood of detection of the proliferation attempt and to calculate the six proliferation resistance measures. The solution of the event tree/fault tree models is a collection of what are called minimal cut-sets. Minimal cut-sets are combinations of occurrences (basic events) along a pathway that allow the pathway consequence to occur. Each minimal cut-set has a probability based on the concatenation of the likelihood of each base event or occurrence in the minimal cut-set.

Based on analyzing the operation of the pyro-processing facility and the daily material flow through the demonstration slice it was decided to evaluate the likelihood of proliferation success for a protracted strategy using the External Uranium Container in the Product Preparation Station in MBA-2. Since the container had been described to have a capacity to carry 3.17 kg of uranium it was assumed in the study that the diversion strategy would have been to perform three diversions in one year. Each diversion attempt will consist of 3.17 kg of TRU metal from the Product Preparation station. A fault tree was prepared for each diversion attempt. For this pilot study screening values were used to determine the likelihood of each basic event (occurrences along a pathway). The enablers (actions taken by the facility owner to defeat safeguards) were given a probability of 1.0; the failures of safeguards personnel were estimated based on human failure probabilities and assigned a screening value of either 2.5E-1 or 5.0E-1, and the failures of the instruments to detect the diversion were also set at screening values of either 2.5E-1 or 5.0E-1. If the analysis had proceeded further, more detailed values would have been developed.

It should be noted that minimal cut-sets which can be obtained by the approach are to be considered "raw data" by analysts, and they must be reviewed for validity and plausibility. If this study had been completed, the minimal cut-sets would have been grouped to represent specific diversion scenarios for each pathway, and these scenarios would have been developed to provide:

- Proliferation Detection Probability;
- Proliferation Pathway Technical Difficulty based on minimal cut-set evaluation;
- Proliferation Pathway Resources Efficiency would have been based on minimal cut-set evaluation.

Material Type would have been developed based on the condition of the material being diverted. In the case of the pathway being analyzed, since the diverted material was TRU metal that had gone through the electro-refining process, the material type would have had value between reactor grade plutonium and deep burn grade plutonium. Proliferation Time in this model was assumed to be one year; the Detection Resources Efficiency would have been determined by an evaluation of the cost of the safeguards involved and the personnel costs required to support them.

A complementary tree-based method, the success tree approach was also used by MIT to implement the pathway analysis of the PR evaluation methodology in this study. The diversion scenario of weapons material (successful diversion of Pu was assumed in this study) was modeled by discrete steps at MBA-2 of the FCF. The diversion scenario was divided into four steps: (1) lying about the amount of the input material, (2) moving Pu to holdup inside MBA-2, (3) moving Pu from holdup to baskets going out of the 'Product Prep' stage, and finally (4) diverting Pu via the Product Prep basket.

A multi-step diversion was assumed in the success tree analysis to get 1 SQ of Pu, because 1 SQ was judged too much to be diverted safely in a single attempt, and without being detected. 10 diversion attempts to obtain 1 SQ and 3 attempts to obtain 1 SQ were modeled respectively as examples. Dependencies between each step diversion were assumed. For example, the success of the nth diversion step depended on the success of the (n-1)st diversion step. This treatment is based upon the inference that manipulations of surveillance equipment such as surveillance cameras are needed in order to fool the cameras, for example, in order to move Pu inside MBA-2 without being detected. If this manipulating or fooling of the cameras is repeated in order to get 1 SQ of Pu, then the possibility of this manipulation being detected should be increased remarkably because of multiple occurrences. On the other hand, safeguards inspectors might become complacent after seeing repeated occurrences and the possibility of being detected could actually go down for multi-step diversion.

Dependencies were assumed for the activities in this success tree model as follows:

1. Lie -> No dependency between the diversion steps is assumed;
2. Move Pu to holdup -> Fooling cameras inside MBA-2 is needed. -> Dependency exists;
3. Move Pu from holdup -> Fooling cameras inside MBA-2 is needed. -> Dependency exists; and
4. Divert Pu via Product Prep outgoing baskets. -> Fooling cameras monitoring the baskets is needed. -> Dependency exists.

Sensitivity analysis done on the dependency of failure probability on previous diversion attempts found that the dependency had negligible effect on the final result, the estimate of the proliferation success probability.

3.3.3 The Markov Evaluation

The Markov model approach developed by BNL is an implementation of the pathway analysis, a key element of the PR&PP evaluation methodology. The Markov chain method has the capability to account for some of the dynamic features of proliferation, namely the large number of uncertainties, the unpredictability of human performance, and the effect of changing conditions with time. In the Markov model approach the normal flow of nuclear material in the fuel cycle (front and back ends) is accounted for and the abnormal flow due to proliferation activities is modeled as a time dependent random process. Major activity modules in the fuel cycle (e.g., a physical process in a recycle facility) and the proliferation pathway (e.g., the act of diversion from a declared facility) are represented by a number of discrete stages in the Markov chain. In addition, absorbing states (terminal stages) are used to represent the effective termination of the proliferation activity due to intrinsic (e.g., radiation) or extrinsic (e.g., international safeguards) barriers. The transition between stages is treated as a random process with a given probability distribution. The transition rate is characterized by time parameters that are based on physical processes. For example, the transition time from one process to the next in the fuel cycle facility is derived from the rate of material flow in the actual recycling process. In modeling safeguards the rate of detecting an anomaly is derived from the frequency of executing safeguards approaches. The realization of the random process at each stage is then a random variable and the expected values of these random variables constitute the state (solution) space. Thus, by mapping the stages of a proliferation scenario into a Markov chain model, the likelihood of all possible outcomes can be determined systematically.

The Markov model approach is highly adaptable and scaleable. It had been applied previously to evaluate the PR of an advanced light water reactor in a misuse scenario and in scenarios that involve diversion from the front and back ends of a once through fuel cycle. In the demonstration study the Markov approach was applied to evaluate the PR measures for a portion of the fuel cycle facility of the ESFR system. As a result of an assessment of the system response to the threats, probabilistic PR measures such as detection probability and failure probability (due to intrinsic barriers and technical difficulties) are calculated directly by the Markov model. Since the model also represents the pathways for which the time and cost parameters can be defined, the path with least cost or the least time can also be calculated using the same model. In addition, the type of diverted material (pathway dependent), proliferation cost, and detection resources can all be estimated based on the pathway analysis.

The demonstration study considered the case of a host-state diversion and the aspiration was to obtain 1 SQ equivalent of TRU in one year. The Markov model was applied to evaluate two diversion strategies and two tactics. The strategies were concentrated (from one place) and distributed diversion, and the tactics were abrupt and protracted diversion. Safeguards approaches considered in the Markov model include audit of nuclear material accounting reports and records, material verification, surveillance and monitoring, and containment. The level of complexity of the Markov model for PR evaluation increased over the different applications. New features have been added by introducing new parameters that influence the values of the transition times. In particular, the following new features were introduced in the demonstration slice study:

1. An effective detection rate was introduced to account for the implementation of multiple safeguards approaches at a given strategic point. Uncertainties related to the accuracy/sensitivity of measurement methods were considered in the model. The potential for false alarm due to oversensitivity of safeguards equipment was accounted for by a new parameter, the confidence level of diversion confirmation.
2. A new state called “diversion failure” was introduced to reflect the inability of the proliferators to overcome the intrinsic barriers originated from either the design of the facility or the properties of the material in the facility.
3. Concealment to defeat or degrade the performance of safeguards was recognized in the Markov model. It was considered as a tactic of the proliferators and it was assumed to prompt more immediate and concerted responses from the safeguards inspectors.
4. Human performance in the safeguards area was incorporated in the Markov model by modifying the time parameter of a human action (e.g., the transition time associated with an inspection) with a success factor that takes into consideration the probability of human errors.

Example cases have been done to demonstrate the effects of modeling features on the PR measures for the fuel facility of the ESFR system. One of the safeguards approaches observed to have a positive impact on successful detection of diversion activities was the proper employment of surveillance cameras because they are able to detect an anomaly quickly. The effects of false alarms on detection probability, failure probability and success probability have been studied. With the presence of intrinsic barriers, a new absorbing state was introduced, diversion failure due to intrinsic barriers. This new absorbing state has several effects on the outcome of the proliferation activity. It reduces the probability of detection because there are two failure terminal states, being detected and diversion failure due to intrinsic barriers. The presence of intrinsic barriers was demonstrated to have the effects of decreasing the transition rate from declared facilities to clandestine facilities and also prolonging the proliferation time. Though the detection probability is lower with intrinsic barriers, the overall success probability for the proliferator is lower due to significant increase in diversion failure from intrinsic barriers. Concealment enables the proliferator more chance to divert material and also fail in the attempt due to intrinsic barriers. Concealment reduces the detection probability and increases the diversion failure probability and the success probability for the proliferator. The impact of human errors is similar to that of concealment. Results of the analysis indicated that human errors have the effects of lowering the detection probability slightly while enabling the proliferators to have more chance to divert material and fail for the same intrinsic barriers. The case of distributed diversion was compared with concentrated diversion for the demonstration slice. The results indicated a lower probability of detection and shorter proliferation time for the distributed case.

Three of the six PR measures were calculated directly by the Markov model and they are the detection probability (DP), proliferation technical difficulty (TD) and proliferation time (PT). Technical difficulty can occur in overcoming intrinsic barriers or in processing the diverted material. A metric PF, probability of technical failure, was used as quantitative realization of the measure for proliferation technical difficulty. The other three measures are derived based on the material type. Material type (MT) is indicated by an index that is based on the type of material at the acquisition stage. By assessing the physical, chemical, and isotopic properties of the diverted material the proliferation cost (PC) and resources required to detect the proliferation can then be evaluated. In the demonstration study, PC was evaluated according to the easiness of converting the diverted material to Pu metal, and so, it is dependent on MT (both isotopic composition and quantity). Detection resources efficiency (DE) is also connected to MT because more resources will be allocated to protect materials that are of interest and use to the proliferators.

The differences in detection probabilities were shown not to be very large among the facilities considered in the demonstration slice. Results from the Markov model suggested that among the 8 facilities, considered in the study, the U-product processing and the electro-refiner are the most proliferation resistant, while TRU extraction is the least proliferation resistant. The main reason for being the most or least proliferation resistant is in the attractiveness of the material (TRU salt versus TRU metal). The presence of significant intrinsic barriers in the material balance area (e.g., operation in hot cell) also has some bearing on the relative resistance to proliferation when comparing different facilities.

Sensitivity analysis for the time parameters for the Markov model has been performed. It was noted that variation in one parameter affects several measures. This behavior is consistent with the fact that the PR measures are not independent of each other. Results of the sensitivity analysis showed that the overall impact of intrinsic barriers is significant for PT and PF and minor for the detection probability. For the designer of a facility, once the recycling process is determined, there is little that can be done to alter the MT of the material in the facility. At the assumed baseline level of detection capability, the overall benefit of increasing safeguards is not obvious from the sensitivity analysis. There seems to be an indication of diminishing return in the cost related to increasing safeguards (increased capability and frequency). Evaluating the impact of diversion rate on proliferation resistance is less conclusive because of uncertainties and assumptions in the safeguards approaches. While decreasing the diversion rate will exploit the uncertainties of the safeguards such that the chances of being detected will decrease, it will also prolong the diversion process and increases the detection probabilities and proliferation time. Sensitivity analyses have been performed to evaluate the impact of increased and decreased diversion rates on the PR measures.

3.4 Accomplishments of the Demonstration Case Study

The demonstration study showed that the methodology developed by the GIF PR&PP Expert Group provides a structured framework for comprehensive evaluation of the PR for a nuclear system.

Results from the three different forms of evaluation (Qualitative, Logic trees, Markov Model) are consistent, although the level of detail and focus of each analysis differed. These differences are due in part to the focus of the analysts and in part to the choice of analytic methods.

In summary, qualitative evaluation is well suited to coarse-level evaluation of a nuclear system where detailed information about the system is not available and results are required quickly. More detailed analysis using methods such as logic trees or Markov modeling are appropriate where more information about the nuclear system is available and where more accurate results (lower uncertainty) are required.

Further analyses on diverse nuclear systems with diverse analytic objectives (e.g., quick study for decision maker; detailed study for designer) should be conducted to gain further insight into appropriate application of analytical methods to PR&PP analysis and to establish a baseline standard/norm for such analysis.

4. Conclusions, Ongoing and Future Activities

The GIF roadmap recommended the development of a comprehensive methodology for evaluation of the proliferation resistance and physical protection robustness of Generation IV Nuclear Energy Systems. An expert group was set up within the GIF initiative for the development of the methodology. For the purpose of the development of the methodology, a hypothetic Generation IV Nuclear Energy System, named Example Sodium Fast Reactor has been set up by ANL. A first development case study was carried out resulting in the release of the revision 5 of the methodology report that is available on the internet for undisclosed distribution.

This paper has summarized the PR&PP methodology paradigm and framework as described in the revision 5 methodology report. The paper has provided also an overview of the Demonstration Case Study, carried out by a task group, in 2005-2006 on a portion of the fuel cycle facility of the ESFR NES, in order to test different approaches for the practical implementation of the methodology.

In the year 2007, the PR&PP Working Group initiated a broader fuel cycle study, involving the evaluation of the full ESFR NES and accounting for different threats. The new demonstration case study, started in 2007 and executed over 2008 aims primarily to show how the PR&PP methodology can provide useful feedbacks to designers at various levels of details, including pre-conceptual design. In particular, the application of the PR&PP methodology has to show that it is possible to evaluate variations in the design and to generate meaningful results.

The study analyses the response of the ESFR nuclear energy system to different proliferation and theft threat strategies. The PR threat strategies considered comprise concealed diversion of material, concealed misuse of the facility and breakout strategies involving overt diversion of material or misuse of the facility. The PP threats will be mainly thefts of weapon-usable material. For the PR threat categories, the actor is the host state with defined objectives and capabilities. For the PP threats the actor is a sub-national group, also with defined objectives and capabilities.

Task groups were created to address each of the four PR&PP threat strategies. The first year of work involved the evaluation of the full ESFR NES in the baseline configuration, corresponding to an actinide burner with conversion ratio (CR) of 0.64 and needing an external feed of U and TRU made of LWR spent fuel elements to be processed on the site. Design variations are being considered in the second year of work. The first design variation is in line with the original baseline design: it is a burner configuration (TRU conversion ratio slightly higher, i.e., 0.73), and foresees also a U and TRU feed made of LWR spent fuel elements. The second design variation considers the reactors in deep burner configuration, with a very low TRU conversion ratio of 0.22 implying a much larger amount of LWR spent fuel elements to be processed on the site. The remaining two core configurations correspond to a self feeding reactor, with CR 1, and to a breeder core (CR 1.2) with both inner and external blanket.

Preliminary results for the evaluation of the baseline design for the diversion [11] and misuse strategies [12] have been presented at the 49th INMM Annual Meeting held in Nashville in July 2008, together with an application of the Markov modeling approach [13] to the diversion strategy [14].

Among the methodological aspects to be further investigated in the future, there is the need to develop systematic approaches for expert elicitation and to develop an approach to uncertainty/sensitivity analysis. Another area for further work will be to refine the methods for display and use of results. Future updates of the methodology will be based on insights of the work done on the demonstration study and on the new study carried out in 2007-2008.

A new term of reference has been issued for the PR&PP renamed Working Group. The new term of reference emphasizes the results reached and stresses the synergy and the collaboration between the PR&PP group and the designers of Generation IV NES. To this aim, all representatives of Generation IV Systems Steering Committees have been invited to the PR&PP plenary meeting held in January 2008 at the CEA Marcoule Centre and a joint workshop has been hosted at Brookhaven National Laboratory in May 2008. One of the first joint activities will be the production of white papers, one for each Generation IV reference design, emphasizing the PR&PP issues and features according to the current design choices and levels. Joint case studies will be carried out in the course of the year 2009.

For the establishment of a PR&PP culture it will be very important to promote the use of the methodology among potential users mainly among designers of advanced nuclear energy systems. By so doing the PR and PP concepts will be considered since the early design phases by accomplishing a PR and PP robustness by design.

Acknowledgements

The full list of the PR&PP group members is herewith annexed, as reported in revision 5 of the PR&PP methodology report [8], together with an updated list of the contributors to the work and to the development case study. For the full list of the previous group members and of the previous contributors to the work see [8]. In addition, all those who provided input and comments for the advancement of the methodology, on the development case study and on the demonstration case study are thankfully acknowledged.

Annex: Proliferation Resistance and Physical Protection Evaluation Methodology Expert Group, as in [8]

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7. References

[1] US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, *A Technology Roadmap for Generation IV Nuclear Energy Systems*, GIF002-00, December 2002.

[2] Peterson, P., Roglans, J., Bari, R., Assessment methodology development for proliferation resistance and physical protection of Generation IV systems. *Proceedings of ANS International Conference – Global 2003, New Orleans, LA*, November 13-18, 2003.

[3] Nishimura, R., Peterson, P., Roglans, J., Bari, R., Kalenchuk, D., Development Of A Methodology To Assess Proliferation Resistance And Physical Protection For Generation IV Systems, Americas Nuclear Energy Symposium, *ANES* 2004, Miami Beach, Florida, October 3-6, 2004.

[4] PR&PP Expert Group and Other Contributors, Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems, Presented by G.G.M. Cojazzi, *27th ESARDA Annual Meeting Symposium on "Safeguards and Nuclear Material Management", IEE Savoy Place,* London, May 10-12, 2005, EUR 21674.

[5] Nishimura, R., Roglans, J., Bari, R., Peterson P., Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems, *INMM 46th Annual Meeting,* July 10-14, 2005, Phoenix, Arizona, U.S.A.

[6] PR&PP Expert Group and Other Contributors, Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems, *Proceedings ESReDA 29th Seminar, Systems Analysis for a More Secure World. Application of System Analysis and RAMS to Security of Complex Systems.* October 25-26, 2005, JRC, IPSC, Ispra, EUR 22112 EN.

[7] Bari, R., Nishimura, R., Peterson, P., Roglans, J., Bjornard, T., Cazalet, J., Cojazzi*, G.G.M., Delaune, P., Golay, M., Haas, E., Rochau, G., Renda, G., Senzaki, M., Therios, I., Zentner, M., Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems: an Overview. *Eight International Conference on Probabilistic Safety Assessment and Management (PSAM 8),* New Orleans, USA, May 14-19, 2006.

[8] PR&PP Expert Group and Other Contributors, *Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems,* Revision 5, November 30, 2006 GIF/PRPPWG/2006/005. Available http://www.gen-4.org/Technology/horizontal/PRPPEM.pdf

[9] PR&PP Expert Group and Other Contributors, *Addendum to the Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems Technical Addendum* to Revision 5, January 31, 2007 GIF/PRPPWG/2006/005-A. To be made available at http://www.gen-4.org/Technology/horizontal/

[10] Bari, R. et. Al., PR&PP *Evaluation Methodology Demonstration Study Interim Report*, January 31, 2007, Available on request.

[11] Zentner, M.D., Coles, G.A., Therios, I.U., A Qualitative Assessment of Diversion Scenarios for an Example Sodium Fast Reactor Using the Gen IV PR&PP Methodology, Presented at 49th INMM Annual Meeting, Nashville, USA, July 14-17, 2008.

[12] Cojazzi, G.G.M., Renda, G., Choi, J-S., Applying the GIF PR&PP Methodology for a qualitative analysis of a misuse scenario in a notional Gen IV Example Sodium Fast Reactor , Presented at *49th INMM Annual Meeting,* Nashville, USA, July 14-17, 2008.

[13] Yue, M., Cheng, L., and Bari, R.A., Markov Based Approach for Proliferation Resistance Assessment of Nuclear Energy Systems, *Nuclear Technology*, Vol: 162, 88, pp. 26-44, 2008.

[14] Yue, M., Cheng, L.Y., Bari, R., Markov Model Application to Proliferation Risk Reduction of an Advanced Nuclear System, Presented at 49th *INMM Annual Meeting,* Nashville, USA, July 14-17, 2008.

The cover of the Bulletin is inspired by Metropolis, a 1925 landmark movie by Fritz Lang.
The cover features future Nuclear Energy Systems against a historical background,
the entire scene immersed in an artificial world.

The collage is based on drawings courtesy of the Generation IV International Forum,
a 1964 image of the 'Ispra 1' research reactor (courtesy of the Joint Research Centre, Ispra)
and a picture of a building in the Vulcania museum, France.

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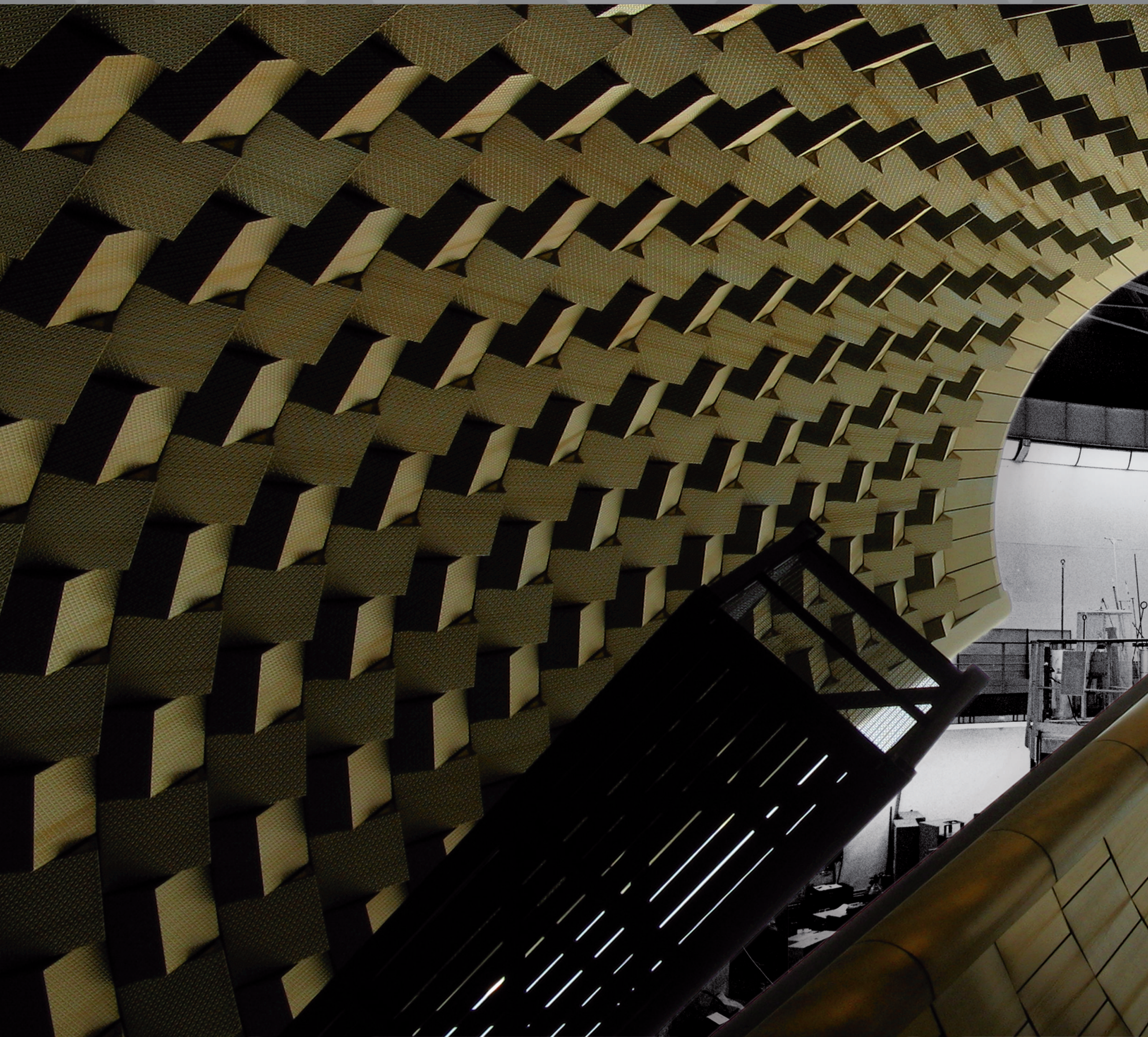

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