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Chapter 7

Generation IV Nuclear Systems: State of the Art and Current Trends with Emphasis on Safety and Security Features

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Additional information is available at the end of the chapter

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1. Introduction

Fifty years ago, on June 26, 1954, in the town of Obninsk, near Moscow in the former USSR, the first nuclear power plant was connected to an electricity grid to provide power. This was the world's first nuclear power plant to generate electricity for a power grid, and produced around 5 MWe [1]. This first nuclear reactor was built twelve years after the occurrence of the first controlled fission reaction on December 2, 1942, at the Manhattan Engineering District, in Chicago, Illinois, US. In 1955 the USS Nautilus, the first nuclear propelled submarine, equipped with a pressurized water reactor (PWR), was launched. The race for nuclear technology spanned several countries and soon commercial reactors, called first generation nuclear reactors, were built in the US (Shippingport, a 60 MWe PWR, operated 1957-1982, Dresden, a boiling water reactor, BWR, operated 1960-1978, and Fermi I, a fast breeder reactor, operated 1957-1972) and the United Kingdom (Magnox, a pressurized, carbon dioxide cooled, graphite-moderated reactor using natural uranium).

A few years after the projects had developed many nuclear safety concepts were extended and then implemented in second-generation nuclear systems, consisting of reactors currently in operation, as the PWR, CANDU (Canadian Deuterium Uranium Reactor), BWR (Boiling Water Reactor), GCR (Gas-Cooled Reactor), and VVER (Water - Water Power Reactor), the latter developed by the former Soviet Union. At this time, other concepts were studied in parallel, such as liquid metal cooled reactors and the reactors with thorium and uranium molten salt, which did not propagate commercially and/or remained in experimental countertops. Operating or decommissioned power reactor designs can be found in [2].

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With operating experience gained in recent decades, digital instrumentation development and lessons learned from the accidents at TMI (Three Mile Island), Chernobyl and recently Fukushima, Generation III and III+ reactor designs have incorporated improvements in thermal efficiency and included passive system safety and maintenance costs and capital reduction. There are several designs of these so called advanced reactors and some are being built in the U.S. and China.

In 2001, nine countries (Argentina, Republic of Korea, Brazil, Canada, Republic of South Africa, United Kingdom, France, United States and Japan) signed the founding document of Generation IV International Forum (GIF) in order to develop nuclear systems that can fulfill the increasing world electric power needs with high safety, economics, sustainability and proliferation resistance levels [3]. The Russian Federation, People´s Republic of China, Switzerland and Euratom have joined this group. Since then, GIF has selected the six most promising reactor system designs to be developed until 2030 and has created research groups on materials, fuel and fuel cycle, conceptual design, safety, thermal-hydraulics, computational methods validation and others in order to develop the necessary technology on an international cooperation basis.

According to IAEA [4], an accessible, affordable and sustainable energy source is fundamental to the development of modern society. Current scenarios predict a global demand for primary energy 1.5-3 times higher in 2050 as compared to today, and a 200% relative increase in the demand for electricity. Nuclear power is an important source that should be considered, because it is stable power on a large scale, with virtually no greenhouse gas emissions and low environmental impact as compared to fossil fuels, and can produce heat for chemical processes in industry and for hydrogen generation. In this context, GIF is seeking to develop more economical, sustainable and safe nuclear reactors, from their fuel cycles to decommissioning and waste treatment, and thus meet the world's energy needs.

This chapter is organized as follows. Section 2 discusses the goals for Generation IV reactor systems, section 3 discusses the current six Generation IV reactor system design description and research on the subject. Section 4 focuses on the first reports concerning reactor safety and risks [5], particularly the Integrated Safety Assessment Methodology (ISAM). Section 5 addresses economic aspects of this new reactor generation [6]. Section 6 concerns the discussion on proliferation resistance and physical protection [7]. Section 7 encompasses a set of general conclusions on the subject, addressing mainly the relevance of these nuclear system concepts.

2. Goals for generation IV nuclear energy systems

Before selecting the reactors that will be part of the Next Generation Nuclear Plant (NGNP), the founding countries of the Generation IV International Forum selected eight key objectives able to make these reactor designs vital in the near future. These eight goals based on concepts of sustainability, economic competitiveness, safety, physical protection and nuclear proliferation resistance are described below [3]:

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1. Sustainability-1 – NGNP will provide sustainable energy generation that meets clean air objectives and provides long-term system availability and effective fuel utilization for worldwide energy production;
2. Sustainability-2 – NGNP will minimize and manage their nuclear waste and notably reduce the long-term administrative burden, thereby improving protection for the public health and the environment.
3. Economics-1 – NGNP will have a clear life-cycle cost advantage over other energy sources.
4. Economics-2 – NGNP will have a level of financial risk comparable to other energy projects.
5. Safety and Reliability-1 – NGNP operations will excel in safety and reliability.
6. Safety and Reliability-2 – NGNP will have a very low frequency and degree of reactor core damage.
7. Safety and Reliability-3 – NGNP will eliminate the need for offsite emergency response.
8. Proliferation resistance and Physical Protection – NGNP will increase the assurance that they are very unattractive and the least desirable route for diversion or theft of weapons-usable materials, and provide increased physical protection against acts of terrorism.

With these concepts in mind, projects already in operation, in test or just conceptual have been analyzed and six promising nuclear reactors have been selected to be designed and built in this endeavor.

3. The six reactor concepts under discussion

Based on GIF goals, six reactor designs have been selected to be developed and constructed by 2030. Such reactors must meet the safety and security, sustainability, economics and proliferation resistance criteria defined as essential for this new generation. That is, projects must: consider the entire fuel cycle to obtain a higher fuel burn-up and consequently less actinides in the final waste, reducing their lifetime in final repositories; increase the thermal efficiency with combined cycles and cogeneration; aim at the production of hydrogen, be intrinsically safe, with negative reactivity coefficients and achieve competitive costs since their construction.

GIF has divided the involved countries in working groups formed by laboratories, universities, and government agencies, according to the experience and interest of each to develop the projects of the following reactors: Very-high-temperature reactor (VHTR), Gas-cooled fast reactor (GFR), Supercritical-water-cooled reactor (SCWR), Sodium-cooled fast reactor (SFR), Lead-cooled fast reactor (LFR) and Molten salt reactor (MSR). For each project specific groups of materials, computational methods, fuel and fuel cycle, thermal-hydraulics, safety and operation and others have been created. The first four reactor concepts are completely defined and the remaining two are in progress.

A major challenge for almost every project is the development of new materials for the reactor primary systems that can tolerate temperatures up to 1000 °C without reducing safety margins. Another point is the need to develop and validate computer codes both in neutronics and thermal-hydraulics projects with little or no available operational experience. These issues and the obtained solutions will be discussed next.

3.1. Very-high-temperature reactor (VHTR)

The VHTR is one of the most promising projects of Generation IV reactors, given the experience in gas-cooled reactors that many countries developed in recent decades. The first reactors of this type were designed by the UK and France for the production of plutonium [8], were graphite-moderated and used CO_2 as a coolant, which limited the maximum temperature at 640 °C, when chemical reactions occur between gas and moderator. Later, the use of helium (although more costly) was justified by having better heat transfer properties, and permit increasing the core outlet temperature.

The current GIF design consists of a helium-cooled reactor with graphite as a moderator, TRISO fuel and coolant outlet temperature above 900 °C. There are two core concepts: the prismatic block-type and the pebble bed-type. The first type follows the line of the High Temperature Engineering Test Reactor (HTTR) developed and built by Japan initially with coolant exit temperature of 850 °C and then 950 °C in April 2004 [9]. The second is the result of the German program, which was later imported by the People's Republic of China and developed in the Republic of South Africa as the Pebble Bed Modular Reactor (PBMR). Both designs use TRISO fuel (see Figure 1), which consists of a spherical micro kernel of oxide or oxycarbide fuel and coating layers of porous pyrolytic carbon (buffer), inner dense pyrolytic carbon (IPyC), silicon carbide (SiC) and dense outer pyrolytic carbon (OPyC).

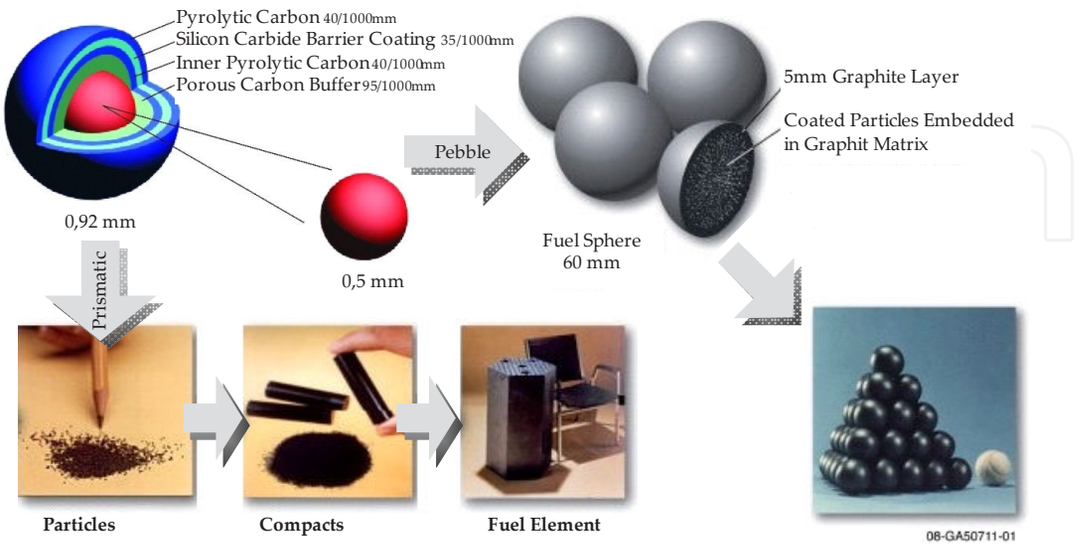


Figure 1. TRISO fuel for prismatic block-type and pebble bed-type cores [10,11].

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The goal of achieving an exit temperature of 1000ºC coolant will allow the VHTR to generate electricity with high efficiency and provide heat for hydrogen production and for refineries, petrochemical and metallurgical industries. For this purpose, research is being done to evaluate the use of other materials, such as uranium-oxicarbide UCO and ZrC, which increase the capacity of TRISO fuel burn-up, reduce the permeability of fission products and increase resistance to high temperatures in case of an accident (above 1600 °C) [12]. Although a once-through uranium fuel cycle is planned reactors have potential for this deep-burn of plutonium and minor actinides, as well as the use of thorium based fuel [13].

The experimental reactor HTTR in Japan and HTR-10 in China support the development of the VHTR, in particular providing important information on safety and operational characteristics. The research group on computational methods validation and benchmarking uses these reactors to validate computer codes in the areas of thermal hydraulics, thermal mechanics, core physics and chemical transport.

3.2. Gas-cooled fast reactor (GFR)

Another project selected by GIF of a gas-cooled reactor is the GFR, the fast-neutron-spectrum, helium-cooled and closed fuel cycle reactor. The selection of this reactor was based on its excellent potential for long-term sustainability, in terms of the use of uranium and of minimizing waste through reprocessing and fission of long-lived actinides [14,15]. In terms of non-proliferation, the objective of high burn-up together with actinide recycling results in spent fuel that is unattractive for handling. The use of gas as coolant means achieving high temperatures, so that this design also has the purpose of providing process heat and enabling hydrogen production.



Figure 2. Updated layout of ALLEGRO featuring two main heat exchange loops [15].

One of the proposed designs is a 1200 MWe reactor, operating at 7 MPa, coolant exit temperature of 850ºC, an indirect combined cycle with He-N_2 gas mixture for intermediate gas cycle and natural convection as passive safety system [14]. At least two fuel concepts that meet the proposed design are being studied: a ceramic plate-type fuel assembly and a ceramic pin-type fuel assembly. In this latter, the fuel assembly is based on a hexagonal lattice of

fuel-pins and the materials used are uranium and plutonium carbide as fuel, and silicon carbide as cladding [16].

An experimental reactor called ALLEGRO (Fig. 2) with 80 MWth will be the first built GFR with the objective of demonstrating the feasibility and qualifying technologies for fuel, fuel assembly and new safety systems. It is important to notice the interface between GFR and VHTR reactors, which use helium as a coolant and power conversion technology using gas turbines and cogeneration, as many efforts on component and materials development are being held together.

3.3. Supercritical-water-cooled reactor (SCWR)

The SCWR concept emerged at the University of Tokyo in 1989 and became a global concern after being selected by the Generation IV International Forum in 2002 [17]. The SCWR is a high-temperature, high-pressure, water-cooled reactor that operates above the thermodynamic critical point of water (374 °C, 22.1 MPa). This last feature eliminates coolant boiling, since both liquid and gaseous states can coexist. Thus, according to [18], the need for recirculation and jet pumps, pressurizer, steam generators, and steam separators and dryers in current LWRs is eliminated. Still, according to [18] the main mission of this reactor design is the generation of low-cost electricity.

A more advanced project is the Japanese 1620 MWe SCWR consisting of a pressure-vessel type, once-through reactor and a direct Rankine cycle system. Figure 3 is a simplified system where steam leaves the pressure vessel at 560 °C and 25 MPa, passes through moisture separator valves and control valves, high and low pressure turbines connected to the generator, condenser, low and high pressure pumps, a deaerator (to remove dissolved gases) to return to the pressure vessel. Because of supercritical water condition, no phase change occurs, which means that the coolant continuously passes from the liquid state to the gaseous state.

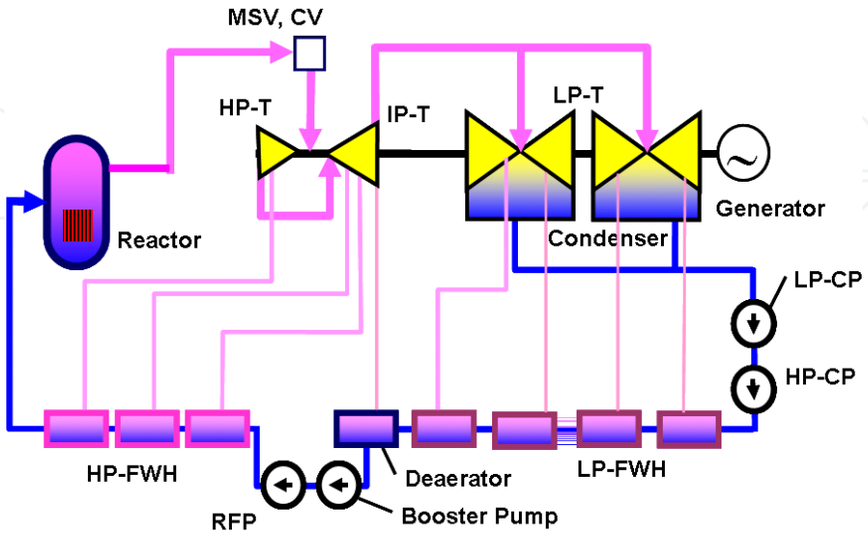


Figure 3. The Japanese Supercritical Water Reactor (JSCWR) [17].

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Another concept proposed by Canada is generically called CANDU-SCWR [19], it is a pressured tube type reactor with fuel channels separating the light water coolant from the heavy water moderator. The supercritical steam is led directly to the high-pressure turbine, eliminating the need of steam generators as the JSCWR.

3.4. Sodium-cooled fast reactor (SFR)

The Sodium-cooled Fast Reactor is among the six candidate designs selected for their potential to meet Generation IV technology goals. It is a fast-spectrum, sodium-cooled reactor and closed fuel cycle for efficient management of actinides and conversion of fertile uranium Three options are under consideration (Figure 4): a large size loop type sodium-cooled reactor with uranium-plutonium oxide fuel (1500 MWe), a medium size pool-type system with 600 MWe and a small size modular type with plutonium-minor actinide-zirconium metal alloy fuel [20].

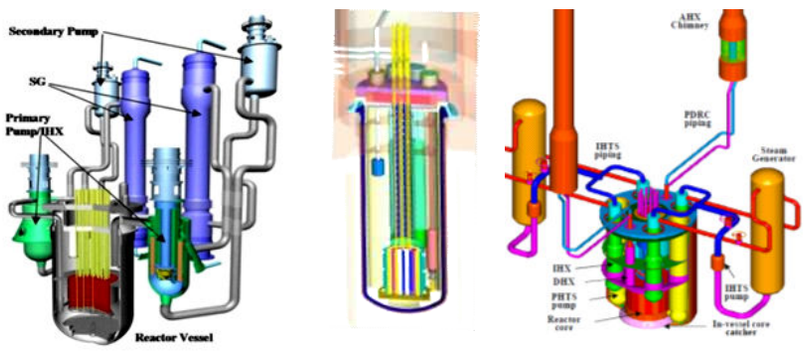


Figure 4. Loop, modular and pool configuration [20,21]

Heat capacity and high thermal conductivity of liquid-metal coolants provides large thermal inertia against system heating during loss-of-flow accidents. Furthermore, due the non-corrosive characteristic of sodium coolant, the reactor core and primary system components do not degrade even over very long residence times in the reactor, so that maintenance requirements are minimized. However sodium reacts chemically with air and water and requires a great coolant system seal.

3.5. Lead-cooled fast reactor (LFR)

The application of lead coolant in nuclear systems began in the 1970s in the ancient Soviet Union with systems cooled by Lead-Bismuth Eutectic (LBE), which were developed for nuclear-powered submarines [22]. Two of the currently proposed designs for international cooperation are the Small Secure Transportable Autonomous Reactor (SSTAR, Fig. 5) and the European Lead-cooled System (ELSY). Both concepts are lead-cooled, with passive decay heat removal and nitride fuels. A closed fuel cycle is expected for the efficient conversion of fertile uranium and actinide management for this reactor.

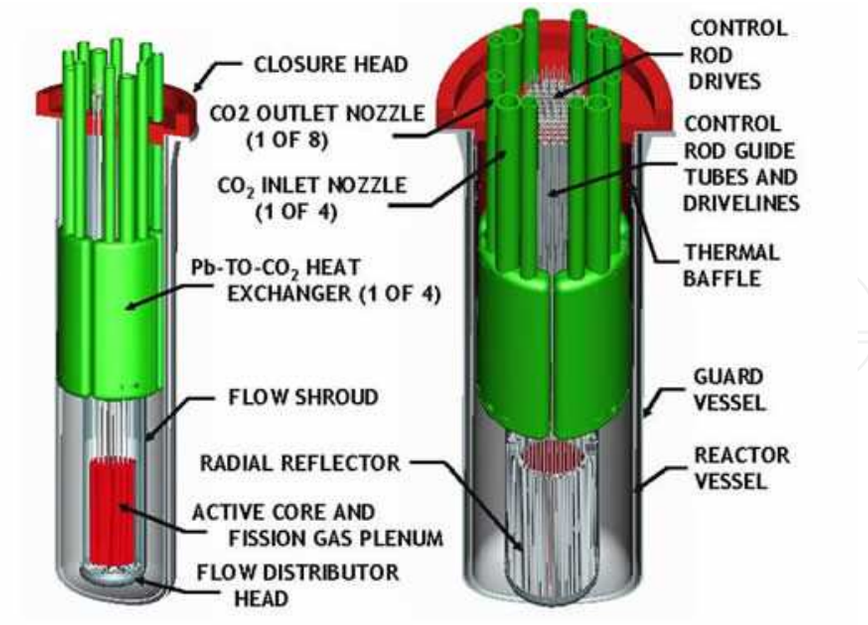


Figure 5. Small Secure Transportable Autonomous Reactor [22].

Lead flows in the reactor core cooling it by natural circulation and passes through Pb-to-CO_2 heat exchangers located between the vessel and the cylindrical shroud. It is noteworthy that lead has been chosen as coolant rather than LBE to drastically reduce the amount of alpha-emitting ^{210}Po isotope produced [22]. Many studies on the choice of fuel and fuel cycle have been proposed since 2004 when an international cooperation through GIF began. Features of this reactor, like enabling fissile self-sufficiency, autonomous load following, operation simplicity, reliability, transportability, as well as a high degree of passive safety, make it a unique proliferation resistant, safe and economical design.

3.6. Molten salt reactor (MSR)

Investigation of molten-salt reactors started in the late 1940s as part of the United States' program to develop a nuclear powered airplane [23]. Today, research on this reactor has been resumed due to its inclusion as one of the six Generation IV reactor types. MSRs have been initially considered as thermal-neutronic-spectrum graphited-moderated concepts, but since 2005 R&D has focused on the development of fast-spectrum MSR concepts. The Molten Salt Fast Reactor core is a cylinder of the same diameter and height, where nuclear reactions occur in the fuel salt. Two options have been considered for the fuel cycle: 233U-started MSFR and TRU(fuelled with transuranic elements)-started MSFR [24]. The salt processing scheme relies on both on-line and batch processes (Fig. 6) to satisfy the constraints for a smooth reactor operation while minimizing losses to waste stream [25]. A simplified schematic of the online processing can be seen in Figure 6, where fuel fission gas extraction and actinide separation occur.

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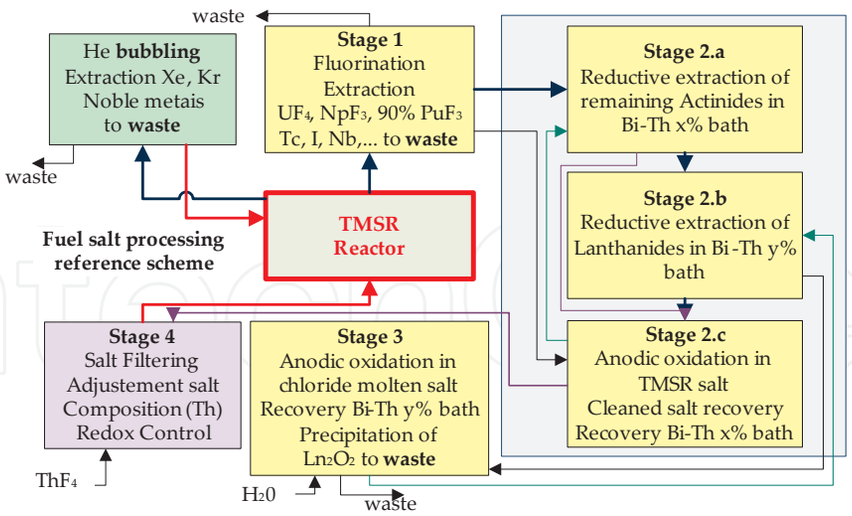


Figure 6. TMSR (MSFR) reference fuel salt processing [25].

A survey of the main features of the reactors discussed here is displayed in Table 1. It is important to remember that these projects are in development and there are different views in the same reactor line. Some reactors are already testing their designs with full construction and will be essential for future validation methodologies. If these six reactors achieve their ultimate goals, they will not only bring more advanced technology, but new concepts of safer, sustainable and economical nuclear reactors. Some of these aspects will be discussed in more detail in the following sections.

	GFR	SCWR	MSR	SFR	VHTR	LFR
Neutrons energy	Fast	Thermal	Thermal or fast	Fast	Thermal	Fast
Power (MWe)	1200	Up to 1500	1000	50 - 1500	~200	19.8 - 600
Fuel type	Plate or ceramic	UO_2 pellet	Molten salt	MOX or Metal alloy	TRISO	MOX or Nitrides
Coolant	Helium	Light water	Molten salt	Sodium	Helium	Lead
Moderador	-	Light water	-	-	Graphite	-
Outlet temperature	850°C	625°C	1000°C	~500°C	1000°C	~500°C
Pressure * (MPa)	7	25	~0.1	~0.1	5 - 7	~0.1
Fuel Cycle	Closed	once through	Closed	Closed	once through	Closed

* Primary system pressure.

Table 1. Survey of Generation IV Nuclear Reactors.

4. The integral safety assessment methodology (ISAM)

4.1. GIF safety philosophy

Since the very beginning, Safety and Reliability issues were extensively considered, representing three of the eight goals of Generation IV Nuclear Energy Systems [26]. Safety, together with waste management and nuclear proliferation risks, remains as one of the key problems of nuclear energy. Consequently, a new generation of Nuclear Power Plants (NPPs) has to face and solve these problems convincingly.

In spite of past nuclear accidents, including the most recent at Fukushima Daiichi, NPPs have an excellent safety record and still remain as a safe and high-power energy production technology without greenhouse gases emissions. Unfortunately, public acceptance continues to be an important impediment.

In this context, the Generation IV International Forum (GIF) safety objectives have been oriented to substantially upgrade safety and enhance public confidence in NPPs, by means of an increasing use of inherent safety features, a major reduction of core damage frequency and by eliminating the need for offsite emergency response in case of accidents [26]. It was immediately recognized the necessity of a standard methodology for Generation IV safety assessments. The methodology would allow a uniform safety evaluation of different NPP concepts with respect to Generation IV safety goals [26].

During the past 60 years safety assessment has evolved, from the initial deterministic analysis through conservative assumptions and calculations, to an increasing application of a best-estimated deterministic approach, in conjunction with probabilistic methods [27, 28]. Best-estimated and probabilistic assessments identify potential accidents scenarios that could be important contributors to risk [28]. As a consequence, Probabilistic Safety Analysis (PSA) is having a more prominent and fundamental role in the design process and licensing analysis, as part of an integrated approach, risk-informed, which also includes Deterministic Analysis and Defense in Depth Philosophy [29]. This new safety assessment conception, integrating probabilistic and deterministic analysis with an extensive application of the Defense in Depth principle, constitutes the basis for the Generation IV safety assessment methodology.

4.2. Phenomena identification and ranking tables (PIRT)

In 2004, the Idaho National Engineering and Environmental Laboratory proposed a Research and Development Program Plan, identifying the R&D needs for the next generation nuclear plant (NGNP) design methods [30]. Later, in 2008, the Idaho National Laboratory prepared a report for the US Department of Energy with a R&D Technical Program Plan for the NGNP methods [31]. Both reports were focused on the development of tools to assess the neutronic and thermal-hydraulic behavior of the Very High Temperature Reactor (VHTR) systems, using a Phenomena Identification and Ranking Tables (PIRT) informed R&D process.

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The first two steps of the proposed methodology are 1) Scenario Identification, where operational and accident scenarios that require analysis are indentified, and 2) PIRT, where important phenomena are identified and ranked for each scenario. In the following stages of the methodology, analysis tools are evaluated to determine whether important phenomena can be calculated and operational and accident scenarios that require study are analyzed.

PIRT was developed in the late 1980s, for the qualification of deterministic safety codes for Light Water Reactors [32, 33]. It is a systematic way of gathering information from experts on a specific subject, and ranking the importance of the information, in order to meet some decision-making goal [34]. An important part of the process is to also identify the associated uncertainties, usually by scoring the knowledge bases for the phenomena. Example of successful PIRT applications in thermal-hydraulics, severe accidents, fuels, materials degradation, and nuclear analysis may be found in [34]. An extensive application of PIRT is reported in [33, 35, 36], for the determination of code applicability for the analysis of selected scenarios with uncertainty evaluations.

4.3. The risk-informed and performance-based approach

In 2007, the United States Nuclear Regulatory Commission (NRC) developed a risk-informed and performance-based regulatory structure for the licensing of future NPPs, with broader use of design specific risk information and applicable to any reactor technology [29]. Defense in Depth remains basic to this framework, providing safety margins to compensate for the uncertainties in the requirements for design, construction and operation.

In this model, PSA and the Licensing Basis Events (LBEs) deterministic calculations are closely linked. LBEs are selected developing a Frequency-Consequence (F-C) curve, which is used together with the plant specific PSA.

The performance-based approach is applied whenever possible, so that performance history is used to focus attention on the most important safety issues. Objective criteria are established for evaluating performance.

4.4. Defense in depth: The objective provisions tree (OPT)

IAEA has also proposed a new safety approach and a methodology to generate technology-neutral safety requirements for advanced and innovative reactors [37]. The document identified several areas requiring further development, such as the replacement of qualitative safety objectives for quantitative ones, the enhancement of Defense in Depth, further development of PSA, the application of methodologies early in the design process, the use of an iterative design process to demonstrate the adequacy of Defense in Depth and a comprehensive review of the existing safety approach.

The Objective Provisions Tree (OPT) methodology [38, 39] was suggested as a tool to systematically examine all possible options for provisions to prevent and/or control challenging mechanisms jeopardizing Defense in Depth. OPT is a systematic critical review of the Defense in Depth implementation. It identifies the required provisions that jointly ensure the

prevention or control of a mechanism that represents a challenge for a safety function, which is part of a Defense in Depth Level. The existence of several challenges for each safety function and several mechanisms contributing to a given challenge leads to a tree structure. The set of provisions, jointly ensuring the prevention of one mechanism, constitutes a Line of Protection (LOP).

The proposed main pillars of the new IAEA safety approach are Quantitative Safety Goals (correlated with each level of Defense in Depth), Fundamental Safety Functions, and Defense in Depth (generalized, including probabilistic considerations). Quantitative Safety Goals are based on the condition that plant states with significant frequency of occurrence have only minor or no potential radiological consequences, according to the F-C curve, called Farmer´s Curve.

Defense in Depth essential characteristics were described as exhaustive, balanced and graduated. "Balanced" means that no family of initiating events should dominate the global frequency of plant damage states. "Graduated" means that a progressive defense excludes the possibility of a particular provision failure to generate a major increase in potential consequences, without any possibility of recovering the situation at an intermediate stage.

4.5. The risk and safety working group (RSWG)

A Risk and Safety Working Group (RSWG) was created in the frames of the Generation IV Forum to develop a safety approach for Generation IV Nuclear Systems. After its initial meeting in 2005, the first major work product, establishing the basis for the required safety approach, was released by the end of 2008 [40]. The document recommended the early application of a cohesive safety philosophy based on a concept of safety that is "built-in, not added-on". The identification of risks must be as exhaustive as possible. The remaining lack of exhaustiveness of the accident scenarios should be covered by the notion of enveloped situations and the implementation of the Defense in Depth principles. A re-examination of the safety approach was proposed, complementing the IAEA suggestions [37] with the following attributes:

- Risk-informed, combining both probabilistic and deterministic information.
- Understandable, traceable, and reproducible.
- Defensible. Whenever possible, known technology should be used.
- Flexible. New information and research results should be easily incorporated.
- Performance-based.

RSWG recommended a safety approach that manages simultaneously deterministic practices and probabilistic objectives, handling internal and external hazards. Besides the Defense in Depth characteristics previously mentioned (exhaustive, balanced and graduated) [37], RSWG added the followings: tolerant and forgiving. "Tolerant" means that no small deviation of a physical parameter outside the expected range can lead to severe consequences (ab-

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sence of cliff-edge effects). "Forgiving" means the availability of a sufficient grace period and the possibility of repair during accident conditions.

RSWG recognized safety related technology gaps for the six reactor concepts selected by the Generation IV Forum in different technical areas such as updated safety approach, fuel, neutronics, thermal aerolics/hydraulics, materials & chemistry, fuel chemistry, passive safety and severe accident behavior. An effective mix of modeling, simulation, prototyping and demonstrations should be used to reduce the existing uncertainties and lack of knowledge.

4.6. The integral safety assessment methodology (ISAM)

During 2008 the RSWG began the development of the methodology for Generation IV safety assessments, stated in [26]. The methodology would integrate PSA and several other techniques, such as PIRT and OPT, with an extensive deterministic and phenomenological modeling [41]. In 2009 the RSWG focused its work on the development of a methodology that was denominated Integrated Safety Assessment Methodology (ISAM) [42].

ISAM consists of five distinct analytical tools which are structured around PSA:

- Qualitative Safety Requirements/Characteristic Review (QSR).
- Phenomena Identification and Ranking Table (PIRT),
- Objective Provision Tree (OPT).
- Deterministic and Phenomenological Analyses (DPA).
- Probabilistic Safety Analysis (PSA).

From its original conception, it was clearly established that ISAM is not intended to dictate requirements or compliance with safety goals to designers. Its intention is solely to provide useful insights into the nature of safety and risk of Generation IV systems for the attainment of Generation IV safety objectives. ISAM will allow evaluation of a particular Generation IV concept or design relative to various potentially applicable safety metrics or "figures of merit" (FOM) [42, 43].

During 2010 the RSWG focused its work on the finalization of ISAM methodology presented in 2009. ISAM was conceived as a methodology providing specific tools to examine relevant safety issues at different points in the design evolution, in an iterative fashion through the development cycle. It is considered well integrated, and when used as a whole, offers the flexibility to allow a graded approach to the analysis of technical issues of varying complexity and importance [5].

Finally, the document describing the Generation IV Integrated Safety Assessment Methodology (ISAM) was available in June, 2011 [5]. According to this report, a principal focus of RSWG is the development and demonstration of an integrated methodology that can be used to evaluate and document the safety of Generation IV nuclear systems.

An important remark is that the safety approach for Generation IV nuclear systems should differ from the one, usually applied to previous reactor generations, in which safety is gen-

erally "added on" by applying safety assessments to relatively mature designs and introducing the results in many cases as "backfits". ISAM is therefore intended to support achievement of safety that is "built-in" rather than "added on" by influencing the direction of the concept and design development from its earliest stages [5].

It is envisioned that ISAM will be a "tool kit" used in three different ways:

- As an integrated methodology, throughout the concept development and design phases, revealing insights capable of influencing the design evolution. ISAM can develop a more detailed understanding of safety-related design vulnerabilities, and resulting risk contributors. New safety provisions or design improvements can be identified, developed, and implemented relatively early.
- Applying selected elements of the methodology separately at various points throughout the design evolution to yield an objective understanding of safety-related issues (such as risk contributors, safety margins, effectiveness of safety provisions, uncertainties, etc.) important to decision makers.
- In the late stages of design maturity, for decision makers and regulators to measure the level of safety and risk associated with a given design relative to safety objectives or licensing criteria ("post facto" application).

ISAM is essentially a PSA-based safety assessment methodology, with the additional strength of other tools, tailored to answering specific types of questions at various stages of design development. The methodology is well integrated. Although individual analytical tools can be selected for separate and exclusive use, the full value of the integrated methodology is derived from using each tool, in an iterative fashion and in combination with the others, throughout the development cycle. Figure 7 shows the overall task flow of ISAM and indicates which tools are intended for each phase of Generation IV system technology development [5].

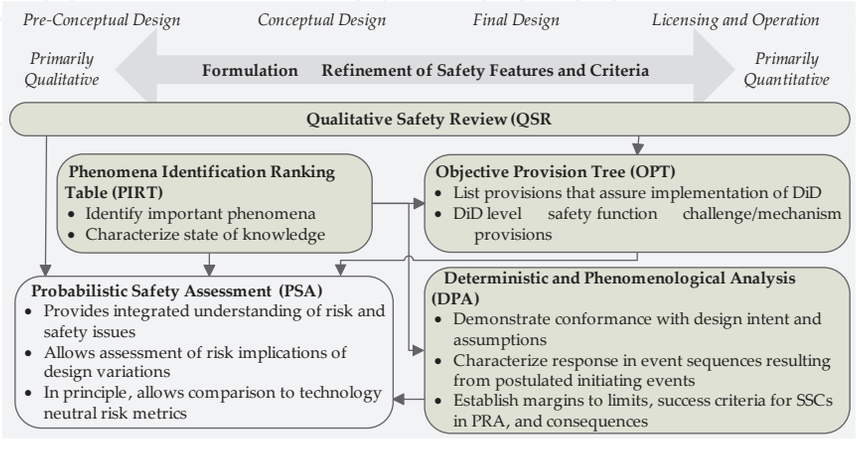


Figure 7. ISAM Task Flow [5].

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The methodology comprises several stages, from the pre-conceptual design to licensing and operation, with the corresponding safety features and criteria moving from primarily qualitative to quantitative. In the early design stages, the main role corresponds to the qualitative safety analysis techniques QSR, PIRT and OPT, but Deterministic and Phenomenological Analysis (DPA) and Probabilistic Safety Assessment (PSA) are introduced earlier in comparison with the current practice for previous reactor generations. DPA and PSA are the key techniques to be applied during the last stages corresponding to final design, licensing and operation, where safety criteria are mainly quantitative.

4.7. The qualitative safety features review (QSR)

Only one of the tools integrated in ISAM is completely new, and was developed specially for Generation IV Reactor Systems: the Qualitative Safety features Review (QSR). QSR is intended to provide a systematic means of ensuring and documenting that the evolving Generation IV system concept of design incorporates the desirable safety-related attributes and characteristics that are identified and discussed in [40].

QSR is conducted using a template structure organized according to the first four levels of Defense in Depth (prevention, control, protection and management of severe accidents). The review is based on an exhaustive check list of safety good practices and recommendations applicable to Generation IV systems. Design options are evaluated, identifying their strength or weakness, and qualified as favorable, unfavorable or neutral in relation with the desirable characteristics.

An exhaustive and detailed check list is essential for the QSR quality. To generate the list, a top-down functional approach is conducted, as shown in Figure 8 [5]. Firstly, the recommendations from RSWG, IAEA standards, INSAG and INPRO guidelines and other references, organized according to the levels of Defense in Depth, are detailed as much as possible, for a technology neutral condition. Finally, the defined neutral characteristics are used to develop the set of specific recommendations for a particular technology. In this process, the characteristics and features are grouped in four classes, moving from general recommendations to detailed specific attributes [5]:

Class 1: Generic and Technology neutral.

Class 2: Detailed and Technology neutral.

Class 3: Detailed and Technology neutral but applicable to a given safety function.

Class 4: Detailed and applicable to a given safety function and specific technology.

RSWG has developed check lists covering desirable Classes 1 to 3 safety characteristics for the first four levels of Defense in Depth. Class 3 recommendations were determined for the safety function “Decay Heat Removal” [5]. It is considered that QSR applications will be important for the identification of Generation IV R&D needs.

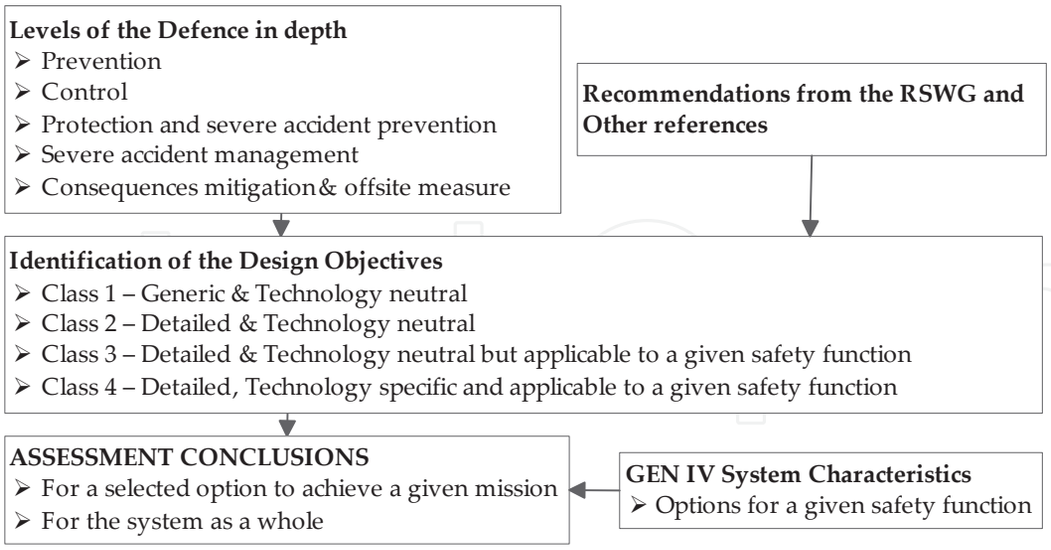


Figure 8. Top-down approach to establish safety recommendations derived from a QSR [5]

4.8. PIRT and OPT

In ISAM, PIRT and OPT are tools that complement each other. PIRT is used to identify phenomena impacting accident scenarios while OPT finds out the necessary Defense in Depth provisions to prevent, control or mitigate the corresponding phenomena. They can be iteratively applied from conceptual to mature designs. Both techniques are basically qualitative tools based on expert elicitation.

PIRT is more expert-dependant due to the complexity and diversity of possible phenomena involved in accident scenarios. The expert panel selects a FOM, for example, the fission products release, the maximum core coolant temperature, etc.; and the PIRT process is applied to each of the identified accident scenarios to determine and categorize the safety-related phenomena and the uncertainties in their knowledge. Usually a four level scale is used to rank the phenomena importance (high, medium, low or insignificant) and existing knowledge (full, satisfactory, partial or very limited). The ranked values of importance and uncertainty are useful to determine R&D effort priorities for accident scenarios. Safety issues located in the region of high importance and large uncertainty have the greatest priorities.

OPT is a systematic and structured top-down method for a fully characterization of the Defense in Depth architecture by identifying in detail the set of objective realistic countermeasures against a great diversity of safety-deteriorating mechanisms. Following a deductive approach, the model goes from each level of defense throughout the safety objectives and physical barriers; down to the safety functions and their challenges, until challenging mechanisms at the lowest tree levels are deduced and the associated sets of objective provisions in opposition to them, constituting LOPs, are determined. Figure 9 shows the tree structure of the OPT process [39].

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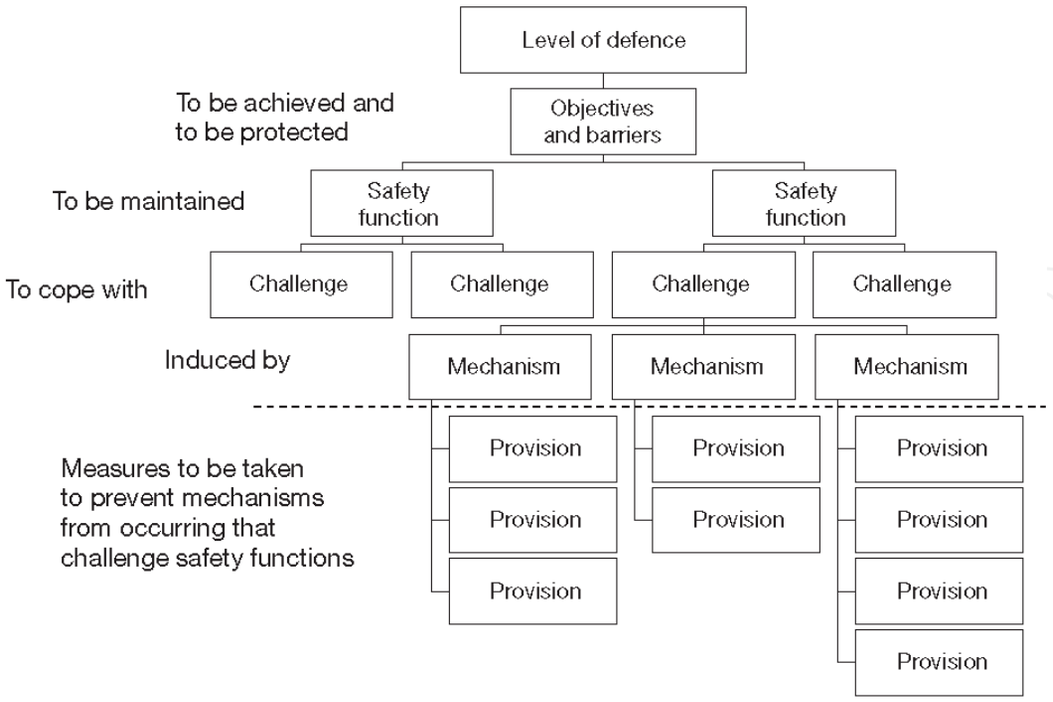


Figure 9. OPT process to determine Defense in Depth objective safety provisions [39]

The application of OPT is expected to contribute from early stages to the built-in safety approach proclaimed as part of Generation IV safety assessment philosophy. Its final objective is to ensure that Defense in Depth satisfies the desirable attributes of being exhaustive, balanced, progressive, tolerant and forgiving [5]. During the pre-conceptual and conceptual design phases OPT will serve as an important guidance to research efforts in order to achieve the mentioned Defense in Depth attributes by means of robust, reliable and simple design solutions. OPT can also identify degradation mechanisms representing feedbacks to PIRT.

4.9. DPA and PSA

PSA is the principal basis of ISAM, but DPA also constitutes a vital component, providing support to PSA, as well as to PIRT and OPT. Following the Risk-informed methodology, ISAM integrates PSA, DPA and Defense in Depth (evaluated using the OPT). The two mutually complemented methodologies, PSA and DPA, contribute to assess important phenomena identified in the PIRT process, and can also evaluate the effectiveness of LOPs deduced by means of OPT. DPA is indispensable to understand NPP safety issues, providing quantitative insights, important as PSA inputs. Best-estimated deterministic computer codes are preferred, incorporating sensitivity analyses to cover the existing uncertainties, depending on the design stage. One important challenge is the upgrade, development and validation of deterministic computer codes and their necessary input data to perform convincing safety assessments of some innovative reactor concepts.

RSWG supports the idea of applying PSA from the earliest practical stages of the design process, and continuing its application iteratively throughout the evolution of the design concept

until its maturity, in the stages of final design, licensing and operation. The PSA scope should comprise both internal and external events. PSA is recognized as a fundamental tool to prioritize properly design and operational issues which are more significant to safety, contributing in this way to a proper balance between costs and safety effectiveness of Generation IV Nuclear Energy Systems. PSA advantages for a systematic understanding and evaluation of risk uncertainties is also remarked. It is expected that PSA will contribute to understand differences in the level of safety of diverse technical proposals and select the designs that better fulfill the selection safety criteria for a given Generation IV reactor concept.

The traditional PSA metrics of Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) remain useful, although the first one does not apply to all the new reactor concepts. It has been recommended the generalization of CDF as "undesirable event with significant source term mobilization". The principal risk metric that should be used for comparing Generation IV concepts is the Farmer's Curve, oriented to the Generation IV Safety objective regarding the elimination of off-site emergency response [5]. To achieve its leading role, PSA must be validated as part of a rigorous quality assurance program, including a peer review conducted by independent experts.

4.10. ISAM: Experiences and perspectives

ISAM is a relatively new methodology which is still under development and adjustment. It is recognized that the methodology will need to be modified or updated based on the lessons and findings derived from the Fukushima accident [5, 44].

An example of a preliminary ISAM application to the Japanese Sodium-cooled Fast Reactor (JSFR) is described in [5]. A PIRT process was conducted for the reactor shutdown by the passive Self-Actuated Shutdown System (SASS) upon an Unprotected Loss of Flow (ULOF) accident. OPT results are presented for the safety function "Core heat removal" of Defense in Depth Level 3, indentifying several mechanisms for the challenge "Degraded or disruption of heat transfer path" and their corresponding LOPs. Applicability of DPA and PSA is also shown.

RSWG demonstration that ISAM can be used to evaluate and document the safety of Generation IV nuclear systems, supported by an extended use of the methodology in practical applications, is only beginning but has excellent perspectives of becoming a future reality. It will certainly depend on international efforts and progress in the materialization of Generation IV reactor concepts. RSWG has been asked for GIF to work toward the provision of increasingly detailed guidance for application of ISAM in the development of Generation IV systems [44].

5. Generation IV: Economical aspects [6]

The economic modeling working group (EMWG) was formed in 2004 for developing a cost estimating methodology to be used for assessing Generation IV International Forum (GIF)

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systems against its economic goals. Its creation followed the recommendations from the economics crosscut group of the Generation IV roadmap project that a standardized cost estimating protocol be developed to provide decision makers with a credible basis to assess, compare, and eventually select future nuclear energy systems, taking into account a robust evaluation of their economic viability. The methodology developed by the EMWG is based upon the economic goals of Generation IV nuclear energy systems, as adopted by GIF: to have a life cycle cost advantage over other energy sources, to have a level of financial risk comparable to other energy projects (i.e., to involve similar total capital investment and capital at risk).

This section briefly describes an economic model for Generation IV nuclear energy systems [6] and the accompanying software [45] in which the guidelines and models were implemented. These tools will integrate cost information prepared by Generation IV system development teams during the development and demonstration of their concept, thus assuring a standard format and comparability among concepts. This methodology will allow the Generation IV International Forum (GIF) Experts Group to give an overview to policy makers and system development teams on the status of available economic estimates for each system and the relative status of the different systems with respect to Generation IV economic goals. Figure 10 displays the structure of the integrated nuclear energy economic model. The following discussion is based on this figure.

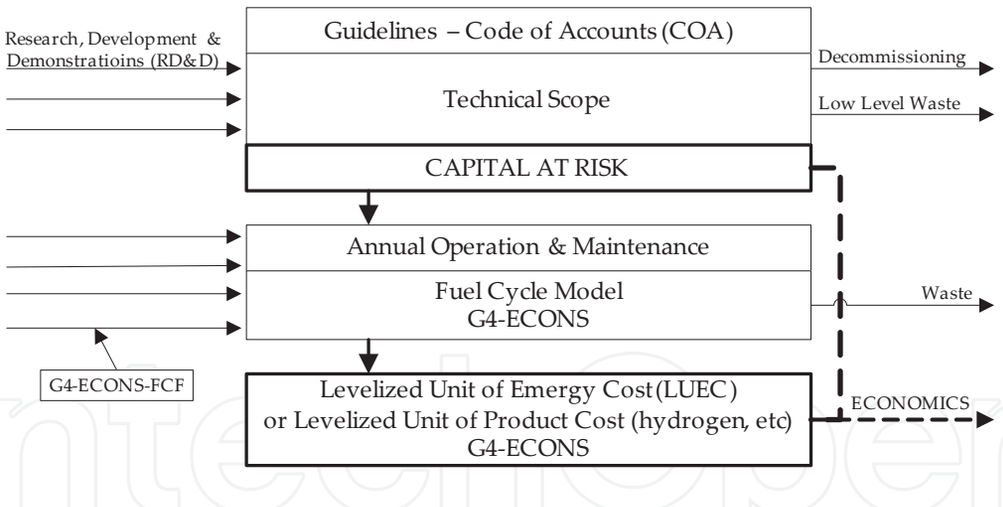


Figure 10. Structure of the integrated nuclear energy economic model [6].

The model is split in four parts: construction/production, fuel cycle, energy products, and modularization.

Cost estimates prepared by system design teams should report the overall direct and indirect costs for reactor system design and construction (base construction cost) and an estimate of the reactor annual operation and maintenance costs. The intent is that these costs be developed using the GIF COA described in [6], prepared by the methods outlined therein. The decision maker, however, needs more than just the overall costs in each life cycle category. Of particular interest are the cost per kilowatt of installed capacity and the cost of elec-

tricity generation (cost per kilowatt-hour) from such systems, including the contribution of capital and non-fuel operations.

Ref. [6] describes how interest during construction (IDC), contingencies, and other supplemental items are added to the base construction cost to obtain the total project capital cost. This total cost is amortized over the plant economic life so that the capital contribution to the levelized unit of energy cost (LUEC) can be calculated. Operation and maintenance (O&M) and decontamination and decommissioning (D&D) costs, along with electricity production information, yield the contributions of non-fuel costs to the overall cost of electricity. These algorithms have been derived from earlier Oak Ridge National Laboratory (ORNL) nuclear energy plant databases (NECDB) [46, 47], to calculate these costs.

Fuel cycle materials and services are purchased separately by the utility or the fuel subcontractor. For fuel cycles commercially deployed, there are mature industries worldwide that can provide these materials and services. Markets are competitive, and prices are driven by supply and demand. The fuel cycle model requires as inputs the amount of fuel needed for the initial core and subsequent equilibrium cores, along with the fissile enrichment of the uranium or plutonium, and, for uranium, the transaction tails assay assumed by the enrichment service provider. The EMWG model uses algorithms similar to those described in [48] to estimate the overall cost for each step and ultimately the unit cost contribution of fuel to electricity cost. Background material on the economic aspects of fuel cycle choices including information on nuclear materials and fuel cycle service unit costs for conventional reactor types that use commercially available fuels can be found in NEA reports [48, 49]. These documents include cost data on fuel reprocessing and high-level waste disposal for closed fuel cycles and spent fuel disposal for the once-through option.

Innovative fuel cycles or fuel cycle steps for which no industrial scale or commercial facilities currently exist, especially for fuel fabrication, reprocessing, and waste disposal are also addressed. For example, the Very-High-Temperature Reactor system will require high-temperature particle fuel and the SFR (Sodium-Cooled Fast Reactor) system might require innovative pyrometallurgical and pyrochemical facilities for fuel fabrication, reprocessing, and re-fabrication. For such systems, price data for fuel cycle services generally are not readily available. Therefore, a unit cost of fuel cycle services, such as $/kgHM (heavy metal) for fuel fabrication, should be calculated using a methodology similar to that used for LUEC calculation for the reactor system. The design team must supply data on the design and construction costs for the facilities, along with an estimate of their annual production rates and operation costs. Algorithms discussed in [6] can produce rough approximations of the unit costs.

The heat generated by some Generation IV systems has the potential for uses other than electricity generation, such as the production of hydrogen by thermal cracking of steam. There are also possible co-production models where the heat is used for both electricity production and process heat applications. The energy products model deals with these issues and is also discussed.

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Cost issues and possible economic benefits that might result from modularization or factory production of all or part of a reactor system are also discussed [6].

The aim is to furnish a standardized cost estimating protocol to provide decision makers with a credible basis to assess, compare, and eventually select future nuclear energy systems taking into account a robust evaluation of their economic viability. To provide a credible, consistent basis for the estimated costs, early estimates of the evolving design concepts are expected to be based on conventional construction experience of built plants. This limitation is desirable from a consistency point of view because it can provide a reasonable starting point for consistent economic evaluation of different reactor concepts.

Of particular interest in this sense is the work by Hejzlar et al [50], related to the Gas Cooled Fast Reactor (GFR). They discuss challenges posed by the GFR when striving for the achievement of balance among the Generation IV goals. According to Carelli et al [51], the nuclear option has to face not only the public opinion sensibility, mainly related to plant safety and waste disposal issues, but also the economic evaluation from investors and utilities, particularly careful on that energy source and in deregulated markets. Smaller size nuclear reactors can represent a viable solution especially for developing countries, or countries with not-highly-infrastructured and interconnected grids, or even for developed countries when limitation on capital at risk applies. A description of Small-Medium size Reactor (SMR) economic features is presented, in a comparison with the state-of-the-art Large size Reactors. A preliminary evaluation of the capital and operation and maintenance (O&M) costs shows that the negative effects of the economies of scale can be balanced by the integral and modular design strategy of SMRs.

6. Proliferation resistance and physical protection [7]

Technical and institutional characteristics of Generation IV systems are used to evaluate system response and determine its resistance against proliferation threats and robustness against sabotage and terrorism. System response outcomes are expressed in terms of a set of measures.

The methodology is organized to permit evaluations to be performed at the earliest stages of system design and to become more detailed and more representative as design evolves. Uncertainty of results is incorporated into the evaluation.

The results are intended for three types of users: system designers, program policy makers, and external stakeholders. Program policy makers will be more likely to be interested in the high-level results that discriminate among choices, while system designers and safeguards experts will be more interested in results that directly relate to design options that will improve their performance (e.g., safeguards by design).

The proliferation resistance and physical protection Working Group has based its specification of the evaluation methodology scope on the definition of the Generation IV proliferation resistance and physical protection goal. The Generation IV Technology Roadmap (DOE,

2002b) [52] formally defined the following proliferation resistance and physical protection goal for future nuclear energy systems:

Generation IV nuclear energy systems will increase the assurance that they are a very unattractive and the least desirable route for diversion or theft of weapons-usable materials, and provide increased physical protection against acts of terrorism.

The definition of proliferation resistance adopted by the Working Group agrees with the definition established at the international workshop sponsored by the International Atomic Energy Agency (IAEA, 2002b)[53].

Formal definitions of proliferation resistance and physical protection have been established as presented next.

Proliferation resistance is that characteristic of a nuclear energy system that prevents the diversion or undeclared production of nuclear material and the misuse of technology by the host state seeking to acquire nuclear weapons or other nuclear explosive devices.

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

The proliferation resistance and physical protection technology goal for Generation IV nuclear energy systems, when combined with the definitions of proliferation resistance and physical protection, is therefore as follows.

A Generation IV nuclear energy system is to be the least desirable route to proliferation by hindering the diversion of nuclear material from the system and hindering the misuse of the nuclear energy system and its technology in the production of nuclear weapons or other nuclear explosive devices.

A Generation IV nuclear energy system is to provide enhanced protection against theft of materials suitable for nuclear explosives or radiation dispersal devices and enhanced protection against sabotage of facilities and transportation. The proliferation resistance and physical protection methodology provides the means to evaluate Generation IV nuclear energy systems with respect to the following categories of proliferation resistance and physical protection threats:

Proliferation Resistance – Resistance to a host state's acquisition of nuclear weapons by:

- Concealed diversion of material from declared flows and inventories;
- Overt diversion of material from declared flows and inventories;
- Concealed material production or processing in declared facilities;
- Overt material production or processing in declared facilities;
- Concealed material production or processing by replication of declared equipment in clandestine facilities.

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- Physical Protection (robustness)
- Theft of nuclear weapons-usable material or information from facilities or transportation;
- Theft of hazardous radioactive material from facilities or transportation for use in a dispersion weapon (radiation dispersal device or "dirty bomb");
- Sabotage at a nuclear facility or during transportation with the objective to release radioactive material to harm the public, damage facilities, or disrupt operations.

Figure 11 illustrates the most basic methodological approach. For a given system, analysts define a set of challenges, analyze system response to these challenges, and assess outcomes. The challenges to the nuclear energy system are the threats posed by potential proliferate states and by sub-national adversaries. The technical and institutional characteristics of Generation IV systems are used to evaluate the system response and determine its resistance to proliferation threats and robustness against sabotage and terrorism. The outcomes of the system response are expressed in terms of proliferation resistance and physical protection measures.

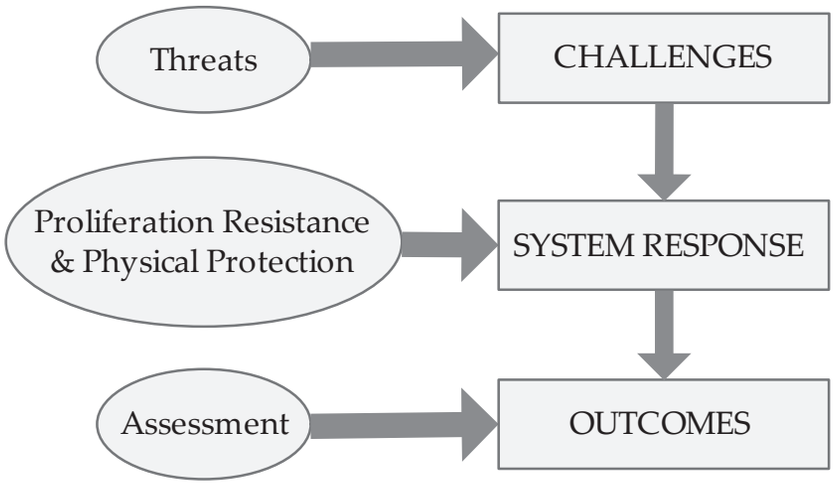


Figure 11. Basic framework for the proliferation resistance and physical protection evaluation methodology [7]

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

The structure for the proliferation resistance and physical protection evaluation can be applied to the entire fuel cycle or to portions of a nuclear energy system. The methodology is organized as a progressive approach to allow evaluations to become more detailed and more representative as system design evolves. Proliferation resistance and physical protec-

tion evaluations should be performed at the earliest stages of design when flow diagrams are first developed in order to systematically integrate proliferation resistance and physical protection robustness into the designs of Generation IV nuclear energy systems along with the other high-level technology goals of sustainability, safety and reliability, and economics. This approach provides early, useful feedback to designers, program policy makers, and external stakeholders from basic process selection (e.g., recycling process and type of fuel), to detailed layout of equipment and structures, to facility demonstration testing.

Figure 12 provides an expanded outline of the methodological approach. The first step is threat definition. For both proliferation resistance and physical protection, the threat definition describes the challenges that the system may face and includes characteristics of both the actor and the actor's strategy. For proliferation resistance, the actor is the host state for the nuclear energy system, and the threat definition includes both the proliferation objectives and the capabilities and strategy of the host state. For physical protection threats, the actor is a sub-national group or other non-host state adversary. The physical protection actors' characteristics are defined by their objective, which may be either theft or sabotage, and their capabilities and strategies.

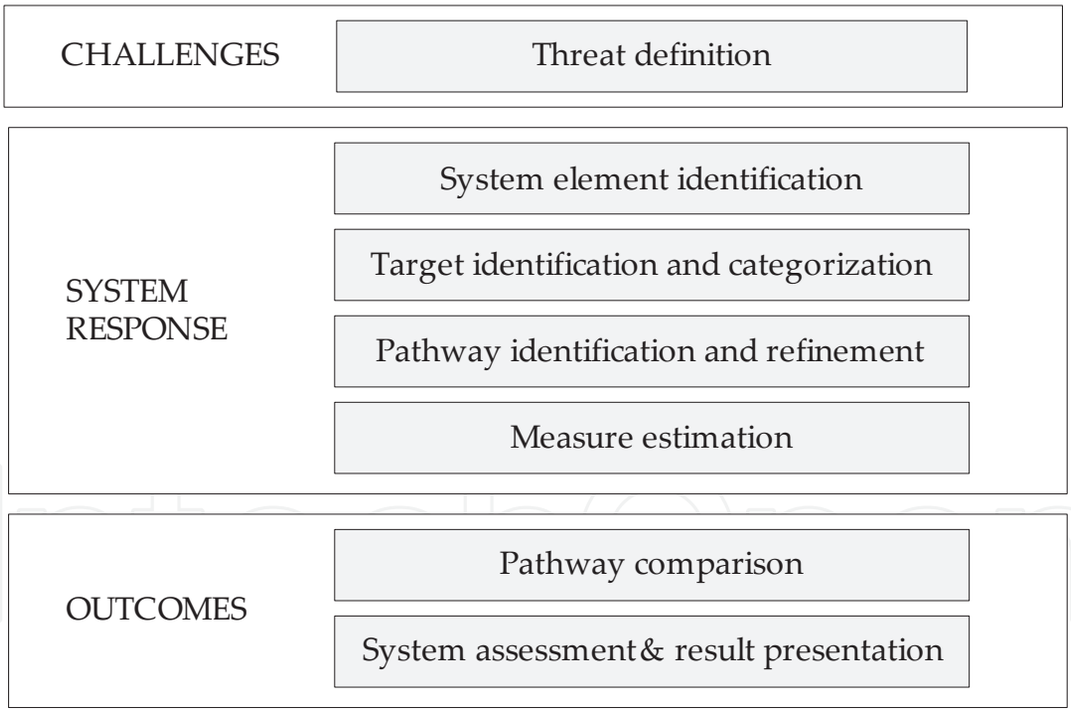


Figure 12. Framework for the proliferation resistance and physical protection evaluation methodology [7]

The proliferation resistance and physical protection methodology does not determine the probability that a given threat might or might not occur. Such evaluations may come from national threat evaluation organizations. The proliferation resistance and physical protection evaluation is based on design features of facilities as well as institutional considerations. Therefore, the selection of what potential threats to include is performed at the beginning of

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a proliferation resistance and physical protection evaluation, preferably with input from a peer review group organized in coordination with the evaluation sponsors. The uncertainty in the system response to a given threat is then evaluated independently of the probability that the system would ever actually be challenged by the threat. In other words, proliferation resistance and physical protection evaluations are challenge dependent.

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

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

1. System Element Identification. The nuclear energy system is decomposed into smaller elements or subsystems at a level amenable to further analysis. The elements can comprise a facility (in the systems engineering sense), part of a facility, a collection of facilities, or a transportation system within the identified nuclear energy system where acquisition (diversion) or processing (proliferation resistance) or theft/sabotage (physical protection) could take place.

2. Target Identification and Categorization. Target identification is conducted by systematically examining the nuclear energy system for the role that materials, equipment, and processes in each element could play in each of the strategies identified in the threat definition. Proliferation resistance targets are nuclear material, equipment, and processes to be protected from threats of diversion and misuse. Physical protection targets are nuclear material, equipment, or information to be protected from threats of theft and sabotage. Targets are categorized to create representative or bounding sets for further analysis.

3. Pathway Identification and Refinement. Pathways are potential sequences of events and actions followed by the actor to achieve objectives. For each target, individual pathways are divided into segments through a systematic process, and analyzed at a high level. Segments are then connected into full pathways and analyzed in detail. Selection of appropriate pathways will depend on the scenarios themselves, the state of design information, the quality and applicability of available information, and the analyst's preferences.

4. Estimation of Measures. The results of the system response are expressed in terms of proliferation resistance and physical protection measures. Measures are the high-level characteristics of a pathway that affect the likely decisions and actions of an actor and therefore are used to evaluate the actor’s likely behavior and outcomes. For each measure, the results for each pathway segment are aggregated as appropriate to compare pathways and assess the system so that significant pathways can be identified and highlighted for further assessment and decision making

The measures for proliferation resistance are:

- Proliferation Technical Difficulty – The inherent difficulty, arising from the need for technical sophistication and materials handling capabilities, required to overcome the multiple barriers to proliferation.
- Proliferation Cost – The economic and staffing investment required to overcome the multiple technical barriers to proliferation, including the use of existing or new facilities.
- Proliferation Time – The minimum time required to overcome the multiple barriers to proliferation (i.e., the total time planned by the Host State for the project).
- Fissile Material Type – A categorization of material based on the degree to which its characteristics affect its utility for use in nuclear explosives.
- Detection Probability – The cumulative probability of detecting a proliferation segment or pathway.
- Detection Resource Efficiency – The efficiency in the use of staffing, equipment, and funding to apply international safeguards to a nuclear energy system.
- The measures for physical protection are:
- Probability of Adversary Success – The probability that an adversary will successfully complete the actions described by a pathway and generate a consequence.
- Consequences – The effects resulting from the successful completion of the adversary's action described by a pathway.
- Physical Protection Resources – The staffing, capabilities, and costs required to provide PP, such as background screening, detection, interruption, and neutralization, and the sensitivity of these resources to changes in the threat sophistication and capability.

By considering these measures, system designers can identify design options that will improve system proliferation resistance and physical protection performance. For example, designers can reduce or eliminate active safety equipment that requires frequent operator intervention.

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

Further literature on the subject of this section comprises, for example, the paper by Bari et al [54], where the general methodology for proliferation resistance and physical protection is discussed and applied. An application to an example sodium fast reactor is discussed in terms of elicitation in [55]. An application concerning nuclear fuel cycles is discussed in [56]. A practical tool to assess proliferation resistance of nuclear energy systems is discussed in [57].

Proliferation resistance is discussed in [58] concerning the mobile fuel reactor, which is not one of Generation IV concepts in discussion, but interesting insights may be found therein. Penner

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et al [59] discuss new reactor designs and construction where a Generation IV design perspective is presented and proliferation resistance is set as an issue of utmost importance. Lennox et al [60] discuss the plutonium issue from the point of view of Generation IV designs.

The molten salt reactor is focused on proliferation issues in Ref. [61] also. Here proliferation considerations are discussed in face of the reactor operation because without the removal of plutonium and uranium from the fuel mixture, the reactivity starts to fluctuate and needs compensation. Uri and Engel [62] discuss non-proliferation attributes of molten salt reactors, as, for example, less plutonium stocks.

Myths of the proliferation resistance approach are focused in Ref. [63].

7. Conclusions

The discussion presented in this chapter clearly shows that much effort has been developed on a worldwide basis for conceiving the reactors that will be in use around 2030. Due to its beginning as a military weapon, nuclear energy is not an energy option that is accepted without strong restrictions in many countries. This resistance has been particularly aggravated immediately after the accidents in Three Mile Island, Chernobyl and the recent one in Fukushima. Many lessons learned from these accidents have been employed in the conception of Generation IV reactors, as many of them had already been implemented in Generation III reactors, like Westinghouse's AP1000.

The Generation IV philosophy for reactor development brings into light concerns about sustainability, economic viability, safety, and security translated into the concepts of proliferation resistance and physical protection. These are new concepts that are playing the dominante roles in reactor development for the future. It is also noteworthy that safety analysis is to be stressed, mainly the application of the risk-informed decision making approach for licensing purposes. Certainly, this integrated phylosophy will do much for turning nuclear energy systems much more acceptable by the final users.

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