IAEA Nuclear Energy Series

No. NP-T-1.11

# Options to Enhance Proliferation Resistance of Innovative Small and Medium Sized Reactors

Basic Principles

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# OPTIONS TO ENHANCE PROLIFERATION RESISTANCE OF INNOVATIVE SMALL AND MEDIUM SIZED REACTORS

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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world''.

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# FOREWORD

One of the IAEA's statutory objectives is to "seek to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world." One way this objective is achieved is through the publication of a range of technical series. Two of these are the IAEA Nuclear Energy Series and the IAEA Safety Standards Series.

According to Article III.A.6 of the IAEA Statute, the safety standards establish "standards of safety for protection of health and minimization of danger to life and property". The safety standards include the Safety Fundamentals, Safety Requirements and Safety Guides. These standards are written primarily in a regulatory style, and are binding on the IAEA for its own programmes. The principal users are the regulatory bodies in Member States and other national authorities.

The IAEA Nuclear Energy Series comprises reports designed to encourage and assist R&D on, and application of, nuclear energy for peaceful uses. This includes practical examples to be used by owners and operators of utilities in Member States, implementing organizations, academia, and government officials, among others. This information is presented in guides, reports on technology status and advances, and best practices for peaceful uses of nuclear energy based on inputs from international experts. The IAEA Nuclear Energy Series complements the IAEA Safety Standards Series.

There has been renewed interest in Member States in the development and deployment of small and medium sized reactors (SMRs). Innovative SMRs in many cases aim to serve regions that are unable to benefit from large nuclear power plant deployment. These may be off-grid areas difficult to access, remote islands or sparsely populated regions with small electric grids. Alternatively, the staggered building of SMRs could offer attractive investment opportunities for States where investment capabilities and the overall energy demand are relatively small. The specific characteristics of SMRs, such as the remote application of innovative SMRs, the use of several small modular reactors on one site and the mode of shipment of fully assembled reactors, may have an impact on the proliferation resistance of SMRs and their associated fuel cycles. The IAEA's International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) has developed a methodology for the assessment of proliferation resistance and physical protection (PR&PP) of innovative nuclear energy systems (INSs), with the potential to include SMRs. Simultaneously, the Generation IV International Forum (GIF) has developed and published an alternative methodology to assess PR&PP of INSs, including nuclear reactors and the associated fuel cycles. Application of the available assessment methodologies, starting from early design stages of nuclear power plants with innovative SMRs and associated fuel cycles, could assist designers in defining a consistent strategy regarding the incorporation of certain intrinsic proliferation resistance and security features in innovative SMR designs to ensure that the necessary extrinsic measures are adequate. This publication provides options for the application of the assessment methodologies.

This publication is the result of collaboration among 12 Member States. R.A. Bari and M.D. Zentner (United States of America) were the IAEA subject matter experts for this publication and developed the initial draft. The IAEA is grateful to all those who assisted in the drafting and review of this publication.

Special thanks are due to F. Depisch, A. Korinny, J. Sprinkle and V.V. Kuznetsov. The IAEA officer responsible for this publication was M.H. Subki of the Division of Nuclear Power.

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# 1. INTRODUCTION

## 1.1. BACKGROUND

Several dozen innovative small and medium sized reactor (SMR) concepts and designs are being developed within national and international R&D programmes, involving both developed and developing countries. According to IAEA classification, small reactors are reactors with an equivalent electric power of less than 300 MW(e) and medium sized reactors are reactors with an equivalent electric power of between 300 and 700 MW(e). SMRs can provide an attractive and affordable nuclear power option for many developing countries with small electric grids, insufficient infrastructure and limited investment capability, or when energy production flexibility is required. SMRs are also of particular interest for cogeneration and many advanced future process heat applications.

Innovative SMRs in many cases aim to serve the customers unable to benefit from large nuclear power plant deployment. Such customers may be located in off-grid areas with difficult access, on remote islands, or in certain States or regions where electric grids and populations are small. Alternatively, the staggered building of SMRs could offer attractive investment opportunities for States where investment capabilities and the overall demand of energy are relatively small. SMRs may someday become part of the energy systems of both developed and developing countries, offering smooth and timely additional capacity and a variety of non-electrical products.

The potential remote applications of innovative SMRs, the use of several small modular reactors together on one site and the mode of shipment of fully assembled reactors may have an impact on the proliferation potential of such reactors and associated nuclear fuel cycles. Therefore, the proliferation resistance features of the reactors need to be evaluated, and enhanced if necessary, to provide cost effective and reliable safeguards measures.

In 2006 and 2007, the IAEA published two status reports on innovative SMR designs that highlighted the technical features of the reactors and their associated fuel cycles that could contribute to enhanced proliferation resistance and physical protection (PR&PP) [1, 2]. In recent years, the IAEA's International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) has developed a methodology for the assessment of PR&PP of innovative nuclear energy systems (INSs) that may include SMRs. Over the same period, the Generation IV International Forum (GIF) has developed and published an alternative methodology to assess PR&PP of INSs, including nuclear reactors and the associated fuel cycles.

Basic tools in the framework are the application of proliferation assessment methodologies designed to provide information to designers and operators on how to improve the proliferation resistance and safeguardability[1] of SMRs. Other efforts to develop the concepts of safeguards by design (SBD) and safeguardability analysis are under way in a number of organizations. The merits of using a structured approach in analysing safeguardability are broadly recognized. There have been new advances in how to do this, and SBD assessment techniques are evolving and improving. Specifically, the approach is based on the GIF PR&PP assessment methodology and the INPRO process of proliferation resistance assessment. The IAEA considers how these particular approaches might be adapted to support SBD and facility safeguardability analysis (FSA) as important elements of proliferation resistance.

Application of the available assessment methodologies, starting from early design stages of nuclear power plants with innovative SMRs and associated nuclear fuel cycles, could assist designers in defining a consistent strategy regarding incorporation of certain intrinsic proliferation resistance and security features into innovative SMR designs to ensure that the necessary extrinsic measures are adequate and cost effective.

## 1.2. OBJECTIVE

The overall objective of this publication is to assist existing and potential interested stakeholders in defining a consistent strategy regarding the incorporation of intrinsic proliferation resistance and safeguardability in the designs of innovative SMRs.

[1] Safeguardability is a function of how well the facility can be effectively and efficiently safeguarded.

The specific objective of this publication is to provide a framework for the application of the assessment methodologies to evaluate the proliferation resistance and safeguardability of innovative SMRs introduced in their designs — beginning in the early stages and as the designs progress.

### 1.3. SCOPE

This publication provides options for those applying proliferation resistance assessment methodologies to support the development of SMRs formulated in the INPRO and GIF frameworks. This publication presents the views of contributors on one such approach, and is expected to interest both innovative SMR designers and potential users of INSs with SMRs.

### 1.4. STRUCTURE

The main sections of this publication raise questions about specific considerations for SMR proliferation resistance and safeguardability. They outline the framework analysis by summarizing both GIF and INPRO methodology, describe the proliferation resistance assessment and SBD approach, present the current implementation of PR&PP in innovative SMRs, include findings from previous studies and derive conclusions.

Appendices I and II include information on an example of a procedure to assist FSA in support of SBD and a template listing required proliferation resistance related design information.

Annexes I and II provide an overview of SMR design and development activities and Member States' perspectives.

## 2. SPECIFIC CONSIDERATIONS ABOUT PROLIFERATION RESISTANCE AND SAFEGUARDABILITY IN SMALL AND MEDIUM SIZED REACTORS

SMRs have innovative design features and technologies that may require new tools and measures for safeguards. For example, existing accountancy tools and measures may need to be modified or further developed for reactors using non-conventional fuel types. In addition, reactors using the $^{233}$U–thorium fuel cycle may present measurement or other safeguards challenges which are yet to be identified. New fuel loading schemes may pose fresh challenges, as reactor cores with extremely long lifetimes may require innovative surveillance tools and measures. Long life sealed core replacements may also present novel accountancy challenges.

### 2.1. STATUS OF SMALL AND MEDIUM SIZED REACTOR DEVELOPMENT PROGRAMMES

The results of various IAEA technical meetings and workshops on SMRs show that multiple module power plants with SMRs may offer energy production flexibility that energy market deregulation might call for in the future in many States. SMRs are also of particular interest for cogeneration and many advanced future process heat applications. In 2014, 13 SMRs will be under construction in six Member States: Argentina, China, India, Pakistan, the Russian Federation and Slovakia. R&D is being conducted on approximately 45 innovative SMR concepts for electricity generation, process heat production, desalination, hydrogen generation and other applications. SMRs are under development for all principal reactor lines:

(a) Light water reactors (LWRs);
(b) Heavy water reactors (HWRs);

(c) Gas cooled reactors (GCRs);
(d) Liquid metal cooled reactors (LMCRs).

#### 2.1.1. Light water reactors

Small and medium sized LWRs are under development in Argentina, Brazil, France, Japan, the Republic of Korea, the Russian Federation and the United States of America.

The Central Argentina de Elementos Modulares (CAREM) reactor, a small, integral type pressurized LWR design with all primary components located inside the reactor vessel and an electrical output of 150–300 MW(e), is under development in Argentina. Construction of a 27 MW(e) CAREM prototype plant started in December 2013.

The fixed bed nuclear reactor (FBNR) is a Brazilian conceptual design that does not require on-site refuelling.

The Flexblue, under development in France, is a small and transportable modular design of 160 MW(e). Operated on the seabed, this water cooled reactor uses naval, offshore and passive nuclear technologies to take advantage of the sea as the infinite and permanently available heat sink.

In Japan, a 350 MW(e) light water cooled reactor with an integral primary system called the integrated modular water reactor has been developed. Validation testing, research and development for components and design methods and basic design development are required before licensing.

The system integrated modular advanced reactor (SMART) design, from the Republic of Korea, has a thermal capacity of 330 MW(th). It received standard design approval from the State's nuclear regulatory authority, the Nuclear Safety and Security Commission, in July 2012.

In the Russian Federation, the following light water SMR designs are under development. The ABV-6M, with an electrical output of 8.6 MW(e), is a steam generating nuclear power plant with an integral pressurized LWR with natural circulation of the primary coolant. It is in the detailed design stage. The RITM-200, designed to provide 8.6 MW(e), is an integral reactor with forced circulation for universal nuclear icebreakers (a type of ship). The VK-300 is a 250 MW(e) simplified water cooled and water moderated boiling water reactor (BWR) with natural circulation of coolant and passive systems. The VBER-300 is a 325 MW(e) pressurized water reactor (PWR) conceptual design that can serve as a power source for floating nuclear power plants. In addition, the Russian Federation is completing the construction of two units of the KLT-40S series, which are to be mounted on a barge and to be used for both electricity generation and process heat production. They are being prepared for startup commissioning. The VVER-300 and VVER-600 are also Russian SMRs in the final design stage and are based on the proven designs of the VVER-440, VVER-1000 and VVER-1200. Another Russian design is the UNITHERM — a PWR which is at the conceptual stage and is based on Research and Development Institute of Power Engineering (NIKIET) design experience in marine nuclear installations.

In the United States of America, four integral pressurized water SMRs are under development:

(1) Babcock and Wilcox's mPower;
(2) NuScale SMR;
(3) Westinghouse SMR;
(4) SMR-160.

The mPower design consists of four 180 MW(e) modules. Its design certification application is expected to be submitted in the fourth quarter of 2014. NuScale Power envisages a nuclear power plant made up of twelve 45 MW(e) modules, and it plans to apply for design certification to the United States Nuclear Regulatory Commission (NRC) in 2015. The Westinghouse SMR is a conceptual design with an electrical output of 225 MW(e). It incorporates passive safety systems and proven components of the AP1000. The SMR-160 is a 160 MW(e) design that does not require pumps to circulate the coolant.

#### 2.1.2. Heavy water reactors

Small and medium sized HWRs are deployed in Argentina, Canada, China, India, the Republic of Korea, Pakistan and Romania. Canada, for example, has developed and deployed the Canada deuterium–uranium (CANDU) reactor series, which offers various power ratings. The enhanced CANDU-6 (EC6) is a basic design with a gross electrical capacity of 740 MW(e) and is based on the CANDU-6 design.

In India, several HWRs in the range of 220–540 MW(e) and 540–700 MW(e) are under construction or in operation. The 304 MW(e) Advanced Heavy Water Reactor with Low Enriched Uranium and Thorium Mixed Oxide Fuel (AHWR300-LEU) design incorporates vertical pressure tubes, low enriched uranium (LEU) and thorium fuel, and passive safety features. It is in the basic design phase.

#### 2.1.3. Gas cooled reactors

Several GCR designs in the SMR classification have been developed globally. China is constructing the high temperature reactor pebble bed module (HTR-PM), which consists of two pebble bed modules that generate 200 MW(e), at the Shidaowan site. South Africa has developed the pebble bed modular reactor (PBMR) conceptual design, with an electrical output of 165 MW(e). The United States of America has two GCR designs: the energy multiplier module ($EM^2$) to generate 240 MW(e); and the gas turbine modular helium reactor (GT-MHR) to produce 150 MW(e). Japan has also developed and operated the High Temperature Engineering Test Reactor (HTTR), which produces a thermal power of 30 MW(th) for experimental purpose.

#### 2.1.4. Liquid metal cooled reactors

A number of small and medium sized LMCRs have been designed and in operation in China, India, Japan, the Russian Federation and the United States of America. The China Experimental Fast Reactor (CEFR), a sodium cooled 20 MW(e) experimental fast reactor with a fuel mixture of plutonium dioxide and uranium dioxide, has been in operation since 2011.

India is building the 500 MW(e) Prototype Fast Breeder Reactor (PFBR), which is expected to be commissioned in 2014.

Japan has developed the 4S (super safe, small and simple) reactor, a liquid sodium cooled fast reactor without on-site refuelling. It is designed to provide 10–50 MW(e) as a very small nuclear reactor design that can be located in a sealed, cylindrical vault underground, with the balance of plant (i.e. turbine, generator, condensers, and switchyard systems, buildings and structures) located above ground.

The Russian Federation’s 300 MW(e) design BREST-OD-300 is a lead cooled fast reactor that uses a two circuit heat transport system to deliver heat to a supercritical steam turbine. The Russian Federation has also developed, and plans to construct, several SVBR-100 units, which are small fast reactors with a lead–bismuth eutectic alloy as the coolant and a power output of 100 MW(e).

In the United States of America, the Power Reactor Innovative Small Module (PRISM) — a 155 MW(e) liquid metal cooled fast breeder reactor — has been developed and an attempt was made to apply for the NRC design certification review.

### 2.2. PROLIFERATION RESISTANCE AND SAFEGUARDABILITY

The notion of safeguardability was introduced early in the development of the PR&PP methodology owing to the challenge of computing the probability of detecting diversion or misuse for a design concept in its early stage of creation [3]. While the notion of safeguardability is not new, it can be seen as a bridge between intrinsic and extrinsic features of proliferation resistance. The PR&PP Working Group has more broadly defined safeguardability as the ease with which a system can be effectively and efficiently safeguarded [4, 5]. Finally, even if the INPRO initiative does not directly refer to safeguardability, this concept is present de facto behind indicators identified as relevant to detectability.

The challenge of addressing the problem of safeguardability of nuclear energy systems has existed since the IAEA was founded. Since the 1980s, at least two different approaches emerged: the first approach concerns developing new safeguarding techniques and equipment to enhance safeguards effectiveness and efficiency. The second approach concerns providing guidelines for designers of new systems to enhance system safeguardability during early design stages.

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. This approach was facilitated by the fact that systems did not change much from a safeguards point of view until the 1990s. Therefore, efforts focused on advancing detection equipment technologies.

However, with the recent advent of international initiatives to design the next generation of nuclear energy systems, the safeguards community has faced the new challenge of the need for new safeguards techniques capable of coping with major changes in the nuclear processes considered for new systems. Some recent studies have addressed the issue of safeguardability of processing facilities from both approaches: by varying safeguards measures and techniques; and also by varying the original process design to increase safeguardability.

In addition, with regard to competing aspects related to safeguardability, some attributes for analysing safeguardability are also relevant for proliferation resistance. However, the impact of these factors might be positive in one case and negative in the other. For example, limited accessibility to nuclear material because of radiological hazards 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 — although in the end one would expect the State, with the resources at its disposal, to overcome this barrier). This illustrates that designing a nuclear energy system that ensures non-proliferation is a challenging task, in which trade-offs on a number of important aspects will be required and optimization of these trade-offs will not always be straightforward.

### 2.3. QUESTIONS ON SAFEGUARDS (NON-PROLIFERATION) CONSIDERATIONS FOR SMALL AND MEDIUM SIZED REACTORS

It is assumed that the proliferation resistance volume of the INPRO methodology, in IAEA-TECDOC-1575 Rev. 1, Guidance for the Application of an Assessment Methodology for Innovative Nuclear Energy Systems: INPRO Manual — Proliferation Resistance [6], can be applied equally to large or small reactors and to their fuel cycles.

IAEA safeguards verify the operator's declarations about activities involving nuclear material. These declarations address the receipts, shipments, storage, movement and production of nuclear material. Inspection intensity depends on the material type (depleted uranium, high enriched uranium (HEU), LEU, natural uranium, plutonium or thorium) and whether the material is irradiated. In addition, the State level approach will take into consideration the technical capabilities of the State, including the possible existence of other nuclear activities (including commercial or university R&D) and the location of the facilities.

Safeguards considerations take into account differences in various aspects:

(a) Accessibility to the nuclear material;
(b) Whether the reactor facility is operated continuously;
(c) How the reactor facility is refuelled;
(d) Location and mobility of the reactor facility;
(e) Existence and locations of other nuclear facilities in the State.

For nuclear material that is normally not available to the operator, questions could arise if the vendor State delivers a sealed unit that operates until the vendor replaces it with a new one, perhaps ten years later, and if there is no storage capability for used fuel in the customer State nor equipment to handle used fuel. For example, can the reactor not only be 'sealed' by the IAEA and treated as an item, but can remote monitoring of the seal readily detect any attempt to open the reactor? If a safety concern arose, would the host State be dependent on the vendor to open the reactor? Otherwise, how could the host State respond to an event that would require opening or removing the reactor?

When comparing equivalent generating capacity (i.e. many SMRs with the same total capacity as one large LWR), issues would deal with whether SMRs will be collocated or separated at different sites. Additional questions would deal with their refuelling schemes and whether they would be different. For example, would refuelling of the SMRs be asynchronous and would there be separate used fuel storage for each module?

In the case of many small, isolated reactors in remote locations (e.g. the Arctic) compared to one large, centrally located reactor with an expansive electric grid, it would be necessary to look at inspector ease of access to the remote site as well as the possibility of building an electric grid. Will the SMRs be load dependent or considered baseload reactors? Will they be a stand alone, sole source of energy supply? Are SMRs offshore on floating barges and tied to State or regional electric grids? If so, then who has responsibility for accidents or malfunctions?

Another specific issue for SMRs would be to specify which minimum amount needs to be taken into consideration. If there is less than one significant quantity of nuclear material at a facility, different rules generally apply than for large quantities of nuclear material — typically, the inspection intensity in the detection of the diversion of one significant quantity is reduced for:

— Less than 8 kg of plutonium;
— Less than 25 kg of $^{235}$U in the form of HEU (>20%);
— Less than 75 kg of $^{235}$U in the form of LEU (<20%).

In this case, different issues might arise depending on the size of the fuel assemblies. Moreover, safeguards questions might arise about the operational impact following diversion of an assembly from the reactor core (will the core still have sufficient nuclear material to operate?), the possibility of a neutron flux distribution in a reasonably sized target area creating a $^{239}$Pu or $^{235}$U breeder environment, and whether small assemblies could be stacked on top of one another when storing the irradiated fuel after removal from the core.

The following considerations could apply to any new installation, including SMRs:

(1) Fuel leasing or supply arrangements that avoid on-site storage of fresh, used or mixed oxide (MOX) fuel would be of interest to safeguards.
(2) The isolation of the site or mobility of the reactor (sea or rail) might be a factor. Consideration is to be given to access issues for both the inspectorate and the adversary.
(3) With regard to remote monitoring, it is necessary to have discussions among the operator, State and the IAEA about small reactors which evaluate the potential of remote monitoring, including transmission of data off-site about the following issues:
    (i) What plans are there for the maintenance of safeguards equipment and safeguards relevant operator equipment?
    (ii) What plans are there for equipment shared between the inspectorate and the operator (e.g. cranes, surveillance and access controls)?
    (iii) Will the State system of accounting for and control of nuclear material (SSAC) be considered a viable partner in verification activities?
(4) To which extent can safeguards measures impact physical protection? Will there be a different approach to physical protection, and how might that affect the safeguards tools (e.g. for a small floating reactor)?
(5) How will the operator staff the reactor? Will hosting an inspection require additional staff for communication?
(6) Will the site or nearby sites have more or less ancillary equipment, such as hot cells, pin replacement capability, fuel storage and nuclear research activities?
(7) Another consideration for SMRs would be to examine whether containment features will be shared by multiple reactors (i.e. underground containment).

# 3. FRAMEWORK ANALYSIS

## 3.1. IMPROVING PROLIFERATION RESISTANCE THROUGH THE SAFEGUARDS BY DESIGN PROCESS

In the past, retrofitting IAEA safeguards into previously constructed facilities in States was costly and technically difficult [7]. Now, the IAEA is cooperating with the international community to implement the SBD process to help to ensure that safeguards are fully integrated into the design process of a nuclear facility — from initial planning through to design, construction, operation and decommissioning. The disciplined application of project management and systems engineering principles is important to ensure successful safeguards implementation that is fully integrated with other disciplines, such as process design and safety. Such an approach will utilize limited financial and human resources more efficiently and achieve non-proliferation objectives more effectively.

By submitting information on facility designs and concepts to the IAEA, designers and operators greatly aid the planning of facility specific safeguards [8, 9]. Accordingly, the IAEA has prepared high level functional safeguards guidance to which designers may refer during the development process [10]. Analyses of systems and safeguards performance (e.g. efficiency and effectiveness) can be applied to identify design features and safeguards tools and measures that minimize life cycle costs of safeguards implementation. Therefore, as new safeguards concepts move from R&D to implementation and the technology is deployed at new facilities, closer cooperation between designers and the IAEA is both possible and beneficial. Safeguards requirements, activities and interactions can be included in formal project planning and execution, thereby facilitating the effective and efficient implementation of safeguards.

As a result of modern, increasingly front loaded design practices, the ability to influence significant key design issues, such as process selection and plant layout, largely ends with the conceptual design step and certainly by the end of the preliminary design. Accordingly, SBD will be integrated as early as possible in the project management and design process for maximum effect. This could allow:

(a) Identification of potential safeguards issues (design changes that are important from a safeguards perspective);
(b) Evaluation of the cost effectiveness of alternate safeguards designs;
(c) Assessment of design innovations for situations that might make standard safeguards tools and measures less effective.

SBD can exhibit a different depth of approach for existing, evolutionary and innovative designs. In existing nuclear facility designs, the only practical and cost effective approach may well be to consider how best to install infrastructure for the traditional safeguards equipment. For evolutionary designs with modest changes from existing facilities, there might be opportunities to affect layout of facility subsystems, the chance to consider innovative sensors or remote monitoring, or opportunities to impact access to a number of nuclear material storage locations, as well as the chance to plan ahead for the necessary infrastructure and space for, and access to, safeguards equipment. For innovative systems, SBD may have much more flexibility when considering innovative safeguards equipment and concepts.

According to IAEA-TECDOC-936, Terms for Describing New, Advanced Nuclear Power Plants [11]:

> "An innovative design is an advanced design which incorporates radical conceptual changes in design approaches or system configuration in comparison with existing practice. Substantial R&D, feasibility tests, and a prototype or demonstration plant are probably required" [p. 9 of Ref. 11].

In these early stages, a designer will consider what is new about the design, and whether these differences present safeguards challenges that would complicate the use of anticipated IAEA safeguards tools and measures. A designer might also work with safeguards experts to investigate radical conceptual changes for safeguards that would also require R&D, testing and evaluation. A structured approach is to be used, and in some cases an elicitation process using subject matter experts may be required to perform this work. The project management team could then balance the advantages and disadvantages of the effects on operations, safeguards, safety and costs.

Similar to the groups responsible for these elements of facility design and development, the safeguards design team needs to work together through the facility design and development cycle (see Fig. 1).

(1) Pre-conceptual design: The earliest design stage where the level of detail may only describe the functional aspects of the facility and the proposed operations, but is still sufficient to conceptually design the safeguards system. A nuclear material accountancy process, likely based on material balance areas and key measurement points, would be defined and a preliminary diversion path analysis performed.

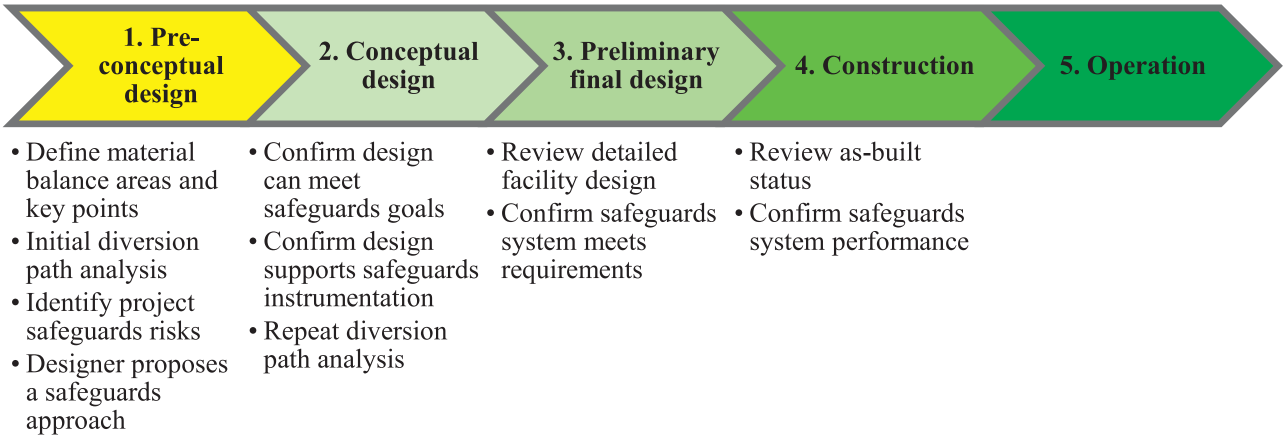


*FIG. 1. Safeguards by design implementation during facility design stages (adapted from Ref. [12]).*

(2) Conceptual design: Conceptual facility design details would be available, including proposed equipment and location and more detailed operations planning. This conceptual design facilitates identification of appropriate systems to enable the IAEA to meet safeguards goals and to ensure that the facility design accommodates the locations and space required for instrumentation and monitoring equipment, among other things. Specifications for the safeguards system would be defined, with proposed equipment locations, utility requirements and data transmission requirements. An assessment would be performed again to confirm that the safeguards requirements would continue to be met, and determine whether there were potential interferences between the safeguards, plant operation and safety systems. Confirmation that the interface with the safety systems does not adversely affect safety is required. Clarification of the data authentication and proprietary issues is important at this stage.

(3) Preliminary final design (i.e. design for construction): Where detailed facility design, dimensions, equipment and planned operations are specified, continued confirmation is required that the safeguards system will still meet specified safeguards requirements, with the minimum interference with plant operation and safety systems.

(4) Construction: Where the facility is being constructed according to the owner or operator's specification, the facility design may be changed during construction, and the safeguards system is to be assessed to ensure that changes in facility design or construction have not compromised safeguards system performance. Calibration and testing of the safeguards systems typically occurs at this stage with simulated materials or controlled radiation sources. The performance of the safeguards system is assessed again, based on the results of these tests, prior to hot operation of the facility (i.e. the introduction of radioactive material at the facility).

By applying SBD at each stage, the designers and operators, as well as the IAEA and regulatory bodies, would have the assurance that the safeguards system and facility design would:

— Advance IAEA objectives by enhancing the safeguardability of new facilities;
— Potentially reduce the time and cost associated with IAEA inspections;
— Potentially reduce costs and inspection schedule impact on facility owners and operators, thereby avoiding costly retrofits and improving cost control.

Implementation of SBD will enhance the ability to safeguard and protect new facilities while mitigating associated life cycle costs through early incorporation of safeguards into the project management and design tasks required for the development of new facilities.

## 3.2. SUMMARY OF APPROACHES FOR EVALUATING PROLIFERATION RESISTANCE AND SAFEGUARDABILITY FOR SMALL AND MEDIUM SIZED REACTORS

One approach to support the SBD process is to apply methodologies previously developed to study proliferation resistance. The most widely accepted methods for assessing proliferation resistance were produced by two international nuclear energy development programmes: the GIF and the INPRO. The GIF and INPRO methodologies were initially intended to be used by different communities, and they represent separate, but related, approaches for evaluating proliferation resistance. Specifically, the GIF methodology was a proliferation resistance analysis methodology developed for technology holders (i.e. the GIF community), whereas the IAEA developed the INPRO proliferation resistance assessment methodology for both users and technology holders. As the methodologies have matured, international efforts to coordinate their use have continued [13]. Several demonstration studies have been completed (see Refs [4, 14]) which illustrate that each approach has unique strengths and weaknesses that make the two complementary. Future proliferation resistance studies will likely integrate elements of each approach. The following summarizes the two methodologies and illustrates how they could be used together to advance the development of the SBD concept.

The GIF approach [5] uses an analytic framework to identify specific proliferation challenges, system responses and the resulting outcomes for a nuclear energy system. This approach identifies acquisition pathways and uses six proliferation resistance measures to quantify and to compare these pathways. Both quantitative and qualitative analysis may be used in the evaluation process.

The INPRO proliferation resistance assessment methodology [6] is structured in a manner similar to other INPRO assessment methodologies (e.g. economics and safety), with a hierarchical structure of top level basic principles, user requirements, indicators, evaluation parameters and acceptance limits.

> "The INPRO proliferation resistance approach identifies a *Basic Principle of Proliferation Resistance* and five *User Requirements* for meeting this Principle, along with seventeen indicators with specific criteria and acceptance limits. Assessors review the non-proliferation characteristics of the nuclear energy system in a given State to determine how well the requirements are met" (p. 2 of Ref. [13], original emphasis).

The review determines the strengths and weaknesses of the nuclear energy system regarding proliferation resistance. Despite these methodological differences, the two approaches have several similarities. For example, the two share a common definition of proliferation resistance.

> "Both approaches have a hierarchal analytical structure involving proliferation resistance principles, high-level evaluation factors and multiple measures or criteria related to each high-level factor. Both approaches treat proliferation resistance as a function of multiple *extrinsic measures* (e.g. safeguards, etc.) and *intrinsic features* (e.g. material attractiveness, etc.), and characterize proliferation resistance in terms of each ... . Both approaches recognize the concept of *barriers* to proliferation, but implement the concept differently. Neither approach aggregates its results into a single numerical value or grade, so that strengths and weaknesses under each of the main evaluation criteria are explicitly considered. Both approaches are primarily technical evaluations that incorporate institutional and policy contexts for the systems under consideration" (p. 3 of Ref. [13], original emphasis).

## 3.3. STEPS IN THE GIF EVALUATION PROCESS

The GIF evaluation process includes nine specific steps that are organized into four main activities:

(1) Defining the work;
(2) Managing the process;
(3) Performing the work;
(4) Reporting the work.

Each step is primarily associated with one of these activities. The nine steps are explained in Fig. 2. Clearly, some level of management is associated with each step. Reporting cannot all be carried out at the end but needs to be generated as the work progresses. Thus, the process is iterative, and sometimes the steps are concurrent.

Note that the steps in the process are numbered in the order they are first performed (as shown in Fig. 2) but are examined under the four main activity areas described.

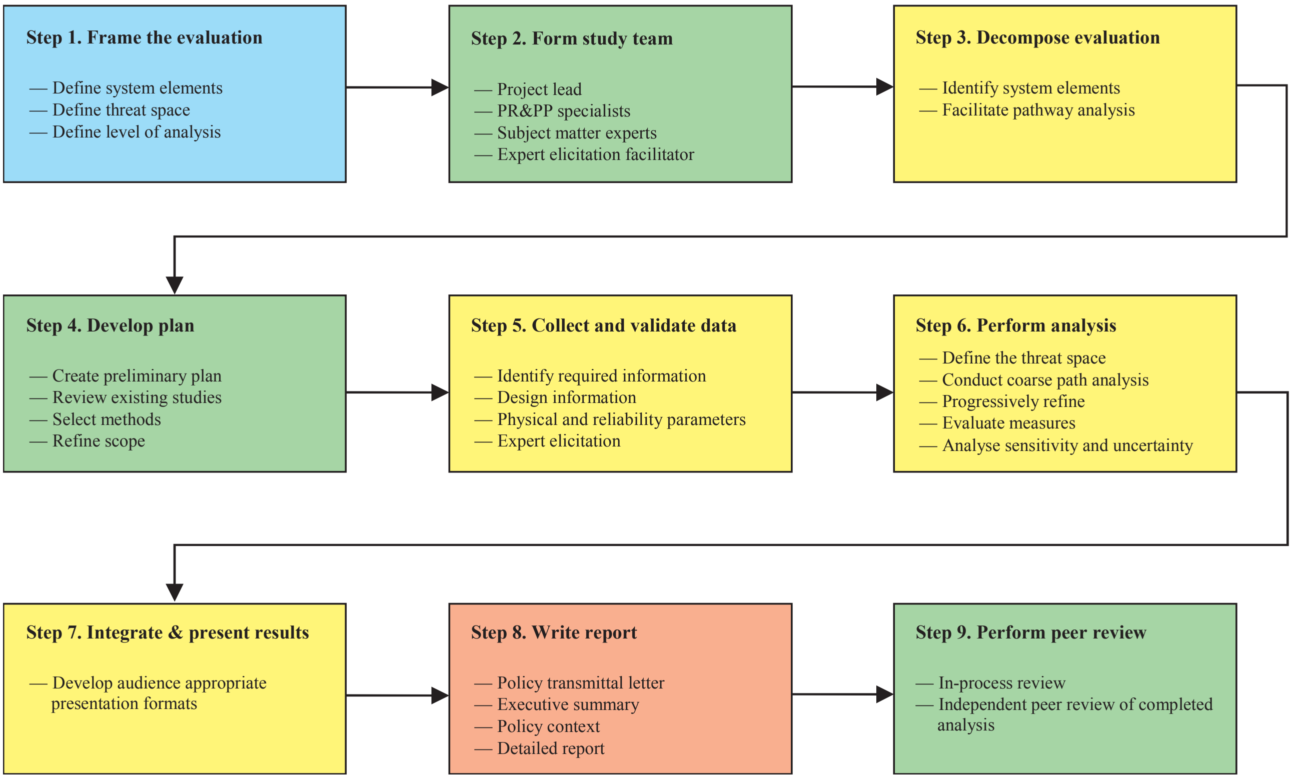


*FIG. 2. Steps in the evaluation process.*

To ensure the completeness and adequacy of results, it is required that:

— Problem is structured systematically;
— Expert analysis team is assembled;
— Competent peer review is ensured.

### 3.3.1. Defining the work (step 1)

*3.3.1.1. Step 1: Frame the evaluation clearly and concisely*

The process of framing an evaluation requires close interaction between the analysts and the sponsors (i.e. the organization that commissions the study — governmental and international agencies, private utilities and design organizations) to specify the scope, in particular the system elements (e.g. facilities, processes and materials) and the range and definition of threats. The institutional context in which safeguards and other international controls would be implemented (e.g. national and international safeguards requirements and regulatory guidance) also needs to be specified in sufficient detail.

The process enables evaluation to be performed at different levels, depending on the sponsor's requirements. From the pre-conceptual design to a fully operational facility, the evaluation is required to become progressively more detailed. The time frame can also dictate the depth of analysis — quick and coarse evaluations may be needed when answers are required within weeks or months and, for some problems, potentially even sooner. Such shortcuts, however, entail a greater degree of uncertainty in the results.

### 3.3.2. Managing the process (steps 2, 4 and 9)

*3.3.2.1. Step 2: Form a study team that provides the required expertise*

The team will include experts in all required and relevant technical areas, as well as experts in conducting the elicitation in an unbiased manner, with full description of the range of opinions. An example of an expert elicitation process for performing this work is provided in Section I.3, Appendix I.

*3.3.2.2. Step 4: Develop a plan describing the approach and desired results*

Before undertaking this major analysis, the evaluation plan is thoroughly developed, reviewed and documented. In addition, staff resources, costs, schedule, form of the results and documentation need to be clearly defined. Milestones need to be developed, particularly for regular reporting to sponsors. A detailed plan for the conduct and use of peer reviews is also important to ensure quality. While developing the plan and implementing the information gathering and analysis tasks, coordination with safety evaluation, safeguards and physical security work for the nuclear energy system could provide significant benefits.

*3.3.2.3. Step 9: Commission peer reviews*

Any evaluation used to support decision making or planned for wide distribution will include a peer review to ensure product quality. Two types of peer review have been widely used and provide different types of support:

(1) In-process peer review or steering committee;
(2) Independent peer review of the completed analysis.

In-process peer review brings an expert group of practitioners and decision makers into the process at regular intervals — perhaps once per quarter — to be briefed on the status of work and any known problem areas. Independent peer review allows an objective and thorough review of the finished product by independent outside experts who have not been involved in the evaluation. Both types of peer review have a potential role in proliferation resistance analysis.

### 3.3.3. Performing the work (steps 3, 5, 6 and 7)

Four steps are involved in this main activity. Steps 3 and 5 prepare for the required analysis, while the bulk of the analysis occurs under step 6, followed by integration of results for presentation in step 7.

*3.3.3.1. Step 3: Decompose the problem into manageable elements*

This step breaks down the nuclear energy system into a set of system elements and threats to permit pathway analysis. Expert judgement may be used to identify system elements and threats that will be covered under qualitative coarse pathway analysis and those that will be subjected to progressive refinement with quantitative analysis.

*3.3.3.2. Step 5: Collect and validate input data*

The quantities and sources of input data depend on the scope of analysis. Validation of input data implies either independent review of the data sources or examination of the consistency and basis for expert elicitation. If information and input data used in the analysis come from classified or sensitive sources, it is necessary for the analyst to ensure that this information is protected appropriately, including the possibility of classified or sensitive evaluation results. Most important in this step is a strong interface with facility designers. Designers need to be key members of the PR&PP evaluation team. Later, when the evaluation is applied to operating facilities, members of the operations team are also to be included.

*3.3.3.3. Step 6: Perform analysis*

The actual evaluation is a multistage process. It addresses the system response and outcomes parts of the methodological approach. The process is fully described in Appendix I.

*3.3.3.4. Step 7: Integrate results for presentation*

Here, an audience appropriate presentation format needs to be developed and the results presented according to this format.

### 3.3.4. Reporting the work (step 8)

*3.3.4.1. Step 8: Write the report*

As noted previously, reporting to the sponsors will be an ongoing process, and elements of the final report may be generated in draft form throughout the process. Ultimately, the analysts are required to provide the results in a form that can be understood by the user, thereby enabling the user to draw appropriate conclusions. If the report contains classified or sensitive information, it may be necessary to include an unclassified summary.

## 3.4. STEPS IN THE INPRO EVALUATION PROCESS

The INPRO methodology for assessment of proliferation resistance provides both a framework for assessing proliferation resistance of a nuclear energy system and guidance to improve it. It is based on the principle that proliferation resistance intrinsic features and extrinsic measures will be implemented throughout the full life cycle of the nuclear energy system to help to ensure that it will continue to be an unattractive means of acquiring fissile material for a nuclear weapons programme. The methodology has one basic principle and five user requirements, along with relevant criteria. The INPRO assessment methodology is published in IAEA-TECDOC-1575 Rev. 1 [6].

The INPRO proliferation resistance methodology defines the basic principle as:

> "Proliferation resistance intrinsic features and extrinsic measures shall be implemented throughout the full life cycle for innovative nuclear energy systems to help ensure that INSs 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" (p. 7 of Ref. [6]).

Intrinsic proliferation resistance features result from the technical design of nuclear energy systems, including those that facilitate the implementation of extrinsic measures (e.g. IAEA safeguards). Extrinsic proliferation resistance measures result from State decisions and undertakings related to nuclear energy systems, such as control and verification measures. To meet the goal defined in the basic principle, INPRO proliferation resistance methodology established five user requirements:

(1) State to establish a sufficient legal framework (e.g. Treaty on the Non-Proliferation of Nuclear Weapons, comprehensive safeguards agreement and regulatory bodies);
(2) Designer to keep the attractiveness of nuclear material low;
(3) Designer to make diversion of nuclear material difficult and detectable;
(4) Designer to incorporate multiple barriers against proliferation;
(5) Designer to optimize costs of proliferation resistance measures, including safeguards.

INPRO methodology criteria are used by the INPRO assessor to check whether, and how well, user requirements have been met by a given nuclear energy system. INPRO criteria consist of one or more indicators, with evaluation parameters basically describing barriers against proliferation and associated acceptance limits. This limit is a target, either qualitative or quantitative, with which the value of an indicator can be compared by the INPRO assessor, leading to a judgement of acceptability (i.e. whether the criteria have been met).

The proliferation resistance assessment is performed at three levels:

- State level;
- Nuclear energy system level;
- Facility level (including facility specific pathways).

The main inputs required for an assessment are results of acquisition path analyses and of evaluations performed by safeguards experts together with designers, operators and government organizations responsible for proliferation resistance issues.

The robustness of barriers against proliferation depends on the State's capabilities. It is not a function of individual barrier characteristics but an integrated function of all of the barriers, and it is measured by determining whether the safeguards goals can be met.

In essence, INPRO proliferation resistance methodology can be summarized as an approach to help project management to ensure that safeguards experts confirm that the nuclear facilities or systems evaluated can be safeguarded effectively and efficiently and that costs for implementing international safeguards are affordable and have been minimized.

# 4. PROLIFERATION RESISTANCE ASSESSMENT AND SAFEGUARDS BY DESIGN

## 4.1. PURPOSE OF PERFORMING ASSESSMENT (DEFINING THE WORK)

Proliferation resistance assessment is generally performed at three levels:

- State level;
- Nuclear energy system level;
- Facility level.

For each of the three levels, experts in international safeguards are asked whether safeguards goals can be met effectively and efficiently. At the State level, the analysis is whether the nuclear energy system will continue to be an unattractive means to acquire fissile material for a nuclear weapons programme. The analysis takes into account State specific conditions such as:

(a) Legal framework established by the State to comply with non-proliferation commitments;
(b) State's nuclear fuel cycle and nuclear capabilities;
(c) State's industrial and scientific infrastructure.

The analysis also examines the possibility of using clandestine nuclear material and activities (State level acquisition path and proliferation potential analyses):

(1) At the nuclear energy system level, the analysis is whether it is possible to produce nuclear material within the nuclear energy system that can be used for a nuclear weapon by misuse and diversion without the risk of early detection, which leads to safeguardability at the facility level.
(2) At the facility level, the analysis is whether the safeguards goals of national and international safeguards organizations can be met effectively and efficiently (SBD).
(3) Weak intrinsic proliferation resistance parameters found in the analysis may be compensated by other intrinsic features or extrinsic measures (higher effort in safeguards).

A number of organizations are involved in developing approaches for performing SBD. The process described here is based on the GIF PR&PP methodology. However, new advances are appearing, and the SBD assessment techniques are evolving and improving. Nevertheless, the ultimate goal of this procedure is to provide a structured approach based on the current state of the art for evaluating nuclear facility safeguardability. The aim is to provide a means for identifying and optimizing barriers and technical features that will make the undetected diversion of nuclear material or misuse of the facility more difficult. When the threats have been characterized and the pathways identified, the safeguards team can characterize each pathway using the set of evaluation measures described here. The results of the individual pathway analyses will be useful in evaluating various safeguards measures, identifying potential weaknesses or alternative approaches, and providing a basis for improving and enhancing facility safeguards where necessary to demonstrate that IAEA safeguards goals can be met. The results also provide a basis for determining whether the facility design facilitates effective and efficient safeguards implementation.

A detailed procedure for performing the SBD analysis at the facility level using the GIF PR&PP methodology is provided in Appendix I and summarized here. As described in Section 3.1, because some information is usually lacking during the conceptual design process, an elicitation process using subject matter experts is generally applied to complete this work.

This assessment and analysis process may be qualitatively and quantitatively addressed using robust system modelling tools. However, a detailed quantitative analysis can be lengthy and time consuming and may only be necessary in some cases. In the proposed SBD process, a need exists for early evaluation of the proposed design (intrinsic features) and related safeguards (extrinsic measures) of a nuclear energy system. This begins with the pre-conceptual design stage and early conceptual design stages, while facility design features can still be changed to accommodate an enhanced safeguards approach. This section describes an approach for employing subject matter experts to perform an analysis to the level of detail desired.

## 4.2. SAFEGUARDABILITY ANALYSIS (DEFINING AND PERFORMING THE WORK)

The following is a set of fundamental questions related to facility safeguardability that need to be answered during the assessment. The answers will influence the design of the facility safeguards system and complement the design progression of the nuclear facility throughout the process.

(1) What are the relevant facility safeguards requirements?
(2) Are there new diversion, misuse or safeguards challenges in the design that are not already addressed by IAEA safeguards approaches for this kind of facility or design?
(3) How can the facility or process be designed to enhance barriers to misuse or diversion and improve detectability?
(4) If the facility design is altered to minimize the potential for nuclear material diversion, what are the associated trade-offs (e.g. reduction in operating efficiency or increased cost)?
(5) Can the design of the safeguards system support international safeguards?
(6) Can the facility design be optimized so that the objectives of the facility designer, owner or operator, national nuclear regulator and the IAEA can simultaneously be met in a cost effective manner?

It is imperative that the location and installation of safeguards equipment be anticipated early in the design process to avoid potential cost and schedule impacts later in the process. For this reason, the facility designer and project design team need a tool to answer the questions posed above.

The pathway analysis will address the following areas:

— Strengths and weaknesses of accountancy verification;
— Identification of potential proliferation strategies (e.g. abrupt diversion, protracted diversion or misuse followed by diversion);
— How the facility can be misused for undeclared production, including production of special nuclear material;
— How target material can be removed undetected by the containment and surveillance system.

The first step in performing a safeguardability analysis is to model the facilities. The next step is to identify potential targets in the various facilities that could be used to divert (acquire) or misuse fissile material. Another important element is identifying potential diversion points in the facilities that could be used to bypass safeguards. Finally, pathways are identified showing how the desired material acquisition could occur. Consideration is to be given to facility operational modes from the design process through to decommissioning, since key information or conclusions may vary depending on the operational mode. The key information to be identified includes:

(a) Potential diversion points.
(b) Material that could be diverted.
(c) The kind(s) of container(s) normally used for moving material.
(d) Potential misuse actions.
(e) Safeguards that would potentially detect improper movement of material, including:
   (i) Safeguards requirements (e.g. for monitoring, containment, surveillance, and assay systems);
   (ii) Location, space and utility requirements for the safeguards components, including power and data transmission;
   (iii) Areas where changes to the nuclear facility design or process could potentially improve the effectiveness of the safeguards system;
   (iv) Areas where changes to the design or process could render an identified pathway technically unachievable;
   (v) The expected material unaccounted for (MUF) and the limit of error for MUF for this facility and each element of the facility.

The results of the pathway analysis illustrate the actions a proliferator would need to undertake for successful misuse or diversion, as well as potential acquisition pathways.

## 4.3. ESTIMATION OF MEASURES (PERFORMING THE WORK)

As each acquisition pathway is assessed, the experts will use an elicitation process to determine the associated value for each measure listed below. Treatment of uncertainty is critical, and experts are required to state explicitly the uncertainty they assign to each ranking. If the experts cannot agree (i.e. reach consensus on a category assignment), then a description of the disagreement will be included in the assessment report, along with a discussion of the overall implications for the conclusions.

The measures used either directly or indirectly for determining safeguardability in this task include:

(1) **Fissile material type**: A categorization of material existing in the facility based on IAEA safeguards definitions.
(2) **Detection resources**: The costs related to staffing, equipment and operational impacts required to apply international safeguards to the nuclear energy system. This measure also includes the cost of any intrinsic features created or changed to meet safeguards goals.
(3) **Detection probability**: The cumulative probability of detecting diversion or misuse along an acquisition pathway.
(4) **Technical difficulty**: The inherent difficulty, arising from the need for technical sophistication and materials handling capabilities, required to overcome the barriers to diversion or misuse.[2]
(5) **Misuse and diversion time**: The minimum time required to overcome the safeguards barriers and divert or misuse the safeguarded material.

A sixth measure often used is proliferation cost, which is defined as the cost to a proliferator to create a nuclear device, including the acquisition, processing and weaponization of the material, but it is not relevant to the cost of implementing a safeguards system.

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[2] Barriers refer to intrinsic barriers (e.g. technical difficulty) and extrinsic barriers (e.g. safeguards) but do not include difficulties in weaponization.

The experts will use information developed during the previous steps as the basis for their final elicitations. When the quantification process is complete for each pathway, the results will be collected in the final elicitation process. These results will provide a basis for evaluating and comparing the interactions of proposed intrinsic features to identify those of maximum benefit to safeguards. These results will be used to ensure that the facility design meets national and international nuclear safeguards requirements, as determined by the responsible national entity and the IAEA, respectively.

### 4.4. SAFEGUARDABILITY CHARACTERIZATION (PERFORMING AND REPORTING THE WORK)

The information developed on safeguardability provides an additional set of inputs to be used in conjunction with the facility design process. The facility's safeguardability is characterized by identifying:

— Relevant safeguards requirements;
— Relevant facility design information;
— Preliminary diversion path (including misuse) analysis and subsequent iterations;
— Prospective IAEA safeguards tools and measures and any envisaged modifications;
— Safeguards system design;
— Trade-off analyses relevant to the optimization of the safeguards system and facility design.

# 5. IMPLEMENTATION OF PROLIFERATION RESISTANCE AND PHYSICAL PROTECTION MEASURES IN INNOVATIVE SMALL AND MEDIUM SIZED REACTORS

### 5.1. ARGENTINA: CAREM

CAREM is a project of Argentina's National Atomic Energy Commission, whose purpose is to develop, design and construct an innovative, simple and small nuclear power plant. This integral type PWR has an indirect cycle with distinctive features that simplify the design and support the objective of achieving a higher level of safety.

#### 5.1.1. Intrinsic proliferation resistance features

*5.1.1.1. Safeguards implementation*

Argentina has much experience in the surveillance and control of nuclear material. For safeguards implementation in the CAREM reactor, all the refuelling tasks will be developed in the reactor hall, which is designed to allow remote monitoring of all nuclear material handling. An important feature is that there is only one entrance–exit point for controlling and monitoring. This design simplifies accountability and surveillance of nuclear items during movement and transfer. Provisions are considered to cover and to seal the spent fuel pool during operation cycles. Intrinsic proliferation resistance features that facilitate the implementation of extrinsic measures are considered in the CAREM design. Long life cores do not provide cost efficient proliferation resistance for CAREM.

The CAREM nuclear power plant has four main handling and storage fuel areas:

(1) Fuel reception area;
(2) Fresh fuel storage room;
(3) Reactor pressure vessel;
(4) Spent fuel storage pool.

Two isolated material balance areas for spent fuel (pressure vessel core and spent fuel pool) have been included to facilitate safeguards implementation and to reduce costs. All the spent fuel will be located in these two material balance areas, and remote monitoring systems will conduct the surveillance. It is important to note that during nuclear commissioning and operation, there will be no possibility of physical access to fissile material.

*5.1.1.2. Core life length*

The core life length could be optimized by selecting, for instance, a common variable to be applied to all steps of the fuel cycle. Safeguards related costs could also be included. The influence of the use of different enrichment grades was analysed, and the results were significant: safeguards costs are very small compared to fuel costs; and they do not significantly impact on the optimum enrichment.

### 5.1.2. Physical protection features

The objective is:

(a) To prevent, detect, avoid, delay and respond to malicious acts affecting the CAREM-25, in particular those aiming:
  — To extract or disperse nuclear material;
  — To sabotage or encroach on a nuclear power plant, which may cause an accident with severe radiological consequences.
(b) To provide the necessary means to prevent the unauthorized use of nuclear material, equipment or installations, as established by the nuclear regulatory authority.
(c) To monitor and control personnel movements to prevent theft and acts of vandalism.
(d) To prevent the unauthorized use or transmission of technologically relevant information.

To conclude, evaluating different proliferation resistance costs will be considered. Intrinsic proliferation resistance features that facilitate the implementation of extrinsic measures are considered in the CAREM design. In its design base, the CAREM-25 considers the loss of a heat sink and a black-out during the grace period. Provisions are considered to protect spent fuels and to facilitate core decay heat removal after the grace period. These provisions could improve safety and intrinsic PR&PP.

### 5.1.3. INPRO studies

Lessons learned from the INPRO studies are reported in Ref. [15]. A detailed analysis was performed to determine the level of proliferation resistance for planned national fuel cycle facilities to be located at a central fuel cycle facility (for the approach used in this analysis, see annex A of Ref. [6]).

Several evaluation parameters were found in the national fuel cycle that were defined as ‘weak’ — that is, isotopic composition of the spent fuel, operation of enrichment and reprocessing facilities, and the absence of an additional protocol in force.

In a second step, Argentina applied the INPRO assessment method (see chapter 3 of Ref. [6]) and concluded that although some weaknesses were identified in the Argentine nuclear energy system, they could be compensated for by an increased safeguards effort. Thus, all INPRO requirements would be met by the national nuclear energy system. The analysts further emphasized that these ‘weaknesses’ were present in any other nuclear energy system that includes enrichment, reprocessing, pressurized heavy water reactor or PWR and emphasized the INPRO recommendation for consideration of improvements over the existing situation in energy systems currently in use.

### 5.1.4. Feedback from the accident at the Fukushima Daiichi nuclear power plant

In its design base, the CAREM-25 considers the loss of a heat sink and a black-out during the grace period. After the grace period, provisions are considered to allow core decay heat removal and containment cooling using a fire extinguishing system or an autonomous system. Additional provisions allow electrical supply to safety related systems using autonomous generation systems. Seismic requirements were also reviewed, and provisions were considered to cover the spent fuel pool. These provisions could improve safety and intrinsic PR&PP.

## 5.2. BRAZIL: FBNR

The FBNR is a small reactor of 70 MW(e) with no requirement for on-site refuelling. It is a pressurized LWR with spherical fuel elements. The reactor is modular in design, and each module is assumed to be fuelled at the factory.

### 5.2.1. Intrinsic proliferation resistance features

The distinguishing feature of the FBNR in terms of non-proliferation is that under shutdown conditions, all fuel elements remain in the fuel chamber, where only a single flange needs to be sealed and controlled for safeguards purposes. The spent fuel of the FBNR is in a form that can be used directly as a source of radiation for irradiation purposes.

The concept is based on both sealing the fuel chamber and denaturing the fuel itself. The sealing of the fuel in the fuel chamber, accompanied by surveillance, enables continuous inspection of the fuel — guaranteeing effective control. The isotopic denaturing of the fissile fuel, both in the $^{233}U$–thorium cycle as well as in the classical uranium–plutonium cycle, would further increase the proliferation resistance, as it will require isotope or chemical separation technology to produce weapons-grade materials.

Adopting a thorium cycle as an intrinsic measure will hinder the possibility of misuse of nuclear materials for nuclear weapons. The mixing of thorium with LEU or plutonium results in the production of $^{233}U$, which is diluted along with $^{235}U$ in $^{238}U$. The access to $^{233}U$ is only possible through isotope separation techniques. In addition, the production of gamma emitting $^{208}Tl$ in the thorium cycle is a hindrance to nuclear proliferation. If the uranium–plutonium cycle is applied, one can increase the $^{238}Pu$ concentration by adding neptunium to the fresh fuel. Starting from a certain concentration of $^{238}Pu$ (around 8%), the alpha decay heat is so strong that the metallic plutonium sphere in a nuclear device, as well as the surrounding chemical explosives, becomes plastic or even melts. This effect means that the fuel of the reactor at any time is less useful for weapons. In any event, LWRs tend to create the less desirable plutonium isotopes over time as the reactor operates normally, thus tending to denature the plutonium as it is created. Thus, the combination of sealing the reactor and the isotopic denaturing of the irradiated fuel will also increase proliferation resistance.

### 5.2.2. INPRO studies

The four types of intrinsic feature of the FBNR are intended to reduce costs and efforts of international safeguards implementation. INPRO methodology requires that the intrinsic proliferation resistance features consist of technical features of a nuclear energy system that:

(1) Reduce the attractiveness for nuclear weapons programmes of nuclear material during production, use, transport, storage and disposal — accomplished by denaturing the fuel;
(2) Prevent or inhibit the diversion of nuclear material — accomplished by sealing the fuel chamber;
(3) Prevent or inhibit the undeclared production of direct use material — accomplished by coating the vessel with a neutron absorber material;
(4) Facilitate verification, including continuity of knowledge — accomplished by the use of a television camera to conduct surveillance of the fuel chamber.

#### *5.2.2.1. Reduced possibility of fissile material production*

The neutron leakage from the reactor is negligible. However, the reactor pressure vessel is cladded with neutron absorbing materials to eliminate the possibility of neutron irradiation to any external nuclear material that could eventually cause embrittlement of the reactor pressure vessel.

## 5.3. CANADA: EC6

The EC6 is a 740 MW(e) pressure tube reactor designed by Atomic Energy of Canada Limited. The design evolved from the CANDU-6 design.

### 5.3.1. Designing for proliferation resistance

#### *5.3.1.1. EC6 intrinsic proliferation resistance features*

Proliferation resistance is a total fuel cycle characteristic and is not limited to the reactor. CANDU natural uranium fuel cycle does not require an enrichment facility. The plutonium isotopic quality is reactor grade (similar to LWRs), despite relatively low average burnup owing to high thermal flux.

#### *5.3.1.2. Operational features*

On-power refuelling requires a complex, automated, monitored process (frequent fuelling is required, at coolant temperature and pressure, in approximately 10 Sv/h fields). The refuelling frequency cannot be significantly increased because it takes place near the maximum capability of the fuelling machine. The excess reactivity of the core is low because it cannot tolerate significant added absorbing material or modification of refuelling scheme.

The refuelling process has high operational transparency. It is automated and remotely monitored under computer control, and core data (in-core flux mapping detectors, zone flux detectors and controller level data) are sensitive to refuelling perturbations. It is also necessary to create continual records for verification, as well as potential on-line data for third party tracking, if desired.

New and used fuel movement is a well defined process that facilitates a robust and effective safeguards approach in which every fuel bundle is tracked as it moves through the facility.

### 5.3.2. Forensic identification of nuclear materials and proliferation resistant nuclear design

In the forensic identification of the origins of nuclear materials, characteristic isotopic signatures of nuclear materials are required. For reactors, signature models for plutonium, thorium and uranium containing fresh and depleted reactor fuels need to be developed. However, the difficulty is that there are the following classes of nuclear materials with which the Canadian Nuclear Safety Commission deals:

(a) Fresh/irradiated fuel from CANDU power reactors (natural uranium fuel);
(b) Fresh/irradiated high and low enriched fuel from research reactors (Slowpoke, National Research Universal, Zero Energy Deuterium-2, Maple and McMaster Nuclear Reactor).

The challenges in developing fuel signature models are due to differences in fuel compositions for different CANDU units, as fuel management among existing CANDUs differs (i.e. exit burnups vary and aging may have an effect on burnup). However, there is no repetitive pattern to differentiate between fuel exiting from different CANDU units. Similar problems to those listed for CANDU fuels may arise for irradiated research reactor fuels, and additional problems may arise, for example, if a reactor is used for fuel or sample irradiations (implied generation of non-typical materials).

In Canada, reactor facility proliferation resistant design features currently rely on the implementation of the safeguards regime including material control measures based on safeguards requirements. When developing proliferation resistant design features of nuclear reactors, consideration needs to be given to the forensic aspect of identifying fuel origins. The design needs not only to minimize the content of sensitive nuclear materials throughout the fuel cycle but also to facilitate the development of its fuel isotopic signature.

## 5.4. CHINA: HTR-PM

China is rapidly developing large size reactors, but it still focuses on the development of SMRs and participates in every activity related to them. In 1992, the Government approved the construction of the HTR-10, a 10 MW pebble bed high temperature gas cooled test reactor. In 2003, the reactor reached full power. The second step of the high temperature GCR application in China began when the HTR-PM project was launched. The first two 250 MW(t) HTR-PMs are under construction at the Shidaowan plant in Shandong Province and together drive a steam turbine generating 200 MW(e). Construction was initially planned to start in 2009. The HTR-PM concept design started in 2002. Based on the safety concept of modular high temperature GCR, the HTR-PM is featured as the pebble bed annular core and standard steam turbine generator system, with proven technology and independent intellectual property.

### 5.4.1. HTR-PM fuel cycle from a non-proliferation perspective

The HTR-PM is a thermal reactor with salient design features, which include graphite as a moderator. Graphite is heat resistant at high temperatures. It has large heat capacity and is easily compatible with helium, which is used as the coolant (single phase, good chemical stability and no neutron absorption). The HTR-PM utilizes a triple coated isotropic ceramic coated particle fuel element, and the core outlet temperature is 750°C.

With regard to HTR-PM fuel handling and storage, the fuel inventory in-core is approximately 420 000 fuel elements, and the level of fresh fuel enrichment reaches 8.5%. The average discharge burnup is 90 GWd/tU, with a maximum of 100 GWd/tU. The number of fresh and spent fuel per day is around 400 fuel elements, and the number of shuffling per day is around 6000 fuel elements. The spent fuel storage takes place on-site for the entire plant life and 40 000 fuel elements are stored in one air cooled tank, next to the reactor building. There is no immediate plan for spent fuel reprocessing.

#### *5.4.1.1. Intrinsic non-proliferation features*

The intrinsic non-proliferation features include high and uniform fuel burnup. The designer of the HTR-PM considers that efficient burning of $^{239}$Pu and relatively heavier plutonium isotopes owing to high burnup and hard spectrum make them less desirable for weapon purposes. Fine burnup measurements and records of fuel elements lead to a good knowledge of the spent fuel in the storage tanks.

## 5.5. INDIA: AHWR300-LEU

The AHWR300-LEU was designed and developed by the Bhabha Atomic Research Centre to achieve large scale use of thorium for the generation of commercial nuclear power. This reactor is designed to produce most of its power from thorium, with no external input of $^{233}$U in the equilibrium cycle.

### 5.5.1. Intrinsic proliferation resistance features

Use of thorium based fuels in general, and LEU–thorium based fuel in the AHWR300-LEU as an example, offers a high degree of intrinsic proliferation resistance, along with enhanced fuel resource sustainability and high levels of intrinsic safety and security. Along with the thoria matrix, higher levels of $^{235}$U enrichment will offer greater all round benefits. Intrinsic approaches to enhance proliferation resistance would depend, to a large extent, on the permissible limits on fuel isotopic vectors. There is perhaps a need to conduct new scientific evaluations and define these limits consistently with some realistic objectives concerning proliferation barriers.

In the AHWR, there is reduction of minor actinides, efficient burning of plutonium and the fissile content reduces from 75% to 23% (residual plutonium in the spent fuel is less). The discharged uranium from the AHWR300-LEU contains 1400 ppm of $^{232}$U.

The AHWR300-LEU offers enough flexibility to accommodate different kinds of fuel cycles. LEU (19.75% $^{235}U$) is a good external feed in the thorium oxide fuel in the AHWR300-LEU, with all reactivity coefficients negative. The LEU content of 21.3% mixed with thorium dioxide gives a high discharge burnup of about 64 GWd/t. However, the average discharge burnup obtained from (thorium, LEU)MOX reduces to almost half if the LEU content is reduced from 21.3% to 15.4%.

*5.5.1.1. LEU–thorium fuel in the AHWR300-LEU to enhance proliferation resistance characteristics*

The use of LEU and thorium leads to reduced plutonium generation in spent fuel with a lower fissile fraction and a higher (around 10%) proportion of $^{238}Pu$. The fissile uranium in the spent fuel amounts to around 8%, and it also contains around 200 ppm of $^{232}U$, whose daughter products produce high energy gamma radiation. These attributes form the basis of the intrinsic proliferation resistant features of the AHWR300-LEU. The composition of the fresh as well as the spent fuel of the AHWR300-LEU makes the fuel cycle inherently proliferation resistant owing to the higher $^{238}Pu$ fraction and the lower fissile content of plutonium.

## 5.6. JAPAN

### 5.6.1. An approach for early deployment of small pressurized water reactors without on-site refuelling

Early deployment of SMRs is essential to enhancing resistance and security. One solution is the effective and full utilization of accumulated operating experience of LWRs. It is suggested to begin with a well proven system that can satisfy basic and current requirements and then adopt newly proven technologies step by step. One new technology is the adoption of higher enrichment fuel of more than 5% by mixing a low percentage of erbium(III) oxide as a burnable poison. This fuel can be handled by current fuel fabrication facilities without violating critical safety limits. The reactor characteristics are almost the same as those with gadolinium burnable poison.

Another new technology is the adoption of coated particle fuel. Studies are now under way to use metallic particle fuel dispersed in a metallic matrix in the cladding tubes. Since metallic particles are denser than the oxide fuel, the fuel has a longer core life when the fissile enrichment is identical. This concept can also improve the capability of confinement of radioactive fission products inside the fuel rods, resulting in greater plant safety.

To fulfil this objective with SMRs without on-site refuelling, an approach that utilizes proven systems as a starter could be one of the options. As an example, Japan’s first nuclear ship — the Mutsu — has been selected as the reference, with modifications based on current components and technologies. New technologies are to be adopted step by step.

## 5.7. RUSSIAN FEDERATION

### 5.7.1. Design concept of the reactor facility based on the VK-100 for regional nuclear power engineering

The VK-100 reactor facility will meet the demands of the power level to solve the problems of heat supply in many regions of the Russian Federation. The path breaking use of technical solutions mastered during the long term operation of the VK-50 provides uniqueness of safety, reliable protection from external impact and long term operation of new modern nuclear heat power plants with the VK-100 reactor. Mastering innovative technical solutions at the demonstration facility will reduce serial construction periods and costs.

# 6. FINDINGS FROM PREVIOUS STUDIES

## 6.1. PREVIOUS STUDIES ON THE GIF METHODOLOGY

### 6.1.1. Example Sodium Fast Reactor

The GIF methodology to evaluate PR&PP was developed with the aid of a series of studies based on a hypothetical nuclear energy system called the Example Sodium Fast Reactor (ESFR) [4]. Lessons learned from the ESFR case study include:

> "— Each PR&PP evaluation should start with a qualitative analysis allowing scoping of the assumed threats and identification of targets, system elements, etc.
> "— Detailed guidance for qualitative analyses should be included in the methodology.
> "— Access to proper technical expertise on the system design as well as on safeguards and physical protection measures is essential for a PR&PP evaluation.
> "— The use of expert elicitation techniques can ensure accountability and traceability of the results and consistency in the analysis.
> "— Qualitative analysis offers valuable results, even at the preliminary design level.
> "— Greater standardization of the methodology and its use are needed.
> "In addition, subgroups noted that during the evaluation process the analyst must frequently introduce assumptions about details of the system design which are not yet available at early design stages. ... As the study progressed, the working group realized that when these assumptions are documented, they can provide the basis for establishing functional requirements and design bases documentation for a system at the conceptual design stage [supporting the SBD concept]. By documenting these assumptions as design bases information, the detailed design of the facility can be assured of being consistent with the PR&PP performance predicted in the initial conceptual design evaluation" (p. 74 of Ref. [5]).

### 6.1.2. Proliferation resistance studies in the United States of America

Section 6.1.2 is based on pp. 75–76 of Ref. [5]. A multilaboratory team of US subject matter experts, including several members of the PR&PP Working Group, used the PR&PP evaluation methodology as the basis for a technical evaluation of the comparative proliferation potential associated with four generic reactor types in a variety of fuel cycle implementations. The evaluation team undertook a systematic assessment, capturing critical assumptions and identifying inherent uncertainties in the analysis. Although the results of the evaluations have not been publicly released, the PR&PP Working Group members involved were able to share their methodological insights with the full PR&PP Working Group, and a summary of the study [16] was presented at the 51st Institute of Nuclear Materials Management Annual Meeting.

The relevance of the insights varies based on the various stakeholders of a PR&PP evaluation: policy makers, system designers, and the safeguards and physical protection communities.

(a) Policy makers:
  (i) An assessment of the proliferation potential of a particular reactor design in the nuclear energy system will consider the system's overall architecture, accounting for the availability and flow of nuclear material in the front and back end of the fuel cycle.
(b) Designers:
  (i) Of the five proliferation resistance measures, the designer will directly influence three:
    — Detection probability;
    — Detection resource efficiency;
    — Material type.

(ii) To enhance detection probability and detection resource efficiency, designers can incorporate features in the design to facilitate easier, more effective and efficient safeguards for inspection and monitoring. For example, minimizing the number of entry and exit points for fuel transfer between system elements will enhance material containment, protection and accountancy, thus partially compensating for any lack of knowledge continuity by visual inspection during a fuel transfer.

(iii) The material type for proliferation resistance is related to the chosen composition of the nuclear material. The designer can optimize the design either to reduce the material's attractiveness (e.g. increase burnup in the uranium fuel to raise the fraction of $^{238}$Pu, thereby lowering the quality of plutonium in the spent fuel) or to make post-acquisition processing of the material more complex, indirectly increasing the technical difficulty for the proliferator.

(c) Safeguards inspectors:

(i) Augmenting inspections for handling and storing fresh and spent fuel would reduce proliferation potential.

(ii) Enhanced inspection of fresh fuel would reduce the proliferation potential of covert diversion and misuse.

(iii) Optimizing material type and material movement pathways to facilitate accountability measurements can make verification more effective and efficient.

## 6.2. PREVIOUS STUDIES ON THE INPRO METHODOLOGY

### 6.2.1. PRADA study

The INPRO proliferation resistance evaluation methodology provides both a framework for assessing proliferation resistance and guidance for improving the proliferation resistance of an INS. To ensure the usability of the INPRO proliferation resistance procedure, the INPRO Phase 2 Collaborative Project on Proliferation Resistance: Acquisition/Diversion Pathway Analysis (PRADA) was proposed by the Republic of Korea at the 10th INPRO Steering Committee Meeting, held in Vienna in December 2006 [6, 17].

The objectives of the PRADA study were to (p. 1 of Ref. [14]):

(a) Develop appropriate methods for identifying and analysing pathways for the acquisition of weapons-usable nuclear material;

(b) Evaluate the multiplicity and robustness of barriers against proliferation for the acquisition pathway by logic trees (i.e. success/failure trees and event trees) and qualitative methods;

(c) Based on these results, recommend an assessment approach for user requirement 4 of the INPRO methodology, regarding the multiplicity and robustness of proliferation barriers.

PRADA focused on identifying and analysing higher level pathways for the acquisition of fissile material for a nuclear weapons programme. A case study was performed using the process known as DUPIC (direct use of spent PWR fuel in CANDU reactors). The team studied the CANDU reactor to develop appropriate methods for the identification and analysis of plausible acquisition paths. The study provided recommendations for assessing the multiplicity and robustness of barriers against proliferation, including institutional, material and technical barriers as well as barriers from international safeguards implementation [14].

The Republic of Korea, which has been developing the DUPIC process, assumed the lead for the project and was supported by Canada, China, the European Commission and the United States of America [14]. The main conclusions of the PRADA study were:

(a) A proliferation assessment is performed at three levels:

(i) State level;

(ii) Nuclear energy system level;

(iii) Facility level (including facility specific pathways).

(b) The robustness of barriers against proliferation depends on State capabilities, and the relevance of specific barriers varies depending on the level of evaluation performed.

(c) The robustness of barriers is not a function of the number of barriers or their individual characteristics, but is an integrated function of the whole and is measured by determining whether the safeguards goals can be met.
(d) The PRADA study demonstrated that the INPRO and GIF methodologies complement each other in a productive way.

#### 6.2.2. INPRO studies for Russian small power reactors

Some work has been conducted to evaluate the proliferation resistance of a variety of Russian designed reactors [18, 19]. The proposed LEU fuel for these systems for export was examined using INPRO user requirement 2: attractiveness of nuclear material and technology. The results showed that the material attractiveness of the fuel in these designs was similar to that of conventional commercial reactors. However, these studies were incomplete, as INPRO user requirements 1 and 3–5 were not evaluated.

# 7. CONCLUSIONS

This publication presents a framework for performing an assessment of proliferation potential and safeguardability in SMR designs. It refers to the existing methodologies developed by GIF and INPRO in the area of proliferation resistance and recent advances in the area of SBD. This publication is intended to be used by SMR designers and operators.

Several innovative SMR designs were reviewed for features that have safeguards or non-proliferation significance. Examples of such features include:

— Longer intervals between refuelling;
— Delivery of assembled cores as one sealed unit;
— Siting in remote locations;
— Novel fuel designs and compositions.

The proliferation assessment and safeguardability evaluation tools developed by GIF and INPRO for large reactor concepts can be equally well applied to these SMR concepts. Features of safeguards significance can be evaluated for their impact on possible diversion paths and potential safeguards measures can be assessed. These methodologies identify the proliferation resistance and safeguards strengths and weaknesses in the assessed SMR. Requirements for new safeguards measures are identified. The assessment methodologies can be made simpler and easier to understand, especially for proven designs.[3] They can also require extensive datasets for in depth analysis.

Of primary importance is the observation that including safeguards requirements, activities and interactions early in project planning can facilitate effective and efficient implementation of safeguards. Awareness of requirements by all stakeholders offers the opportunity for synergy between competing requirements and for improved designs. In addition, as new safeguards concepts move from R&D to implementation and new technology is deployed at a new facility, closer cooperation between designers and the IAEA can be both possible and beneficial. Even for advanced designs, there is an opportunity to include consideration of safeguards in the project phase following design certification, when the engineering details and construction project management are created before the start of construction.

SBD concepts are evolving and improving. Opportunities for cooperation with the IAEA can be utilized. Work on analysing safeguardability is ongoing by diverse organizations, and this will result in improved approaches for the analysis of facility concepts. Work performed using the approaches described in this publication can contribute to improved understanding of mitigating proliferation concerns and can provide valuable insights into the best methods to enhance proliferation resistance and to more economically, effectively and efficiently safeguard the innovative SMRs.

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[3] The two year INPRO collaborative project entitled Proliferation Resistance and Safeguardability Assessment Tools (PROSA) is currently addressing the improvement of such tools.

## Appendix I

## PERFORMING THE ANALYSIS: EXAMPLE OF A PROCEDURE TO ASSIST FACILITY SAFEGUARDABILITY ANALYSIS IN SUPPORT OF SAFEGUARDS BY DESIGN

### I.1. INTRODUCTION

The concept of performing SBD has recently been gathering momentum, as the nuclear industry begins its expected expansion. Analyses of safeguardability might be used to:

(a) Compare and evaluate the effectiveness of nuclear safeguards measures;
(b) Optimize the designer's proposed safeguards approach;
(c) Objectively and analytically evaluate nuclear facility safeguardability (i.e. ability to meet domestic and international nuclear safeguards requirements);
(d) Anticipate IAEA safeguards requirements and identify potential ways in which to cost effectively address safeguards challenges posed by the facility design;
(e) Compare, evaluate and optimize barriers within the facility and process design to minimize the risk of misuse or diversion of nuclear material;
(f) Systematically evaluate cost and operating efficiency trade-offs between variations in the safeguards system and the facility design;
(g) Provide a documented paper trail and foundation for the national nuclear regulator and the IAEA to affirm that the proposed safeguards system and facility design will meet national regulations and international safeguards requirements.

The purpose of Appendix I is to present the views of the authors on how the GIF PR&PP methodology might be adapted to support a structured approach for evaluating nuclear facility safeguardability, identifying and optimizing barriers and technical features that will render more difficult and detectable any misuse of the facility to produce undeclared material or to divert any nuclear material. The approach will compare nuclear safeguards measures and facility design features, and will identify potential changes to enhance safeguards effectiveness and efficiency. This procedure will thus provide a means to optimize the proposed facility safeguards approach.

It is important to ensure that the team assembled to perform this work is considered an important element of the project design team — on the same level as safety, security and operations planning. As with the groups responsible for these elements of facility design and development, the safeguards design team needs to work together through the facility design and development cycle.

### I.2. PURPOSE AND SCOPE

The purpose is to describe an expert elicitation process for FSA that would provide a basis for accomplishing the following tasks:

(1) **Diversion path analysis**: Define system elements containing nuclear material balance areas and key measurement points, and identify and evaluate technical barriers to mitigate either the diversion of nuclear material or the potential misuse of the facility to produce undeclared nuclear materials.
(2) **Safeguardability analysis**: Determine whether the nuclear facility would meet national and international nuclear safeguards requirements (i.e. based on detection goal quantity, timeliness for detecting a diversion and net probability for detecting a potential diversion).
(3) **Project risk analysis**: Identify potential 'hot-spots' in the design (i.e. locations where the ability to meet requirements with current technology is in question or at risk).
(4) **Cost and trade-off analysis**: Determine design, construction, operating costs and costs for implementing proposed safeguards approaches and facility or process design options.

(5) **Analysis of MUF**: Model the performance of the safeguards measures prior to the completion of the facility design to determine whether the proposed safeguards measures and system would meet domestic and international requirements (i.e. the requisite accuracy and performance to detect the diversion of significant quantities of nuclear material).

The FSA procedure will address the misuse and diversion pathway analysis and involves:

(a) Subdividing the nuclear facility into system elements;
(b) Identifying potential targets (nuclear material and processes to be protected) within each element;
(c) Evaluating the projected costs of necessary safeguards measures;
(d) Identifying and evaluating all potential sequences of events (pathway analysis) that could result in successful misuse or diversion of special nuclear material from the facility;
(e) Determining the values of the safeguardability measures (defined below) for each element of each pathway;
(f) Preparing a report describing the results of the study.

This process could be qualitatively and quantitatively addressed using robust system modelling tools. However, a detailed quantitative analysis can be lengthy and time consuming. In the proposed SBD process, a need exists for early and timely evaluation of the proposed design (intrinsic features) and related safeguards (extrinsic measures) of a nuclear energy system. To support a typical request for such information, this FSA procedure describes an approach for employing subject matter experts to perform the analysis to the level of detail desired. This process will provide both the high quality, technically supported information necessary to complete task 1 of the FSA and the basis for completing tasks 2–5.

## I.3. SELECTION OF EXPERTS FOR ELICITATION

When selecting team members for an SBD expert elicitation process, the team will be assembled in conjunction with the project design team, and the SBD experts considered an integral element of the design team. The following three questions are to be addressed.

### I.3.1. Who is considered an expert?

An expert is a person especially knowledgeable and experienced in the field of interest at a level of detail sufficient for reliably providing solicited data. For example, for a particular reactor design or refuelling process, the expert may be familiar with the reactor physics or the process chemistry, but may not be knowledgeable about specific operational logistics related to moving material or safeguarding it. Similarly, an expert on international safeguards may have only a very general understanding of the details of a particular reactor design, enrichment or reprocessing system. Experts will be chosen based on the types of question and the level of detail they will be expected to address.

### I.3.2. What kinds of experts are required to provide relevant data?

Three types of experts may be used when performing an expert elicitation informed safeguardability assessment, as characterized by their specific areas of knowledge and experience:

(1) The first type is a nuclear energy subject matter expert with technical knowledge about, or experience with, design, construction and operation of a particular nuclear energy system or systems. This may include experience in reactor technology and operations, fuel reprocessing or enrichment systems. Note that nuclear energy subject matter experts are not necessarily experts in safeguardability.
(2) The second type is an expert in international safeguards who is knowledgeable about international threats and possible means for misuse or diversion from nuclear energy systems. These experts might only be generalists in the area of nuclear energy systems or safeguardability assessment.
(3) The third type is an expert in implementing the methodology for proliferation resistance and safeguardability assessments.

In some cases, an expert may legitimately represent more than one of these groups.

### I.3.3. When is it appropriate to use an expert elicitation to assess facility safeguardability?

Expert elicitation is a tool used to solicit information from experts in situations where there is insufficient data, when such data would be too difficult to collect with traditional research methods, or when a need exists to perform a high level qualitative assessment in a timely fashion. The aforementioned attributes of expertise (i.e. knowledgeable at the necessary level and experienced in the right area) are important for gathering the appropriate information and reaching acceptable and usable conclusions.

This section describes the safeguardability analysis process intended to provide answers to the following set of fundamental questions. The answers to these questions will influence the design of the facility safeguards system, and will complement the design progression of the nuclear facility and process:

(a) What are the relevant facility safeguards requirements?
(b) What are the risks to the project of not meeting these requirements?
(c) What are the potential nuclear material diversion or misuse pathways in the facility or process?
(d) Which barriers prevent theft, diversion or misuse of nuclear material?
(e) How can the facility or process be designed to enhance barriers to misuse or diversion?
(f) If the facility design is altered to minimize the risk of nuclear material diversion, what are the associated trade-offs (e.g. reduction in operating efficiency or increased cost)?
(g) Can the design of the safeguards system support domestic and international safeguards?
(h) Can the facility design be optimized so that the objectives of the facility designer, owner or operator, national nuclear regulator and the IAEA can be met simultaneously in a cost effective manner?

It is imperative that the location and installation of safeguards equipment be anticipated early in the design process. If not, the requisite safeguards measurement accuracy, and hence the safeguards objectives, may not be attainable. In this worst case, the facility may be found to be ‘unsafeguardable’. For this reason, the facility designer and project design team need a tool to answer the questions posed above.

The pathway analysis will address how material can be diverted or how the facility can be misused. In order to perform the analysis, the nuclear energy system being assessed needs first to be defined. It can be described in the task statement prior to the elicitation process or can be modelled during the elicitation process. The level of detail required will be dictated by the specific problem to be addressed and the state of the facility design.

The nuclear energy system will be subdivided, as appropriate, into system elements, which could be a facility, part of a facility or a collection of facilities. If possible, a suitable level of detail for these elements would include a definition of material balance areas for the nuclear energy system or subsystem of interest. If the system elements are not yet defined at this stage, they could be developed later at a high level as part of the elicitation process. Key information to be defined or developed includes:

- Material types that exist or can exist within a system element;
- Required safeguards for material existing or potentially existing in the system element;
- Operations that can occur in a system element and identification of how or if these operations can be modified;
- Material movement that normally occurs into and from a system element.

With these considerations in mind, when potential misuse and diversion targets are being identified, a number of factors need to be evaluated as described in the following sections. The first important task in these analyses will be to identify and to describe material that exists or could exist in the facility, because this will define the safeguards requirements.

A diagram, or a set of diagrams, is required — prepared either prior to the elicitation or as a product of this step — that displays the nuclear energy system under consideration, delineates its elements, and either describes or references the equipment, operations, material and existing or planned safeguards relevant to each element. This diagram may evolve and become more detailed as the analysis continues. This diagram (or diagrams) will be part of the analysis documentation.

## I.4. TARGET ANALYSIS

The target analysis is performed using an elicitation process that involves collecting or developing target related information for a specific nuclear energy system or subsystem. A target analysis is performed for each system element of interest. Potential targets are materials that can be misused or diverted for use in a nuclear explosive device. Targets are identified considering a number of features: Table 1 addresses target material factors; Table 2 considers facility factors; and Table 3 describes safeguards considerations. As experts perform the target analysis, these factors are to be included.

TABLE 1. MATERIAL FACTORS

| Factor | Considerations |
|---|---|
| Chemical | Degree of difficulty in separating or extracting weapons material |
| Detectability | Passive detection capability<br>Active detection capability<br>Hardness of radiation signature<br>Uniqueness of material signature<br>Uncertainties in detection equipment |
| Isotopic | Critical mass<br>Degree of isotopic enrichment<br>Difficulty presented by radiation to weapons design<br>Heat generation rate<br>Spontaneous neutron generation |
| Mass and bulk | Concentration of material<br>Ease of concealment |
| Misuse potential | Ability to transform the material to make it more attractive for use in a nuclear explosive device<br>Ability to introduce fertile material into the system to function as a target |
| Possible radiological dose | Amount of remote handling required |

TABLE 2. FACILITY FACTORS

| Factor | Considerations |
|---|---|
| Available mass | Amount of potentially weapons-usable material at a given point |
| Diversion detectability | Form of material<br>Detection equipment uncertainty<br>Type of material and processes with respect to accountability |
| Facility accessibility | Difficulty and time to perform operations<br>Frequency of operational opportunity to divert<br>Manual, automatic and remote operations<br>Specialized equipment needed |
| Potential for facility misuse | Processes, materials or equipment that could be modified, adjusted, substituted or otherwise changed such that weapons-usable material, or precursors to weapons-usable material, could be produced or modified |
| Time material available[a] | Time materials in a particular area are available for diversion or misuse |

[a] The time when a specific fissile material could be left unattended/uncontrolled or at risk of being diverted and/or stolen.

TABLE 3. SAFEGUARDABILITY ISSUES

| Considerations |
| --- |
| Ability of safeguards systems to detect illicit activities |
| Availability and access of safeguards inspectors to process information |
| Adequacy of containment and surveillance systems to detect diversion or misuse |
| Degree of safeguards incorporation into process design and operation |
| Minimum detectability limits for material in system elements |
| Precision and frequency of monitoring |
| Response time of detectors and monitors |
| Time frames when IAEA inspectors are on-site |

An important part of the target analysis is to use the information described by the IAEA material categories to define the safeguards requirements for the system element (see Table 9, Section I.5.1). All system elements in the nuclear energy system may be assessed, or the expert team may select specific system elements to be evaluated. In any event, it is important that the reasons for this selection be recorded.

The purpose of the target analysis is to assess each element of the nuclear energy system (i.e. each system element) for potential target material, potential diversion points and potential for facility misuse to produce undeclared target material. Special emphasis is given to identifying physical locations in the system element (i.e. exit or diversion points) from which material could be removed. This assessment is performed without regard for a particular diversion or misuse strategy. Later, potential strategies, target information and relevant diversion points will be assembled in a pathway analysis. Key information developed in this step includes:

(a) Identification of potential diversion points from the system element;
(b) Identification of target material that could be diverted from the system element through possible diversion points;
(c) Identification of potential opportunities for misuse of the facility to produce undeclared nuclear material.

Table 4 provides an example of target analysis documentation that will be completed during the elicitation. Filling in the information is largely a consensus process. If there is disagreement about information developed in this analysis, dual or multiple tables reflecting more than one point of view can be established. The key information developed for this table includes:

— Potential diversion points from the system element;
— Material that could be diverted from the system element;
— Potential misuse strategies;
— The kind(s) of container(s) normally used for moving material into, or from, the system element;
— Safeguards that would potentially detect improper movement of material from a system element.

Assessment activities will include:

(1) Safeguards requirements, including monitoring, containment, surveillance and assay systems;
(2) Location, space and utility requirements for the safeguards components, including power and data transmission;
(3) Areas where changes to the nuclear facility design or process could potentially improve the safeguards system’s effectiveness;
(4) The expected MUF and limit of error for MUF for this facility and each system element.

TABLE 4. EXAMPLE TARGET ANALYSIS WORKSHEET: SYSTEM ELEMENT NO. 1

| Diversion points (exits) | SE-1 exit-1 | SE-1 exit-2 |
|---|---|---|
| Target ID | 1 | 2 |
| Target description | Cask of LWR fuel bundles | One or two LWR fuel bundles |
| Target material character | Irradiated U-235 and transuranic metal | Irradiated U-235 and transuranic metal |
| Containers | Casks | Casks |
| Container use location | Outside cask storage area | Transit between outside cask storage area and indoor spent fuel storage facility |
| Normal container material | Spent fuel elements | Spent fuel elements |
| Process | Storage | Unloading |
| Operational state | Normal storage | Cask movement |
| Safeguards | Cameras<br>Inventory | Cameras<br>Inventory |

**Note:** SE – system element; LWR – light water reactor.

Consideration is also to be given to operational modes. The team determines whether the key information or conclusions developed for a system element might vary based on operational modes.

Table 5 compiles information on potential targets, including an estimate of the potential for misuse of the nuclear energy system to produce direct use material. An overview of the summary table of targets and potential diversion points is shown in Table 6.

TABLE 5. NUCLEAR ENERGY SYSTEM TARGET LIST

| Target ID | Target description | Target material character | IAEA material category | Misuse possible |
|---|---|---|---|---|
| 1 | Cask of LWR fuel bundles | Irradiated U-235 and transuranic elements | Direct use | No |
| 2 | LWR fuel bundle(s) | Irradiated U-235 and transuranic elements | Direct use | Yes |

**Note:** LWR — light water reactor.

TABLE 6. TARGETS AND DIVERSION POINTS

| Target ID | Diversion points (exits) | Pathways |
|---|---|---|
| 1 | SE-1 exit 1 | SE-1-1A |
| | SE-2 exit 1 | SE-2-1A |
| 2 | SE-1 exit 2 | SE-1-2A |

**Note:** SE — system element.

## I.5. PATHWAY ANALYSIS

The pathway analysis is performed as an elicitation process whose goal is to identify potential diversion or misuse pathways based on possible strategies and related targets. Using the elicitation process, these pathways are developed for selected potential diversion points from a specific nuclear energy system or system element.

The target analysis assesses possible diversion points with potential for diverting a target of interest. The pathway analysis identifies and describes the set of physical actions that are to be taken. The information required for the pathway analysis includes:

— Potential diversion points (see Table 4);
— Description of potential targets (see Table 5);
— Summary table of targets and potential diversion points (see Table 6);
— Potential strategies (e.g. abrupt diversion, protracted diversion or misuse followed by diversion).

The results of the pathway analysis identify the actions required for successful misuse or diversion and relevant descriptions of potential pathways (see Tables 7 and 8 for examples).

TABLE 7. EXAMPLE PATHWAY ANALYSIS WORKSHEET

| Target ID | Target description | Diversion points | Potential strategies | Pathway ID | Key pathway events | Pathway description |
|---|---|---|---|---|---|---|
| T1 | Cask of LWR fuel bundles | SE-1 exit 1 | Abrupt diversion | SE-1-1A | Use heavy trucks and trailers to move casks<br>Fool or disable cameras<br>Compromise inventory measurement records | Cask of LWR spent fuel bundles is in the LWR cask parking lot<br>Camera is compromised<br>Diverter takes cask and hauls away to concealed processing facility<br>Key measuring point controls are compromised |
| | | SE-2 exit 1 | Abrupt diversion | SE-2-1A | Send back a loaded cask instead of an empty cask<br>Use heavy truck and trailer to move cask<br>Fool or disable the camera<br>Compromise the inventory measurement records | A full cask of LWR from the spent fuel facility is sent back instead of an empty one<br>Camera is compromised<br>Diverter takes cask and hauls away to concealed processing facility<br>Key measuring point and transfer measuring point controls are compromised |

**Note:** LWR — light water reactor; SE — system element.

TABLE 8. EXAMPLE PATHWAY ANALYSIS WORKSHEET FOR MISUSE

| Target ID | Target description | Diversion points | Potential strategies | Pathway ID | Misuse actions | Pathway description |
|---|---|---|---|---|---|---|
| T4 | Waste containing transuranic elements from electro-refiner process | SE-3 exit 1 | Protracted misuse and diversion combined | SE-3-1M | Electro-refiner modified to increase transuranic content of waste<br>Site receives waste container, does not send to established and controlled waste storage location<br>Compromise inventory measurement records | Electro-refiner modified to increase transuranic content of waste<br>Actor compromises the material inventory system, collects normal transuranic elements via waste container and sends to concealed facility over a period of time |

**Note:** SE — system element.

Actions required for success include not only physical movements of material but all ‘enabling actions’ that need to be performed. Enabling actions include compromising or disabling detection systems, compromising movement records and falsifying inventories. As shown in Tables 7 and 8, pathway descriptions include:

(a) Identification and description of the target;
(b) Identification of the diversion point and the strategy considered (e.g. misuse or diversion);
(c) Identification of key pathway events, including enabling actions to compromise or disable facility safeguards and detection systems, compromising movement records or falsifying inventories;
(d) Description of the diversion or misuse scenario (i.e. all actions that need to be performed to gain the material of interest).

Each pathway is assessed, and experts determine the appropriate value of each measure (i.e. fissile material type, detection resources, detection probability, technical difficulty and diversion time) for that pathway. The quantification will be performed as an elicitation process whose goal is to determine values for each measure. Treatment of uncertainty is critical, and experts need to state explicitly the uncertainty they assign to each ranking. If the experts cannot agree (i.e. reach consensus on a category assignment), then a description of the disagreement will be included in the assessment report, along with a discussion of the overall implications on the report conclusions.

The measures used for determining safeguardability in this task include:

(1) **Fissile material type**: A categorization of material existing in the facility based on IAEA safeguards definitions.
(2) **Detection resources**: The cost related to staffing, equipment and funding required to apply international safeguards to the nuclear energy system. This measure also includes the cost of any intrinsic features created, or changed, to meet safeguards goals.
(3) **Detection probability**: The cumulative probability of detecting diversion or misuse along a pathway.
(4) **Technical difficulty**: The inherent difficulty, arising from the need for technical sophistication and materials handling capabilities, required to overcome the barriers to diversion or misuse.[4]
(5) **Misuse and diversion time**: The minimum time required to overcome the safeguards barriers (i.e. the total time planned by the proliferator). For safeguardability, this would be the elapsed time required for successful diversion or misuse.

[4] Barriers refer to intrinsic barriers (e.g. technical difficulty) and extrinsic barriers (e.g. safeguards).

The experts will use the information developed during the previous steps as the basis for their final elicitations, as outlined in Sections I.5.1–I.5.5. When the quantification process is complete for each pathway, the results will be assembled in a final elicitation process. The results are used to ensure that the facility will meet national and international nuclear safeguards requirements (i.e. based on detection goal quantity, timeliness for detecting a diversion and net probability for detecting a potential diversion). The results will provide a basis for evaluating and comparing the interaction of proposed intrinsic features and related safeguards.

### I.5.1. Fissile material type

Fissile material existing in each system element will be identified early in the pathway analysis (see Table 4, in Section I.4), as this is an important element in defining potential pathways. As each system element is evaluated in more detail, misuse scenarios may be identified, the resulting undeclared material characterized and relevant pathways developed.

The IAEA defines two classes of materials that require safeguards (see Table 9). The existence or potential existence of these materials (either as part of normal operation or as a result of facility misuse) in a nuclear energy system will categorize the level of safeguards required and will be the basis for an initial estimate of safeguards cost. Accordingly, this is a key element of the pathway analysis.

TABLE 9. IAEA MATERIAL CATEGORIES, RELATED ISOTOPES AND SIGNIFICANT QUANTITIES

| IAEA material category | Materials | Significant quantity |
|---|---|---|
| Direct use nuclear material | Pu[a] | 8 kg |
| | HEU (U-235 ≥ 20%) | 25 kg |
| | U-233 | 8 kg |
| Indirect use nuclear material | LEU (U-235 < 20%) | 75 kg |
| | Natural U | 10 t |
| | Depleted U | 20 t |
| | Th | 20 t |

**Note:** HEU – high enriched uranium; LEU – low enriched uranium.
[a] For plutonium containing less than 80% $^{238}$Pu.

### I.5.2. Detection resources

Detection resources are defined as the labour, equipment and funding required to meet international safeguards objectives for the nuclear energy system under consideration. These objectives are to detect:

(a) Diversion of 8 kg of plutonium in the form of unirradiated fresh plutonium dioxide or MOX fuel within one month of possible diversion or, in the form of irradiated core or spent fuel, within three months of possible diversion;
(b) Diversion of 25 kg of $^{235}$U in the form of unirradiated HEU ($^{235}$U ≥ 20%) within one month of possible diversion or, in the form of irradiated core or spent fuel, within three months of possible diversion;
(c) Diversion of 8 kg of $^{233}$U, produced by breeding from thorium, in the form of irradiated fuel within three months of possible diversion or separated $^{233}$U within one month of possible diversion;
(d) Diversion of 75 kg of $^{235}$U in the form of depleted, natural or low enriched uranium ($^{235}$U < 20%) within one year of possible diversion;
(e) Diversion of 20 t of thorium within one year of possible diversion;
(f) Possible facility misuse for undeclared irradiation and production of 8 kg or more of plutonium or $^{233}$U, or other undeclared nuclear activities within one year.

The safeguards system that detects misuse or diversion of nuclear material from a nuclear energy system relies upon two basic elements:

(1) Nuclear material accountancy;
(2) Containment and surveillance.

The costs of establishing these safeguards are dependent on a number of factors, and expert elicitation is a key to determining these costs at new facilities. To support the process and to help to ensure consistency in these evaluations, a reference table of cost estimates for implementing IAEA safeguards measures, in both US dollars and human effort (person days of inspection effort), needs to be developed. Tables 10 and 11 are to be filled out, thereby defining the nuclear material accountancy and containment and surveillance safeguards measures established for each system element. As part of this iterative process, the elicitation team will review intrinsic features of the facility design to evaluate the effects of alternate designs and facility layout on the overall cost of detection resources.

TABLE 10. EXAMPLE NUCLEAR MATERIAL ACCOUNTANCY FEATURES

| System element | Nuclear material accountancy features | Estimated accountancy feature costs |
|---|---|---|
| A. System element inventory | | |
| B. Production records | | |
| Use and production of nuclear material in system element | | |
| Transfers into and from facility | | |
| C. SSAC reports | | |
| D. IAEA inspector visits | | |

**Note:** SSAC – State system of accounting for and control of nuclear material.

TABLE 11. EXAMPLE FACILITY CONTAINMENT AND SURVEILLANCE FEATURES

| System element | Containment and surveillance features | | | |
|---|---|---|---|---|
| | Location | Purpose | Installation cost | Operating and maintenance cost |
| Cameras | | | | |
| Seals | | | | |
| Item counters | | | | |
| Scene change detector | | | | |
| Portal monitors | | | | |
| Movement detectors | | | | |
| Radiation detectors | | | | |
| Crane monitors | | | | |
| Sample analysis | | | | |
| Other instruments and detectors | | | | |

### I.5.3. Detection probability

Detection probability is defined as the cumulative probability of detecting the occurrence of misuse or diversion during a segment or pathway. Section I.5.2 shows the IAEA safeguards detection requirements for the various material types. The detection probabilities specified to meet these requirements in these time frames can vary depending on the facility and material. The goal for the safeguards system could be, for example, a 90% chance of detection during the specified time period, with a 5% chance of a false positive.

Experts need to determine whether these goals can be met within the existing safeguards system. The experts will be polled on the likelihood of pathway detection using the information provided or developed and documented in the pathway analysis (see Tables 7 and 8 concerning actions, Table 9 for the material being misused or diverted, and Tables 10 and 11 for facility safeguards and specified facility configurations, respectively).

The value and the logic behind the reasoning will be included in Table 12. It is important that the experts consider the uncertainty involved in determining pathway detection probability. Any calculations done to determine the detection probability need to be incorporated into supporting documentation and referenced in the 'Reasoning and comments' column of Table 12.

TABLE 12. PATHWAY DETECTION PROBABILITIES

| Pathway ID | Detection probability | Comparison with detection goal | Reasoning and comments |
|---|---|---|---|
| | | | |

### I.5.4. Technical difficulty

Technical difficulty is defined as the inherent difficulty, arising from the need for technical sophistication and materials handling capabilities, required to overcome multiple safeguards barriers — both intrinsic features and extrinsic measures.

Experts need to determine the difficulty of performing the acts described in the pathway for misuse and diversion of material. Specifically, the experts are to review:

(a) Actions described in the pathway analysis tables (Tables 7 and 8);
(b) Material to be misused or diverted (Tables 1, 7 and 8);
(c) Facility factors (Table 2);
(d) Safeguards systems (Tables 3, 10 and 11).

The experts will determine the technical feasibility of the proposed actions, which will fall into three categories:

(1) The proposed pathway is credible, the necessary support technology exists (e.g. target manufacturing and support for concealment), and the actions required are within the resources and capabilities of a typical State.
(2) The proposed pathway may be feasible, but successful accomplishment would require extraordinary support technology.
(3) The proposed pathway is not credible.

Only pathways that fall into the first category will be considered further in these safeguardability studies. However, Table 13 is to be filled out completely and maintained for all pathways, as the pathway feasibility may need to be reconsidered as the facility design matures or is modified.

TABLE 13. PROLIFERATION TECHNICAL DIFFICULTY

| Pathway ID | Feasibility (1, 2, 3) | Discussion and reasoning |
| --- | --- | --- |
| | | |

### I.5.5. Misuse and diversion time

Misuse and diversion time is defined as the minimum time required to acquire at least one significant quantity (i.e. the total time planned by the proliferator for the project). Experts need to determine the minimum time necessary to perform successfully the planned misuse or diversion for the strategy defined by each pathway. This would include any necessary facility modifications and the subsequent time required to perform the actions necessary to divert the target material. The experts also need to consider the detection time requirements for various nuclear materials (Section I.5.2) and determine whether the actions in the planned pathway could be performed within those time frames (for either an abrupt or protracted diversion).

In addition, the experts will review the actions described in the pathway analysis tables (e.g. Tables 7 and 8, in Section I.5) and determine the time required for such actions. The value and logic behind the reasoning is to be included in Table 14.

TABLE 14. MISUSE AND DIVERSION TIME

| Pathway ID | Misuse/diversion time | IAEA required material detection time | Discussion and reasoning |
| --- | --- | --- | --- |
| | | | |

## I.6. SAFEGUARDABILITY CHARACTERIZATION

Because the safeguardability measures described above are not mutually independent, and their degree of interdependence may vary from facility to facility, it is not enough simply to characterize the safeguardability of a nuclear energy system or subsystem in terms of the measures described above.

In addition, the significance of each safeguardability measure may vary from problem to problem. Therefore, the measures are not considered to form the safeguardability characterization themselves. Rather, they provide the basis for completing the safeguardability analysis.

The results of this work will form the basis for the proposed safeguards analysis report (SGAR) (as described in Ref. [20]). The information developed during the safeguards characterization described above will provide information to be used in the SGAR, including:

- — Relevant safeguards requirements;
- — Relevant facility design information;
- — Preliminary diversion path analysis and subsequent iterations;
- — Prospective safeguards approach proposed by the designer;
- — Safeguards system design;
- — Trade-off analyses relevant to the optimization of the safeguards system and facility design.

## I.7. REPORT PREPARATION

The results of this safeguardability analysis will be used to [20]:

(a) Identify and confirm the requisite safeguards measures to help to provide the basis for the facility safeguards system;
(b) Identify and evaluate the functional requirements for the specific safeguards measures and system (i.e. safeguards monitoring, containment, surveillance and assay systems);
(c) Identify the location, space and utility requirements for the safeguards components, including power and data transmission;
(d) Identify areas where changes to the nuclear facility design or process could improve the effectiveness of the safeguards system.

Table 15 provides an example of a table of contents for a report describing this work. This report will go through several iterations as the design matures.

TABLE 15. EXAMPLE TABLE OF CONTENTS

1. RELEVANT FACILITY DESIGN INFORMATION
2. RELEVANT SAFEGUARDS REQUIREMENTS
3. PROSPECTIVE SAFEGUARDS APPROACH PROPOSED BY THE DESIGNER
4. PRELIMINARY DIVERSION PATH ANALYSIS AND SUBSEQUENT ITERATIONS
5. SAFEGUARDS SYSTEM DESIGN
   5.1. MUF analysis
   5.2. Optimization of the facility and process design (as it pertains to safeguards)
   5.3. Trade-off analyses relevant to the optimization of the safeguards system and facility design
6. EVALUATION OF THE EXPECTED PERFORMANCE OF THE SAFEGUARDS SYSTEM, AS VALIDATED BY CONFIRMATORY TESTING AND CALIBRATION OF THE SAFEGUARDS EQUIPMENT AND COMPONENTS DURING COMMISSIONING

APPENDIX I: EXPERT ELICITATION PROCESS DESCRIPTION

I.1. Introduction
I.2. Expert elicitation process summary
I.3. Safeguardability assessment methodology summary
I.4. Nuclear energy system description
   I.4.1. Energy system configuration
   I.4.2. Process description
   I.4.3. System elements
   I.4.4. Safeguards summary
I.5. Pathway analysis
   I.5.1. Diversion, misuse and concealed facility
   I.5.2. Target identification
   I.5.3. Diversion point identification
   I.5.4. Pathway identification
   I.5.5. Analysis description

TABLE 15. EXAMPLE TABLE OF CONTENTS (cont.)

## Appendix II

## TEMPLATE OF INFORMATION REQUIREMENTS

### [Reactor Design Name] template listing required proliferation resistance related design information

Date

Authors

### II.1. OVERVIEW OF TECHNOLOGY

This section provides a general overview of information required to perform proliferation resistance evaluations. Detailed technology summaries are to be referenced if available.

Key reactor parameters such as power, coolant, moderator (if any), power density values, fuel materials (this could be covered under fuel cycle information), inlet and outlet conditions, coolant pressure, neutron energy spectrum, fuel enrichment level and refuelling scheme (continuous, batch or battery-type) are to be included here. Table 16 provides an example of information to be included. More detailed descriptions of required information are provided in the following section.

TABLE 16. TEMPLATE OF INFORMATION ON REQUIREMENTS

| System element | Element description | Relevant information provided by designers |
|---|---|---|
| I. Reactor description | Reactor name | |
| | Reactor type (e.g. fast, thermal, pool, loop, pressure tube or block) | |
| | Moderator (e.g. water, heavy water, graphite or none) | |
| | Coolant type (gas, water, metal or salt) | |
| | Operating temperatures and pressures | |
| | Core configuration (e.g. blanket or extra space for targets) | |
| | Special nuclear materials production potential (neutron spectrum, target locations and consequence of normal operation) | |
| | Core diameter | |
| | Core height | |
| | System enclosure (e.g. containment, confinement, underground or floating) | |
| | Physical arrangements as they affect access control for material and vital equipment | |
| | Safety design (active or passive) to mitigate consequences from sabotage | |
| | Plant security layout (as available) | |
| | Details of safeguards layout | |
| | Fresh fuel storage (amount and location) | |
| | Spent fuel storage (amount, location and time on-site) | |
| | Purpose (e.g. electricity generation, process heat, coal liquefaction, desalination, oil recovery or hydrogen production) | |
| | Thermal cycle (e.g. PWR, BWR, direct cycle helium or metal to water) | |

TABLE 16. TEMPLATE OF INFORMATION ON REQUIREMENTS (cont.)

| System element | Element description | Relevant information provided by designers |
|---|---|---|
| II. Reactor power | Reactor power MW(th) | |
| | Reactor power MW(e) | |
| | Cycle length | |
| | Refuelling scheme (batch, continuous or partial core) | |
| III. Fuel data | Fuel type (particles, rods, blocks or liquid) | |
| | Fuel description (ceramic, MOX, metal, nitride, dispersion fuel or tristructural isotropic fuel) | |
| | Enrichment (plutonium or uranium) | |
| | No. of fuel items (e.g. assemblies, tubes or pebbles) in the core | |
| | Weight of fuel items | |
| | Discharge burnup (GWd/t) | |
| | No. of fuel items replaced per cycle | |
| | Active fuel size or length | |
| IV. Potential target information | Mass of heavy metal per fresh fuel assembly | |
| | No. of fresh fuel items to constitute one significant quantity | |
| | No. of spent fuel items to constitute one significant quantity | |
| | Can fuel items be modified to allow misuse? | |
| V. In-country fuel cycle information | Fuel recycling and reprocessing facilities | |
| | Enrichment facilities | |
| | Fresh and spent fuel transportation | |
| | Off-site spent fuel storage | |
| | Facility waste streams | |
| | Facility waste storage | |
| VI. Proliferation resistance features incorporated into design | Potential targets for diversion | |
| | Potential targets for misuse and undeclared production | |

**Note:** PWR — pressurized water reactor; BWR — boiling water reactor; MOX — mixed oxide.

Design information that is particularly important to assess proliferation resistance will include:

(a) Potential fuel types (including high level characteristics of fresh and spent fuel);
(b) On-site fuel storage and transport methods;
(c) Approach for ensuring safety and all associated vital equipment (for confinement of radioactivity and other hazards, reactivity control, decay heat removal and exclusion of external events);
(d) Physical arrangement approach associated with access control and material accounting for fuel (a potential theft target);
(e) Access control to vital equipment (a potential sabotage target) — this information will be described at a very high level.

## II.2. REQUIRED DESIGN INFORMATION

The required design information includes a description of the types of fuel cycle being considered for the SMR system under consideration. If possible, a name is to be provided for each major category of the system being considered, so that they can be distinguished. A material flow diagram is included, if available. The discussion ideally includes a description of major waste streams that might contain weapons-usable material or could be used to conceal the diversion of weapons-usable material. Table 16 can be used to summarize the data, and the following information will be included. Information which is currently unknown or incomplete needs to be clearly noted. If there is more than one qualitatively distinct option for any of these characteristics, this also needs to be stated (and related table entries will be modified accordingly).

(1) Reactor power (thermal and electrical, process heat and hydrogen);
(2) Fuel types considered;
(3) Form (e.g. ceramic, metallic or dispersion);
(4) Main fertile material (uranium vs. thorium);
(5) Fissile material;
(6) Enrichment (rough range);
(7) Source of fissile material (initial and reload cores);
(8) Fuel inventory (mass in-core and ex-core);
(9) Use of in-core fertile blankets;
(10) Use of in-core transmutation targets (e.g. for minor actinides or low level fission products);
(11) Discharge burnup;
(12) Refuelling frequency and mode (on-line vs. off-line);
(13) Recycling approach;
(14) Centralized versus collocated recycle facilities;
(15) Recycled constituents (plutonium only, plutonium and neptunium, or all transuranic elements);
(16) Homogeneous versus heterogeneous recycling of minor actinides or low level fission products;
(17) Recycling technology;
(18) Reprocessing technology (if any);
(19) Refabrication;
(20) Areas where material would be handled, providing perspectives to assess access control and material accounting and containment, and surveillance;
(21) Recycling efficiency (recovery fractions targeted for minor actinides, plutonium and uranium);
(22) Waste forms.

## II.3. CURRENT SYSTEM DEVELOPMENT STATUS

If available, timelines for key system development stages are to be included (e.g. a very high level Gantt chart or something similar). Status or plans on development of a safeguards approach and any pertinent information concerning domestic safeguards and associated physical protection schemes also need to be included.

## II.4. PROLIFERATION RESISTANCE CONSIDERATIONS INCORPORATED INTO DESIGN

These considerations provide a high level, qualitative overview on those elements of the system design that create potential benefits or issues related to possible proliferation threats. The considerations include any assumptions about international safeguards approaches related to this design concept. If necessary, any need for the development of a specific safeguards approach that would be unique to the concept under consideration is to be noted. Particular characteristics of the system related to item counting or bulk accountancy of materials are to be identified. Proposed material balance areas need to be tentatively identified, and issues related to continuity of knowledge and self-protection characteristics of irradiated fuel are to be discussed.

The considerations also address potential diversion and undeclared production targets (i.e. misuse of the facility) available within the system for each of the major fuel cycle variants.

## II.5. NEXT STEPS AND PATH FORWARD

The next steps and the path forward, to be developed jointly by the IAEA team and the participating State design teams, will review any outstanding issues related to proliferation resistance for each concept and related fuel cycles, the areas of known strength in the concept, and future plans for integration and assessment of proliferation resistance for the concept. This section will conclude with a bullet list of identified proliferation resistance R&D needs for the system concept.

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## Annex

## STANDPOINTS ON PROLIFERATION RESISTANCE FROM SMALL AND MEDIUM SIZED REACTOR DESIGNERS AND DEVELOPERS IN MEMBER STATES

### A–1. INDIA

#### A–1.1. Some intrinsic approaches to enhance safety, security and proliferation resistance: AHWR300-LEU

The growth of nuclear power worldwide requires satisfactory technological responses to the challenges of very high levels of safety and security (required for land based units close to population centres and transportable nuclear power plants), high degrees of fuel use efficiency and superior waste disposal options. To extend the global reach, technological approaches are required which provide an ability to perform with lower levels of technological infrastructure and which address proliferation concerns.

### A–2. ITALY

#### A–2.1. Remote monitoring: A key characteristic common to all systems

Remote monitoring is suitable for unattended and remotely controlled operation, safeguards efficiency, and savings and system complexity. It is used more and more frequently by the IAEA for safeguards. Considering the location of the innovative small and medium sized reactors (SMRs) (i.e. those that are difficult to access, are on remote islands or sparsely populated regions), remote monitoring is a key characteristic to enhance both intrinsic and extrinsic proliferation resistance and physical protection (PR&PP). An analysis of the PR&PP characteristics of more than 45 innovative SMRs shows that some of the designs consider remote monitoring as a proliferation resistance measure, the majority as a physical protection measure, and some do not identify their consideration (or approach). Although this does not necessarily mean that remote monitoring concepts are not going to be developed and introduced at a later stage, it shows that due attention has not always been given to it at the design stage.

The importance of remote monitoring for PR&PP needs to be emphasized to designers because remote monitoring will help to detect diversion, misuse and sabotage. It can be performed in key points of sensitive processes, in sensitive instruments or components, in sensitive areas and generally in all parts that the PR&PP experts, with the help of designers, can identify as critical for diversion, misuse or sabotage. Remote monitoring makes use of cameras, instruments, components or seals, and it could be the monitoring of physical parameters that indicate diversion, misuse or sabotage. From a neutron designer point of view, the core neutron design is the result of an optimization process based on the knowledge of materials behaviour and the application of physical laws within defined design constraints. Because any optimization freezes the neutron design, PR&PP improvement by design at the core level needs to be based on a clear and self-consistent set of PR&PP requirements (design constraints) given at a very early stage in order to be feasible and efficient.

#### A–2.2. Comments and expectations from the core designer point of view

While waiting for information concerning format and topics in the template collecting the designer’s data, PR&PP experts and core designers need to work together to understand, identify and eventually propose new physical constraints. The scenario studies, including fuel cycle development, could become an interesting tool to improve PR&PP. The results of assessment methodology, weakness identification and resolution, and nuclear power plant site specific conditions, among others, is a series of design constraints for PR&PP at the design level. It is important that PR&PP improvement by design also involves education and training at national or international levels.

### A–2.3. PR&PP by design improvement: Scenario studies contribution

Scenario studies, while not providing all the information for detailed proliferation resistance evaluations, are an interesting tool to improve proliferation resistance features (in the early design phase). Proliferation resistance intrinsic feature improvement can be achieved through breeding gain modulation, burnup rate, isotopic detailed fuel composition, fissile materials significant quantity inventory, nuclear material attractiveness, heating rate, radiation dose rate and other information. Proliferation resistance extrinsic feature improvement can be achieved through the whole fuel cycle development, including operative conditions of enrichment, fabrication and reprocessing plants, interim storage, other stocks and waste management. Proliferation resistance performance comparison between different systems is feasible only at the same ‘boundary’ conditions. Proliferation resistance feature improvement can be achieved through an optimized strategy between different options concerning design and operation conditions of all the scenario components.

## A–3. RUSSIAN FEDERATION

### A–3.1. Analysis of non-proliferation characteristics of small and medium power innovative nuclear power plant designs

According to the IAEA non-proliferation recommendations, the basic principle of organizing a fuel cycle for exported reactors is to ensure that the importing State receives fresh nuclear fuel for reactors directly from the exporting State or through a fuel bank under IAEA control. The spent fuel is returned to the exporting State in accordance with international agreements and under IAEA control. This approach enables the elimination of proliferation potentials at such fuel cycle stages as fuel enrichment, reprocessing and final burial. For reactor systems with on-site refuelling (comparatively large systems such as the VBER-300 and GT-MHR, a joint US–Russian project), nuclear fuel is transported in the form of ready to use fuel assemblies. For reactor systems without on-site refuelling (such as the ABV-6 and KLT-20), nuclear fuel is transported inside the inoperative reactor core.

The methodology of non-proliferation analysis reflects the nuclear materials protection level provided by the nuclear system itself and the scale of assessment to be added. The methodology makes it possible to compare reactor systems from the viewpoint of their resistance to proliferation of nuclear materials at the stage of their use in the reactor. The analysis results show that the proliferation resistance of the KLT-40S reactor plant is close to the value determined for pressurized water reactor (PWR) plants, and that the proliferation resistance of the ABV reactor plant is significantly higher than for the PWR and is close to a maximum value, determined for the GT-MHR with low enriched uranium (LEU). From the viewpoint of proliferation resistance, the nuclear power plant option with reactors that operate without on-site refuelling is quite attractive. Proliferation resistance of these nuclear power plants is estimated to be very high. It is important to develop some ranking for consideration of proliferation resistance characteristics according to their significance, in order to arrive to some aggregated parameters of proliferation resistance.

### A–3.2. Future prospects of small and medium sized nuclear power plants

Coated particles are suggested as a new elemental base for atomic energy, especially for SMRs. Their physical and chemical features are unique. They can retain the fission products at a temperature up to 1600°C in steam media; the surface for heat transfer is ten times larger than fuel rods; and the margin to the heat transfer crisis is 20. When the coated particles are directly cooled, the average time of heat transfer from the fuel to the coolant is approximately 0.03 s. This feature provides virtually instant compensation of any positive reactivity at the expense of coolant moderator evaporation. In light water reactors (LWRs), where the coolant is a neutron moderator, the use of coated particles enables the creation of reactors with real inherent safety. The LWRs then have determined safety levels when the possibility of essential fission product exit is eliminated for any serious accidents, including reactor vessel destruction, aeroplane crashes, or any acts of terrorism and hostile actions. The heat hydraulic features of coated particles based fuel assemblies with cross-sectional coolant flow provide a maximum enthalpy increment and a maximum steam content in the centre. Such features allow the use of coolant boiling in fresh fuel assemblies for effective spectral control (compensation of reactivity charge for burnup). Simultaneously, it is possible to ensure that the average temperature of the mixed coolant at the outlet of the reactor core is equal to the saturation temperature (i.e. it is possible to use the potential of a water coolant at a maximum degree and to increase the efficiency to 39%). The features of a new elemental base allow essential decreases of capital expenditures and fuel factors. Experiments demonstrate the possibility of coated particle use as a new elemental base while achieving unique technical, economical and safety features. The Russian Federation has the technological processes and possibilities for coated particles production.

## A–4. UNITED STATES OF AMERICA

### A–4.1. Technical approach to enhancement of proliferation resistance of nuclear power plants with innovative small and medium sized reactors

The science based approach to PR&PP of the Generation IV International Forum (GIF) methodology first deals with challenges, then moves to system response and concludes with outcomes. Evaluations consider the State context (objectives, capabilities and strategies), system design features relevant to PR&PP, safeguards and security contexts, and the three stages of acquisition, processing and weaponization. The system response is described by pathways (i.e. potential sequences of events or actions followed by the proliferant State or adversary to achieve its objective). A pathway is composed of segments that can be action, target or system elements, and these segments can be internal or external to the system being assessed. Multiple pathways show that there are no simple answers, owing to the fact that results are sensitive to underlying assumptions about existing capabilities and objectives of adversaries, and they validate the decision not to roll up analysis into a single figure of merit (i.e. a single acceptance or success criterion). Owing to the interface of intrinsic and extrinsic measures, safeguardability is a key consideration. Evaluations provide valuable feedback to system design, according to types of targets being created and physical arrangements. Possible next steps for performing PR&PP evaluations of SMRs would underline the need to define questions to be answered, system design information, safeguards context, fuel cycle architecture and a reference (baseline) for comparison. It is equally important to assemble teams consisting of designers, safeguarders and analysts, conduct evaluations, and also include safety and economic evaluations. Overall, it is the insight gained from the disciplined process of performing the evaluation that is of value, and not just the final results. The benefits of PR&PP evaluations need to be sought early in the design process of SMRs.

### A–4.2. A qualitative assessment of diversion scenarios for an Example Sodium Fast Reactor using the GEN IV PR&PP methodology

There are three stages in proliferation resistance pathway analysis: acquisition, processing and fabrication. The acquisition stage is evaluated in detail in the assessment; the processing stage is evaluated at a high level; while fabrication is not considered. With regard to the proliferation resistance analysis process overview, the challenges (threats) are first analysed, then the system response (proliferation resistance assessment) is analysed and finally the outcomes (measures) are considered. The challenge is the diversion threat and target selection. The system response to diversion can be pathway development, preliminary safeguards determination, pathway analysis and proliferation resistance determination. The threat description can be selected through the range of possibility and threat characteristics relevant to the diversion analysis limited by the current scope. Target identification begins by breaking the facility into system elements for analysis. The Diversion Qualitative Pathway Analysis begins with a review of every target in light of specific threats under consideration:

(a) Examine every potential target and evaluate the material type of the target;
(b) Identify the possible physical mechanisms that could be used to remove the material;
(c) Identify the physical and design barriers to removal;
(d) Identify the safeguards barriers that protect each physical mechanism;
(e) Hypothesize ways to defeat the safeguards;
(f) Outline qualitative pathways for removal of each target;
(g) Perform a coarse qualitative evaluation of the measures for each diversion pathway.

This methodology can be used to compare and distinguish the proliferation resistance of different design choices. It can provide useful information to authorities, officials and designers.

### A–4.3. The attractiveness of materials in advanced nuclear fuel cycles for various proliferation and theft scenarios

It is first necessary to distinguish between diversion and theft. Material attractiveness is one of several considerations when evaluating either proliferation resistance or robustness of a nuclear energy system. Proliferation resistance is the characteristic of a nuclear energy system which impedes the diversion of 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) is that characteristic of a nuclear energy system that impedes the theft of materials suitable for nuclear explosives or radiation dispersal devices and the sabotage of facilities and transportation by subnational entities and other non-host State adversaries.

There is a safeguards and security benefit with respect to safeguards to diluting the reprocessing end products with lanthanides or uranium (reprocessed, natural or depleted). However, there is no silver bullet to solve the safeguards and security issue. None of the proposed flowsheets examined to date justifies reducing international safeguards or physical security protection levels. All reprocessing products that have been evaluated need to be rigorously safeguarded and provided with the highest levels of physical protection.

## A–5. FRANCE

### A–5.1. Specific national requirements on small and medium sized reactors: proliferation resistance and physical protection aspects

Proliferation resistance is one of the issues in the global development of innovative SMR technology for immediate and near term deployment, especially because many SMR concepts incorporate advanced or innovative fuel cycle. Proliferation resistance needs to be ensured when developing any nuclear facility or project, especially when it includes innovative technology. However, the expected time frame for achieving deployment of the technology is in the immediate and near term. In addition, such first of a kind engineering technology will most likely expand in States embarking on a nuclear power programme. SMR projects will therefore have to turn to innovation but be based on existing sound experience.

The deployment of SMRs will have to abide by international instruments. The first is the Treaty on the Non-Proliferation of Nuclear Weapons (NPT), which applies to States Parties, and the related IAEA comprehensive safeguards agreements and additional protocol, for which France calls for prompt universalization. The vendor will have to comply with the nuclear suppliers' group directives on export control when making the decision to export an SMR or parts of it. The buyer will have to be in compliance with its international obligations under the NPT, which means that such a scheme cannot apply to States Parties that, for instance, are under United Nations Security Council sanctions. In addition, the buyer needs to have ratified the Convention on the Physical Protection of Nuclear Material. There will also be a commercial contract between the two lead companies involved and, in most cases, a bilateral agreement between the two States. Other treaties or conventions will apply, such as the Vienna Convention on Civil Liability for Nuclear Damage or the Convention on Nuclear Safety.

An immediate or near term deployment, notwithstanding its innovative aspects, will ideally rely on known concepts. PWRs represent an adequate response, so that nothing specific in the concept or anything additional need to be developed. The choice of LEU (around 5%), albeit based on economics since such fuel does exist off the shelf for nuclear power reactors, provides an interesting answer to non-proliferation issues. The technology of PWRs, implying the use of sealed reactor pressure vessels, provides additional guaranties with regard to non-diversion issues. The IAEA could also implement physical seals to ensure that a given vessel has not been opened between refuelling operations. A batch approach, by which the full core is replaced at once with relatively long reload cycles (typically 3–5 years), would also reinforce non-proliferation issues related to plutonium, especially since the necessary burnup rate would produce less plutonium and with a lower grade with regard to military applications. Physical protection could benefit from the relatively small size of SMRs. Based on land, SMRs could, for instance, be installed into metallic containments and buried under concrete roofs, which is not feasible for large PWRs.

Similarly to large power reactors, SMRs could also be installed on land. Owing to their size, off-shore solutions can also be considered for isolated coastal regions.

With regard to safeguards and physical protection, experience exists with irradiated fuel as well as for mixed oxide (MOX) fuel. This does not mean that it is envisaged to use MOX fuel within SMRs, but since the requirements for MOX fuel transportation are more stringent than for other types of nuclear fuel, it makes sense to build on experience gained under more demanding conditions. Hence, the case of transportation and loading of land based SMRs need not lead to unexpected difficulties. For offshore cases, there is the need for more innovative solutions. Two main options, depending on whether reloading is made on the spot or by transporting the SMRs to a dedicated reloading facility, will have to be envisaged, based on usage and site conditions (i.e. an SMR connected to a larger electric grid or to a sole provider in an isolated area, and distance to appropriate facilities). A comprehensive review is currently being conducted involving French nuclear stakeholders.

Taking into account that SMRs respond to a growing need for nuclear energy worldwide, and that they are suitable for States embarking on a nuclear power programme with smaller grid size networks or isolated areas, it is considered that SMRs need to capitalize on existing experience in operating Generation III reactors and to reflect on lessons learned from the Fukushima accident into both operational and future plants. Hence, PWRs seem to be the best technology option, building on existing safeguards, safety and security international experience.

## A–6. INDONESIA

Indonesia has been an active member of the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) since 2004. The State has conducted ongoing activities of preparing a Nuclear Energy System Assessment (NESA), since it is planning to deploy SMRs onto small and remote islands in the future. The assessment covers the complete nuclear energy system. It also includes the preparation of user requirements based on basic principles and requirements of INPRO methodology, including PR&PP as part of the design requirements for reactor safeguardability. Indonesia will be one of the States embarking on a nuclear power programme that will benefit from utilizing the INPRO methodology.

### A–6.1. Prospects and security requirements of nuclear power plants with small and medium sized reactors

Development of highly proliferation resistant SMR concepts to meet the needs of States which plan to use nuclear energy needs to be continued. Nuclear power plants with SMRs have application prospects in Indonesia, where national energy resources are limited — especially oil and gas. Indonesia has a good capability to manage, operate and maintain research reactors, as it has three research reactors currently in operation. Indonesia is an archipelago State with 17 508 islands. The main islands (excluding Java), such as Sumatra (including Bangka–Belitung), Kalimantan, Sulawesi, the Maluku Islands and West Papua, have limited electric power capacity grid systems. Therefore, SMRs are suitable reactors for these islands based on the rule of thumb that additional capacity should not exceed 10–20% of the grid capacity. Indonesia might implement very small reactors (less than 150 MW(e)) in remote areas.

SMRs may enter and contribute to the overall energy system through the generation of electricity and production of process heat for coal liquefaction, seawater desalination, enhanced oil recovery and hydrogen production. The status of candidate SMR technologies and their economic aspects should be clear and transparent in the following areas:

- Proven technology (at least three years of successful operating experience);
- Investment costs;
- Security of fuel supply (and security of supply of other components);
- Technology transfer;
- Based on Indonesian regulatory requirements, all nuclear power plants, including SMRs, are to be located on land.

Indonesia adheres to the following requirements in the area of proliferation resistance:

(a) State commitments, obligations and policies regarding non-proliferation need to be adequate.
(b) The attractiveness of nuclear material in a nuclear energy system for a nuclear weapons programme needs to be low. This includes the attractiveness of undeclared nuclear material that could credibly be produced or processed.
(c) The diversion of nuclear material needs to be reasonably difficult and detectable.
(d) Nuclear energy systems need to incorporate multiple proliferation resistance features and measures. The combination of intrinsic features and extrinsic measures, compatible with other design considerations, is to be optimized (in the design and engineering phase) to provide cost efficient proliferation resistance.

### A–6.2. Study of SPINNOR and VSPINNOR by Bandung Institute of Technology, Indonesia

The Small Power Reactor, Indonesia, No On-site Refuelling (SPINNOR) and the Very Small Power Reactor, Indonesia, No On-site Refuelling (VSPINNOR) are concepts of the small lead–bismuth cooled nuclear power reactors with fast neutron spectra, which can be operated for more than 15 years without on-site refuelling. They are based on a concept of a long life core reactor developed in Indonesia since the early 1990s in collaboration with the Research Laboratory for Nuclear Reactors, at the Tokyo Institute of Technology.

SPINNOR and VSPINNOR provide some technical features to reduce the attractiveness of their nuclear materials for weapon programmes, prevent the diversion of nuclear materials and undeclared production of direct use materials, and facilitate nuclear material accounting and verification. These proliferation resistance features include:

(a) Plutonium composition in the reactor core is unattractive for weapon purposes.
(b) Isotopic contents of the fuel provide a radiation barrier that complicates fuel handling, therefore reducing the attractiveness of fuel theft.
(c) Fuelling, refuelling and decommissioning of SPINNOR and VSPINNOR are assumed to be performed at a factory.
(d) The reactor vessel is sealed and assumed never to be opened at the site. In addition, it is very difficult to open the vessel and to remove the core because it is covered by many components, such as a steam generator, a pump and a cooling pool filled with high temperature lead–bismuth eutectics. Due to decay heat, the fuel is at a high temperature.
(e) As the vessel is sealed, there is no possibility of using excess neutrons generated in the core to produce nuclear weapon materials.

#### *A–6.2.1. Technical features and technological approaches used to facilitate physical protection of SPINNOR and VSPINNOR*

SPINNOR and VSPINNOR are assumed to have their vessels sealed and never opened during operation at the site and during transportation from and to the factory. The reactor compartments are assumed to be housed within a reinforced containment to anticipate serious accidents caused by sabotage or explosions. Nearly all serious accidents can be effectively prevented, controlled and mitigated by inherent safety features, passive safety systems and through large margins to fuel melting and coolant boiling.

The user consideration document (UCD) for Indonesia’s nuclear power plant is a set of documents consisting of seven common user considerations typical for Indonesia, which define common characteristics to be used by potential users of future nuclear power plants (including SMRs) in Indonesia. All suitable nuclear energy systems are included, but special attention is given to SMRs and non-electrical applications of nuclear energy. The UCD covers general requirement characteristics of desired nuclear systems and associated services and support. It also includes specific requirements that may be stipulated by Indonesia.

With regard to the UCD proliferation resistance safeguards regime, the vendor State is not to impose any additional requirements on intrinsic features against nuclear proliferation. The proliferation resistance of the nuclear energy system is expected to be the same as that in the vendor State. No additional proliferation resistance provisions need to be incorporated into the design of the nuclear energy system. In addition, the vendor State is to design the nuclear power plant for safeguards friendliness to the current IAEA safeguards regime. Safeguards friendliness includes provisions to easily install monitoring, supervision and accounting systems.

For UCD physical protection and security, the nuclear power plant design is to incorporate technical features and provisions for physical protection to protect against theft, sabotage and acts of terrorism through the integration of plant arrangements and system configuration with plant security design, in accordance with international guidance and the practices and regulations of Indonesia. These technical features include any intrinsic measures such as fences, walls, doors and gates, alarms and closed circuit television.

The physical protection system is to be based on the national current evaluation of the threats (design basis threats). A physical protection system has the specific objective to prevent adversaries from successfully completing a malicious act and thereby achieving their purpose. A clear description of this adversary threat is an essential prerequisite for assured and effective physical protection. A design basis threat or other appropriate threat statement provides threat based criteria against which the nuclear energy system protection system can be designed and evaluated.

The concept of the security systems needs to protect the plant against acts of industrial sabotage, terrorism and riots which could lead to threats to public health and safety or to considerable financial consequences to the plant owner. The plant security system needs the required system objectives, design simplicity, economics and reliability.

The security system needs to provide high assurance of the following:

— Protection of the general public and plant personnel against the potential release of radioactive materials to the environment, resulting from successful acts of sabotage;
— Protection of plant structures and equipment from damage, theft and vandalism;
— Protection against the potential loss of revenue resulting from successful acts of sabotage shutting the plant down for extended periods;
— Protection from adverse publicity resulting from successful acts of sabotage or unauthorized and undetected intrusion into the plant protected area.

The concept of nuclear security and physical protection for fuel cycles needs to be clearly defined in the following:

— Nuclear material accounting and control at facilities — State system of accounting for and control of nuclear material (SSAC);
— Physical protection of nuclear material and radioactive sources;
— Security of radioactive sources;
— Physical protection from illicit trafficking of nuclear material and radioactive sources.

### A–6.3. Status of Indonesia's infrastructure for security and physical protection requirements

A national detection strategic committee involving all off-site responses to deal with theft and sabotage of radioactive material and sources, and prevention of sabotages of nuclear facilities is under preparation. In coordination with police and military stationed around the Serpong nuclear area, such a committee has been initiated for nuclear facilities at the operator level. The national detection strategic committee related to nuclear security will be established with members from the National Nuclear Energy Agency (BATAN), Nuclear Energy Regulatory Agency (BAPETEN), police, military and customs. The joint training exercises are periodically conducted by BATAN and supported by off-site response forces (chemical, biological, radiological and nuclear task forces, armed forces, district police, air force, special squad for counterterrorism, forensic laboratory and bomb squad). The activities are well documented.

At the operator level, the concept of nuclear security and physical protection has been understood and is clearly defined. Personnel receive training courses and participate in workshops on nuclear material accounting and control at facilities (SSAC), security of radioactive sources, and physical protection and illicit trafficking of nuclear material and radioactive sources. BATAN has established a nuclear security and physical protection organization for the BATAN nuclear research centres located at Serpong, Bandung, Yogyakarta and Pasar Jum'at. BATAN is responsible for arranging security and physical protection for all BATAN nuclear facilities, as stated in BATAN Chairman Decree No. 392 Year 2005, on BATAN organizational structure and task. The results of evaluation concluded that the nuclear security organization at the Serpong site has to be handled by one command. The IAEA has also reviewed all regulation and legislation related to nuclear security. In order to develop the regulation of security and physical protection, Indonesia has issued Government Regulation No. 43/2006 on the Licensing of Nuclear Reactors and several decrees from the BAPETEN Chairman regulating licensing, inspection and penalties, including the implementation of physical protection system during use, storage and transportation.

### A–6.4. Overview of the proliferation resistance and security assessment for nuclear energy system

In Indonesia, the nuclear option is under consideration for reasons of energy security, environmental sustainability, diversification and economic competitiveness. It is necessary to meet the legal requirements, national and international regulations, and technical requirements (meet the best safety, security and safeguards standards at time of deployment). The nuclear option will be utilized not only for electricity generation but also for sea water desalination, coal liquefaction, hydrogen production and steam for enhanced oil recovery. It must be proven and meet local grid size and economic requirements, according to which the electricity generation cost from the nuclear power plant needs to be lower than that from other energy sources in the same region. It is expected to be easy to construct, operate, maintain and improve national capability in technological know-how, gradually increasing in national participation on a step by step basis. Flexible financing arrangement (soft loan and vendor State participation in the investment) as a result of lower upfront capital cost (i.e. better affordability) is one of the driving rationales for Indonesia when considering SMR technology adoption.

Many innovative SMRs offer certain intrinsic proliferation features. All nuclear power plants with innovative SMRs provide for the implementation of standard or advanced security measures to ensure physical protection against internal or external human actions of a malevolent nature. International cooperation has become the hallmark of these non-proliferation and security efforts. International efforts are focused both on assisting States in strengthening their programmes and on building regional and global networks for combating transnational threats.

#### *A–6.4.1. National requirement for proliferation resistance*

Indonesia has adopted the basic principles and requirements that have been provided by INPRO as written in the Guidance for the Application and Development of Sustainable Nuclear Energy System (NES) in Indonesia [A–1]. These basic principles and requirements are used to assess nuclear energy systems, including the basic principles and requirements for PR&PP.

## A–7. MOROCCO

### A–7.1. PR&PP features of advanced small and medium sized reactors

To enhance PR&PP features of advanced SMRs, it is necessary:

(a) To develop and implement technologies and approaches that will strengthen proliferation resistance and nuclear security of the SMRs at the two levels:
   (i) Intrinsic level, via developing new proliferation resistant technologies and designs;
   (ii) Extrinsic level, via furthering and reinforcing measures at the institutional level to accomplish safeguards procedures in a more effective and efficient way.
(b) To establish, at the regional level, a nuclear security and proliferation resistance human resources network.
(c) To provide a detailed methodology for the assessment of proliferation resistance and nuclear security of each SMR.
(d) To provide assistance to States embarking on a nuclear power programme in the development of a legal framework for implementation of the non-proliferation and physical protection obligations and measures.

### A–7.2. Overview of small and medium sized reactors

The National Electricity Office (Office national de l'électricité, ONE) investigated the possibility of introducing nuclear energy to diversify the energy portfolio. It conducted investigations consisting of siting and feasibility studies of the first nuclear power plant project. As far as the siting studies were concerned, a general review of the Sidi Boulbra site studies was conducted in September 1995, with the attendance of the ONE General Director, senior officials from the concerned national organizations and experts from the IAEA. The review concluded that:

(a) Sidi Boulbra site had no characteristic that would prevent the construction of a nuclear power plant, and on the contrary, it presented several positive aspects regarding the areas of demography, environment and potential external aggressions;
(b) Sidi Boulbra studies were conducted in full accordance with recommended IAEA guidelines.

In addition, IAEA experts recommended that monitoring the changes in site characteristics should be performed. The feasibility study in Morocco was updated and generated some findings. The nuclear reactors market survey adopts 'commercial availability' and 'technology proveness' among selection criteria. Considering the size of the electricity grid in Morocco, the following reactors were selected to complete the remaining chapters of the feasibility study:

— PWR 1000 MW(e);
— Pressurized HWR 700 MW(e) (CANDU-6);
— VVER 1000 MW(e).

Morocco has been taking steps in terms of preparedness and has been operating a TRIGA Mark II research reactor. As an INPRO Member, it is planning to integrate nuclear options into its energy production system by 2020. Innovative SMRs can be a good solution for seawater desalination (cogeneration system). Morocco is working to finalize its legal framework, especially in the nuclear security field. ONE is working to integrate nuclear security and non-proliferation considerations into technical specifications for its future nuclear power plant.

With regard to the nuclear legal framework within national nuclear legislation, a substantial set of national nuclear laws already exists. These laws regulate the three areas of licensing nuclear installations, radioprotection and civil liability. They were sufficient for permitting the construction and operation of Morocco's first research reactor. This package of national nuclear laws has been updated, and a global national nuclear law is presently in the process of signature.

For enhancing nuclear security activities in Morocco, the Nuclear Security Support Centre was also established with the following objectives:

(i) To develop and implement a national nuclear security training programme, tailored to Morocco's needs in nuclear security and reflecting the current and future nuclear infrastructure of the State;
(ii) To establish a nuclear security human resource network at the national and regional level, and to coordinate nuclear security support activities at the national level. Finally, it fosters a nuclear security culture in Morocco.

When focusing on SMRs, the Moroccan experience mostly concerns the Tan-Tan desalination project. A pre-project study of a nuclear desalination demonstration plant was jointly conducted by China and Morocco, with the support of the IAEA. The plant production was 8000 $m^3$ per day of potable water using a 10 MW(th) nuclear heating reactor developed by China, coupled to a high temperature multieffect distillation process. China was responsible for the conceptual design of the reactor system, the desalination plant and the coupling scheme between them, the investment estimation of the nuclear desalination demonstration project and the economic analysis of the water production cost, preliminary site evaluation and environmental impact assessment. Morocco was responsible for providing user requirements, as well as the necessary data and information for economic analysis and preliminary assessment of the plant site and environmental impact. The project is technically viable and feasible. However, it is not economically viable compared to other means of producing potable water. The project is currently on hold.

# REFERENCE TO ANNEX

[A–1] NATIONAL NUCLEAR ENERGY AGENCY, Guidance for the Application and Development of Sustainable Nuclear Energy System (NES) in Indonesia, BATAN, Jakarta (2006).

# DEFINITIONS

*The definitions given below may not necessarily conform to those adopted elsewhere for international use.*

**acquisition.** A high level stage of a proliferation resistance pathway, considering the set of activities conducted to acquire nuclear material in any form. Acquisition starts with the decision to acquire nuclear material and ends with the availability of nuclear material.

**analysis.** (a) The consideration in detail to discover essential proliferation resistance features or meaning.
(b) The breakdown of proliferation resistance features into components or essential features.

**assessment.** (a) The classification of proliferation resistance with respect to its importance.
(b) The act of judging or assessing a proliferation resistance situation or event.

**barrier.** A characteristic of a nuclear energy system that impedes proliferation.

**capabilities.** The elements an actor can draw upon to conduct the necessary steps inherent in each pathway. For proliferation resistance actors, capabilities are characterized in terms of general technical skills and knowledge, general resources, uranium resources, general industrial capabilities and specific nuclear capabilities.

**design basis threat.** A bounding characterization of the possible challenges to the facility to aid design.

**detection probability.** A proliferation resistance measure that expresses the cumulative probability of detecting the action described by a pathway or segment. At the coarse analysis level, it is a performance objective rather than a measure to be estimated. The IAEA Safeguards Glossary 2001 Edition defines it as:

"The probability, if diversion of a given amount of nuclear material has occurred, that IAEA safeguards activities will lead to detection."

**detection resource efficiency.** A proliferation resistance measure capturing the staffing, equipment and funding required to apply international safeguards to the nuclear energy system. Detection resource efficiency can be only qualitatively estimated at a coarse analysis level but can be quantitatively estimated at a refined level on the basis of safeguards system design.

**evaluation methodology.** The overall process of examining a nuclear energy system or a system element to determine its PR&PP robustness.

**extrinsic (institutional) measures.** (a) Relating to the actions undertaken to impede proliferation, sabotage or theft by States or institutions. These actions may be institutional, legal or operational in nature.
(b) The noun 'measures' is often used in this context, for example 'extrinsic measures' to enhance proliferation resistance. Such use is not to be confused with the different PR&PP use of 'measures' found in this publication to mean bases or standards of comparison. Owing to this difference, PR&PP refers to intrinsic and extrinsic measures. Examples of extrinsic measures to combat proliferation are international laws, treaties, protocols, import and export agreements, and the application of international safeguards and verification activities (including any safeguards measurement equipment employed).

**fabrication.** A high level stage of a proliferation resistance pathway considering the activities conducted to manufacture and to assemble nuclear explosive devices. Fabrication starts with the availability of nuclear weapons material ready for use in a nuclear explosive device (e.g. plutonium in metallic form) resulting from the processing stage or from direct acquisition and ends with the availability of one or more nuclear explosive devices.

**facility.** (a) A reactor, critical facility, conversion plant, enrichment plant, fabrication plant, reprocessing plant, isotope separation plant or separate storage installation.
(b) Any location where nuclear material in amounts greater than one effective kilogram is customarily used.

**fissile material type.** A proliferation resistance measure categorizing material based on the degree to which its characteristics affect its utility for use in nuclear explosives. It is estimated on metal material immediately prior to the fabrication stage.

**intrinsic.** Relating to the inherent properties or physical design features of a nuclear energy system or component. An intrinsic feature is likely to be very difficult or impossible to alter, and is therefore very robust and desirable. The term may be applied both to proliferation resistance and to physical protection. Intrinsic proliferation resistance features impede proliferation, while intrinsic physical protection features deter sabotage or theft. The beneficial action of an intrinsic proliferation resistance feature may be indirect (i.e. by enabling the application of a more cost effective or robust extrinsic feature). An example of an intrinsic proliferation resistance feature would be such a high heat rate that a material is rendered unusable for a weapon.

**measures.** The few high level parameters that can be used to express proliferation resistance robustness. Use of this term is not to be confused with another frequent use (e.g. safeguards measures) to indicate the set of extrinsic actions or procedures for material and facility control and protection.

**objectives.** The desired end point for the actor (i.e. the goal to be achieved). For example, in the proliferation evaluation the objective can be expressed in terms of the number of nuclear explosive devices with specified characteristics. For proliferation resistance actors, objectives are limited to the acquisition of nuclear weapons and further characterized in terms of the number of nuclear weapons, their reliability, the ability to stockpile them, their deliverability and production rate.

**outcomes.** In the context of PR&PP evaluation, the results of the system response analysis.

**pathway analysis.** For a given set of threats, identification of potential sequences of events that lead to the undesirable outcome (proliferation, sabotage or theft) and the estimation of the system response. For proliferation resistance, according to the scope of the evaluation, pathway analysis may involve the complete set of proliferation stages (acquisition, processing and fabrication) or only a subset. Each proliferation stage may be composed of one or more segments.

**pathways.** In proliferation resistance, potential sequences of events and actions followed by adversaries to achieve objectives. A pathway is composed of segments.

**pathway segment.** A distinct part of a pathway.

**processing.** A high level stage of a proliferation resistance pathway considering the set of activities conducted to convert the nuclear material obtained in the acquisition stage into material ready for use in a nuclear weapon.

**proliferation cost.** A proliferation resistance measure capturing the economic and staffing resources required to overcome the multiple barriers to proliferation. The measure is estimated in US dollars and might be scaled (e.g. against the total resources available to a proliferant State for military expenditures).

**proliferation resistance.** The characteristic of a nuclear energy system 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.

**proliferation technical difficulty.** A proliferation resistance measure capturing the inherent difficulty, arising from the need for technical sophistication and materials handling capabilities, required to overcome the multiple barriers to proliferation.

**safeguardability.** The ease with which a system can be effectively and efficiently safeguarded. Safeguardability is a property of the whole nuclear system and is estimated for targets on the basis of characteristics related to the nuclear material, process implementation and facility design.

**safeguards.** Activities conducted by an independent agency to verify that commitments made by States under safeguards agreements are fulfilled.

**sponsor.** Organization that commissions the PR&PP evaluation (e.g. governmental and international agencies, private utilities and design organizations).

**strategy.** A description in the PR&PP evaluation, in general terms, of the ways in which the actor may achieve its objective.

**system elements.** Facilities to be included in the PR&PP evaluation. For proliferation resistance, system elements are the collection of facilities inside the nuclear energy system where diversion, acquisition, processing or fabrication could take place. For physical protection, system elements are facilities in the nuclear energy system that can be, or can contain, targets for physical protection threats.

**system response.** In the context of proliferation resistance, the resistance that a nuclear energy system provides against proliferation.

**target.** For proliferation resistance, nuclear material that can be diverted or equipment and processes that can be misused to process undeclared nuclear materials or can be replicated in an undeclared facility.

**threat.** A description of a potential menace consisting of information about the actor and its strategy. A proliferation resistance threat can be described by defining the objectives, capabilities and strategy of a proliferant State.

**threat space.** A full inventory of potential threats.

## ABBREVIATIONS

| | |
|---|---|
| AHWR300-LEU | Advanced Heavy Water Reactor with Low Enriched Uranium and Thorium Mixed Oxide Fuel |
| BAPETEN | Nuclear Energy Regulatory Agency (Indonesia) |
| BATAN | National Nuclear Energy Agency (Indonesia) |
| BWR | boiling water reactor |
| CANDU | Canada deuterium–uranium |
| CAREM | Central Argentina de Elementos Modulares |
| DUPIC | direct use of spent PWR fuel in CANDU reactors |
| EC6 | enhanced CANDU-6 |
| ESFR | Example Sodium Fast Reactor |
| FBNR | fixed bed nuclear reactor |
| FSA | facility safeguardability analysis |
| GCR | gas cooled reactor |
| GIF | Generation IV International Forum |
| GT-MHR | gas turbine modular helium reactor |
| HEU | high enriched uranium |
| HTR-PM | high temperature reactor pebble bed module |
| HWR | heavy water reactor |
| INPRO | International Project on Innovative Nuclear Reactors and Fuel Cycles |
| INS | innovative nuclear energy system |
| LEU | low enriched uranium |
| LMCR | liquid metal cooled reactor |
| LWR | light water reactor |
| MOX | mixed oxide |
| MUF | material unaccounted for |
| NESA | Nuclear Energy System Assessment |
| NPT | Treaty on the Non-Proliferation of Nuclear Weapons |
| NRC | Nuclear Regulatory Commission (United States of America) |
| ONE | Office national de l’électricité (Morocco) |
| PR&PP | proliferation resistance and physical protection |
| PRADA | proliferation resistance: acquisition/diversion pathway analysis |
| PWR | pressurized water reactor |
| SBD | safeguards by design |
| SGAR | safeguards analysis report |
| SMART | system integrated modular advanced reactor |
| SMR | small and medium sized reactor |
| SPINNOR | Small Power Reactor, Indonesia, No On-site Refuelling |
| SSAC | State system of accounting for and control of nuclear material |
| UCD | user consideration document |
| VSPINNOR | Very Small Power Reactor, Indonesia, No On-site Refuelling |

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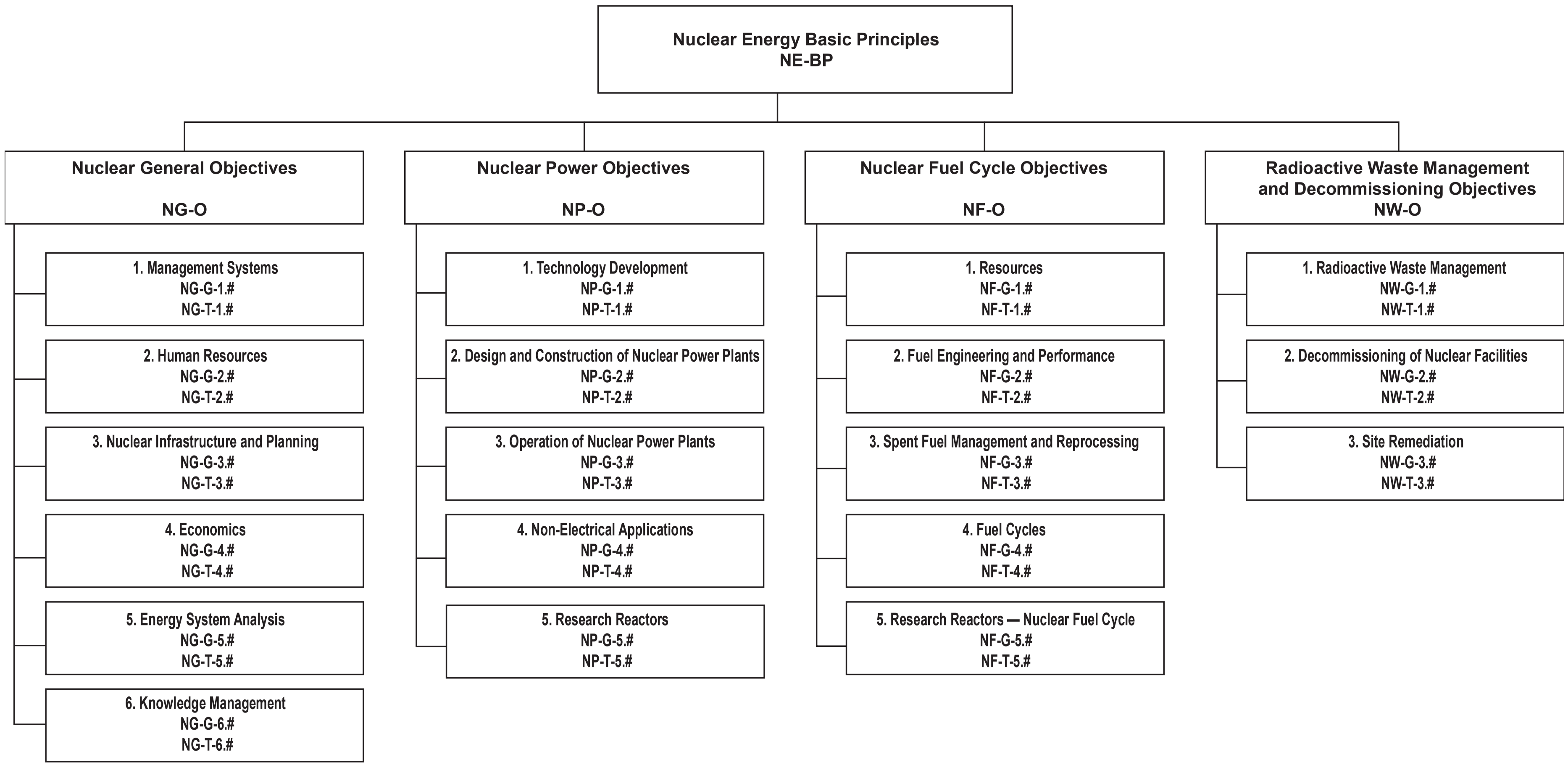


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