IAEA-TECDOC-2010

IAEA TECDOC SERIES

IAEA-TECDOC-2010

Approach and Methodology for the Development of Regulatory Safety Requirements for the Design of Advanced Nuclear Power Reactors

Case Study on Small Modular Reactors

IAEA

International Atomic Energy Agency

IAEA SAFETY STANDARDS AND RELATED PUBLICATIONS

IAEA SAFETY STANDARDS

Under the terms of Article III of its Statute, the IAEA is authorized to establish or adopt standards of safety for protection of health and minimization of danger to life and property, and to provide for the application of these standards.

The publications by means of which the IAEA establishes standards are issued in the **IAEA Safety Standards Series**. This series covers nuclear safety, radiation safety, transport safety and waste safety. The publication categories in the series are **Safety Fundamentals**, **Safety Requirements** and **Safety Guides**.

Information on the IAEA's safety standards programme is available at the IAEA Internet site

www.iaea.org/resources/safety-standards

The site provides the texts in English of published and draft safety standards. The texts of safety standards issued in Arabic, Chinese, French, Russian and Spanish, the IAEA Safety Glossary and a status report for safety standards under development are also available. For further information, please contact the IAEA at: Vienna International Centre, PO Box 100, 1400 Vienna, Austria.

All users of IAEA safety standards are invited to inform the IAEA of experience in their use (e.g. as a basis for national regulations, for safety reviews and for training courses) for the purpose of ensuring that they continue to meet users' needs. Information may be provided via the IAEA Internet site or by post, as above, or by email to Official.Mail@iaea.org.

RELATED PUBLICATIONS

The IAEA provides for the application of the standards and, under the terms of Articles III and VIII.C of its Statute, makes available and fosters the exchange of information relating to peaceful nuclear activities and serves as an intermediary among its Member States for this purpose.

Reports on safety in nuclear activities are issued as **Safety Reports**, which provide practical examples and detailed methods that can be used in support of the safety standards.

Other safety related IAEA publications are issued as **Emergency Preparedness and Response** publications, **Radiological Assessment Reports**, the International Nuclear Safety Group's **INSAG Reports**, **Technical Reports** and **TECDOCs**. The IAEA also issues reports on radiological accidents, training manuals and practical manuals, and other special safety related publications.

Security related publications are issued in the **IAEA Nuclear Security Series**.

The **IAEA Nuclear Energy Series** comprises informational publications to encourage and assist research on, and the development and practical application of, nuclear energy for peaceful purposes. It includes reports and guides on the status of and advances in technology, and on experience, good practices and practical examples in the areas of nuclear power, the nuclear fuel cycle, radioactive waste management and decommissioning.

APPROACH AND METHODOLOGY FOR THE DEVELOPMENT OF REGULATORY SAFETY REQUIREMENTS FOR THE DESIGN OF ADVANCED NUCLEAR POWER REACTORS

The following States are Members of the International Atomic Energy Agency:

AFGHANISTAN
ALBANIA
ALGERIA
ANGOLA
ANTIGUA AND BARBUDA
ARGENTINA
ARMENIA
AUSTRALIA
AUSTRIA
AZERBAIJAN
BAHAMAS
BAHRAIN
BANGLADESH
BARBADOS
BELARUS
BELGIUM
BELIZE
BENIN
BOLIVIA, PLURINATIONAL STATE OF
BOSNIA AND HERZEGOVINA
BOTSWANA
BRAZIL
BRUNEI DARUSSALAM
BULGARIA
BURKINA FASO
BURUNDI
CAMBODIA
CAMEROON
CANADA
CENTRAL AFRICAN REPUBLIC
CHAD
CHILE
CHINA
COLOMBIA
COMOROS
CONGO
COSTA RICA
CÔTE D'IVOIRE
CROATIA
CUBA
CYPRUS
CZECH REPUBLIC
DEMOCRATIC REPUBLIC OF THE CONGO
DENMARK
DJIBOUTI
DOMINICA
DOMINICAN REPUBLIC
ECUADOR
EGYPT
EL SALVADOR
ERITREA
ESTONIA
ESWATINI
ETHIOPIA
FIJI
FINLAND
FRANCE
GABON
GEORGIA
GERMANY
GHANA
GREECE
GRENADA
GUATEMALA
GUYANA
HAITI
HOLY SEE
HONDURAS
HUNGARY
ICELAND
INDIA
INDONESIA
IRAN, ISLAMIC REPUBLIC OF
IRAQ
IRELAND
ISRAEL
ITALY
JAMAICA
JAPAN
JORDAN
KAZAKHSTAN
KENYA
KOREA, REPUBLIC OF
KUWAIT
KYRGYZSTAN
LAO PEOPLE'S DEMOCRATIC REPUBLIC
LATVIA
LEBANON
LESOTHO
LIBERIA
LIBYA
LIECHTENSTEIN
LITHUANIA
LUXEMBOURG
MADAGASCAR
MALAWI
MALAYSIA
MALI
MALTA
MARSHALL ISLANDS
MAURITANIA
MAURITIUS
MEXICO
MONACO
MONGOLIA
MONTENEGRO
MOROCCO
MOZAMBIQUE
MYANMAR
NAMIBIA
NEPAL
NETHERLANDS
NEW ZEALAND
NICARAGUA
NIGER
NIGERIA
NORTH MACEDONIA
NORWAY
OMAN
PAKISTAN
PALAU
PANAMA
PAPUA NEW GUINEA
PARAGUAY
PERU
PHILIPPINES
POLAND
PORTUGAL
QATAR
REPUBLIC OF MOLDOVA
ROMANIA
RUSSIAN FEDERATION
RWANDA
SAINT KITTS AND NEVIS
SAINT LUCIA
SAINT VINCENT AND THE GRENADINES
SAMOA
SAN MARINO
SAUDI ARABIA
SENEGAL
SERBIA
SEYCHELLES
SIERRA LEONE
SINGAPORE
SLOVAKIA
SLOVENIA
SOUTH AFRICA
SPAIN
SRI LANKA
SUDAN
SWEDEN
SWITZERLAND
SYRIAN ARAB REPUBLIC
TAJIKISTAN
THAILAND
TOGO
TONGA
TRINIDAD AND TOBAGO
TUNISIA
TÜRKİYE
TURKMENISTAN
UGANDA
UKRAINE
UNITED ARAB EMIRATES
UNITED KINGDOM OF GREAT BRITAIN AND NORTHERN IRELAND
UNITED REPUBLIC OF TANZANIA
UNITED STATES OF AMERICA
URUGUAY
UZBEKISTAN
VANUATU
VENEZUELA, BOLIVARIAN REPUBLIC OF
VIET NAM
YEMEN
ZAMBIA
ZIMBABWE

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".

IAEA-TECDOC-2010

APPROACH AND METHODOLOGY FOR THE DEVELOPMENT OF REGULATORY SAFETY REQUIREMENTS FOR THE DESIGN OF ADVANCED NUCLEAR POWER REACTORS

CASE STUDY ON SMALL MODULAR REACTORS

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2022

For further information on this publication, please contact:

Safety Assessment Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria
Email: Official.Mail@iaea.org

Printed by the IAEA in Austria
September 2022

IAEA Library Cataloguing in Publication Data

Names: International Atomic Energy Agency.
Title: Approach and methodology for the development of regulatory safety requirements for the design of advanced nuclear power reactors / International Atomic Energy Agency.
Description: Vienna : International Atomic Energy Agency, 2022. | Series: IAEA TECDOC series, ISSN 1011–4289 ; no. 2010 | Includes bibliographical references.
Identifiers: IAEAL 22-01536 | ISBN 978–92–0–139922–9 (paperback : alk. paper) | ISBN 978–92–0–139822–2 (pdf)
Subjects: LCSH: Nuclear power plants. | Nuclear reactors — Safety regulations. | Nuclear reactors — Safety measures. | Nuclear power plants — Design and construction.

FOREWORD

Member State interest in advanced nuclear power plant designs, particularly small and medium sized or modular reactors (SMRs), has increased in recent years. SMRs are seen as a viable option for electricity production as well as for non-electric applications to meet climate change goals. Advanced nuclear power plant designs, including SMRs, currently in the design or licensing phases encompass a variety of reactor technologies. These technologies are not limited to water cooled reactors but also include high temperature gas cooled reactors, molten salt reactors and liquid metal cooled fast reactors. SMR designs put forward several advanced design safety features. The hazards and challenges to the physical barriers associated with advanced nuclear power plant designs and advanced design safety features are different from those associated with water cooled reactors.

The IAEA safety requirements for the design of structures, systems and components of nuclear power plants, established in IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design, were primarily developed for land based, water cooled stationary nuclear power plants designed for electricity generation or heat production applications. Therefore applying these requirements to other reactor technologies or reactor designs needs to also include engineering judgement.

Even though considerable progress has been made in both the design and licensing of nuclear power plants, and SMRs in particular, there is a need for further guidance to assist Member States in developing, adapting or updating national safety requirements for nuclear power plant designs. Taking the current safety approach for advanced nuclear power plant designs into consideration, this publication presents a methodology that applies an integrated risk informed, objective oriented, performance based approach. This methodology aims to assist Member States in applying the engineering judgement process to assess how safety requirements for the design of structures, systems and components for advanced nuclear power reactors can be implemented. The methodology presented in this publication considers the advanced design safety features of SMRs, including specific examples.

The IAEA would like to thank those experts who participated in the consultancy meetings organized for the preparation of this publication and contributed to its review. The IAEA officer responsible for this publication was J. Luis Hernandez of the Division of Nuclear Installation Safety.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

Advanced nuclear power plant (NPP) designs include both innovative and evolutionary reactor designs encompassing different reactor technologies including water cooled and non-water cooled [1]. Among them, small and medium sized or modular reactor (SMR) designs have been the subject for discussion in several IAEA fora from the 1960s [2] and 1970s [3] until more recently, for example 2013 [4], 2014–2021 [5], 2017 [6], and 2019 [7]. In the last years, advanced NPP designs, together with SMR designs, are considered as a viable option for electricity production as well as for non-electrical applications [8].

The IAEA booklet "Advances in Small Modular Reactor Technology Developments – A Supplement to IAEA Advanced Reactors Information System (ARIS)" [9] identifies more than 50 different SMR designs, considering different reactor technologies including a variety of coolants, nuclear fuel and neutron spectra. Recently, significant progress has been made both in the reactor design requirements and the licensing processes for SMRs, especially for two technological categories: water cooled reactors (WCRs, such as NuScale, CAREM-25 and Akademik Lomonosov) and high temperature gas cooled reactors (HTGRs, such as HTR-PM and GTHTR300). Correspondingly to this progress and the need to ensure reliable and sustainable energy sources, successful deployment of SMRs is expected to generate improved performance and economical effectiveness [10], and may also result in a significantly increased level of safety.

Ensuring a high level of safety for advanced nuclear power reactors, and particularly for SMRs, is a very complex and multidimensional task, and it is achieved partly through robust and reliable design provisions. Compared to previous nuclear power plant designs, advanced NPP designs may bring forward features such as more robust designs through a reinforced defence in depth, stronger protection against internal and external hazards, and reduced environmental impact in all plant states, including the frequency of large radioactive releases. Most SMR designs have similar features as advanced NPP designs and their designers claim to incorporate further advanced design safety features including concepts such as modularization, modularity, integral or compact designs and extensive use of passive safety systems. Furthermore, since SMR technologies encompass different nuclear reactor technologies, as advanced NPP designs do, these lead to differences in the associated hazards and challenges to barriers as well as on specific design features compared to large WCRs. The advanced safety features in SMR designs are aimed at better coping with accidents, e.g. providing a larger grace period time for operator action after the onset of anticipated operational occurrences or accident conditions. Additionally, advanced NPP designs, including SMRs, aim at providing further substantiation for the justification of practical elimination of accident sequences leading to large or early radioactive releases.

Adequacy of design provisions for ensuring safety for all NPP designs is to be evaluated in the licensing process in each individual State in accordance with the national legal and regulatory framework. Advanced NPP designs are currently being considered by several States for design and construction [1]. Along with the deployment of advanced NPP designs, other States, including both States embarking on a nuclear power programme and States with an already established legal and regulatory framework, are interested in including SMRs in their nuclear portfolio. Such framework might be only partially directly applicable for advanced nuclear power reactor designs, which may be significantly different from technologies used in operating NPPs. To facilitate licensing of innovative technologies in those States, there might be a need to update the existing regulatory design requirements or at least to develop guidance for interpretation of existing regulatory design requirements. States such as Canada, China, Japan, Russian Federation, the United Kingdom and the USA have already undertaken reviews of their regulatory requirements and guidance to assess the applicability to these technologies.

In response to such need, this TECDOC is intended to help in the process of updating the regulatory guidance for licensing of advanced NPP designs, and particularly for SMRs, in States with an established system of nuclear regulations and guides. This publication identifies the typical areas for updating and proposes the basic steps for such updating. It can also be helpful to States embarking on nuclear power programmes in their preparation for development of their own legislation.

IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [11], could be used as a reference publication for such updating. SSR-2/1 (Rev. 1) [11] establishes key design safety requirements for new land based stationary NPPs with WCRs, including SMRs, such as those related to design extension conditions (DECs) and the practical elimination of accident sequences that could lead to an early or large radioactive release. Additionally, SSR-2/1 (Rev. 1) [11] reinforces the application of the defence in depth (DiD) concept by requiring, as far as practicable, the independence of design safety features considered at different levels of DiD.

It is well recognized that the specific safety requirements for the design of NPPs presented in SSR-2/1 (Rev. 1) [11] are built on the consensus of Member States, supported by the large operating experience with WCRs. They incorporate technology neutral requirements in section 3 ‘Management of safety in design’ and section 4 ‘Principal technical requirements’, technology inclusive requirements in section 5 ‘General plant design’, and technology specific requirements in section 6 ‘Design of specific plant systems’. Therefore, those design requirements are primarily applicable to land based, stationary NPPs with WCRs, as some advanced NPP designs are. For NPPs with reactor technologies other than WCRs, the application of those requirements may be applied with a consideration of engineering judgement, to determine the requirements that are appropriate for new NPP designs. IAEA-TECDOC-1936, Applicability of Design Safety Requirements to Small Modular Reactor Technologies Intended for Near-term Deployment [12] presents a preliminary analysis of the applicability of the requirements established in SSR-2/1 (Rev. 1) [11] for to two specific SMR reactor technologies (WCR and HTGR). The requirements in SSR-2/1 (Rev. 1) [11], complemented by Ref. [12], can be applied, with judgement, to other SMR technologies.

A general methodology for developing technology neutral safety requirements for innovative reactor designs has been proposed in IAEA-TECDOC-1570, Proposal for a Technology-Neutral Safety Approach for New Reactor Designs [13], published in 2007. The present publication is based on the methodology introduced in Ref. [13] but reflects the currently valid IAEA safety standards and the latest status in the development of SMRs.

This TECDOC has interfaces with several other IAEA activities as well as other publication already published or being developed. There are several activities aimed at supporting technology developments of advanced NPP designs, and particularly for SMRs, which are beyond the scope of this publication, focused on safety issues only. This TECDOC takes into account the outcomes of the SMR Regulators’ Forum, which evaluated the applicability of some IAEA safety standards to SMRs and identified the most relevant issues for the IAEA to develop additional guidance [14]. As a valuable source of information about various SMR specific design features, the IAEA booklet on SMRs [9], published since 2012 at two-year intervals, has been used. In addition to the above-mentioned publications, a Safety Report is currently under preparation to offer a high level view on the whole system of IAEA safety standards with regard to advanced NPP designs, in particular SMRs. Other publications under preparation will go in deeper detail, for example, to address issues associated with the safety design approach and implementation of the DiD concept, deterministic and probabilistic safety analysis for main SMR technologies, review of applicability of Safety Guides for the design of systems for SMR technologies, and safety, security and safeguards by design for SMRs.

1.2. OBJECTIVE

This TECDOC aims at proposing a technology neutral methodology for adaptation of the regulatory safety requirements considered for the design of NPPs to be applicable for advanced NPP designs and proposed a case study on SMRs. The methodology is based on the review performed on the existing safety requirements for NPP design, established in SSR-2/1 (Rev. 1) [11]. Key issues associated with the design of SMRs are broadly discussed. Applicability of the methodology is demonstrated by representative examples of selected safety topics and design safety requirements.

The TECDOC is intended mainly for use by regulatory bodies and their technical support organizations but can also be used by other organizations involved in design, licensing, manufacture, construction, modification, maintenance, operation and decommissioning of SMRs. The proposed methodology is expected to be primarily used by States with established regulatory frameworks covering design requirements for existing NPPs, but may also be helpful for States embarking in nuclear power programmes.

The TECDOC is also intended to contribute to harmonization of approaches and standards in Member States when establishing regulatory design safety requirements during the licensing processes of advanced NPP designs, with a focus on SMRs.

1.3. SCOPE

The TECDOC focuses on the proposed methodology to assess how the safety requirements for the design of NPPs, established in SSR-2/1 (Rev. 1) [11], need to be adapted to be applicable for advanced NPP designs, with a focus on SMRs, in a technology neutral manner.

The TECDOC covers design safety features for reactor technologies used in the most typical SMRs in a technology neutral manner (for both water cooled and non-water cooled SMRs). The proposed methodology is illustrated by several examples of the adaptation of design safety requirements from SSR-2/1 (Rev. 1) [11].

The scope of this TECDOC does not include other safety requirements – such as those related to site evaluation, safety assessment, operation, radiation protection, and regulatory framework for safety – which might need to be revised for adaptation.

1.4. STRUCTURE

This TECDOC consists of three sections and one appendix.

Following the introduction, Section 2 describes the most typical SMR designs and discusses key design features considered as important for updating design safety requirements.

Section 3 introduces the risk informed, objective oriented, performance based approach and presents the proposed stepwise methodology and review process for developing or updating national regulatory design safety requirements.

The Appendix provides examples of the application of the proposed methodology as a demonstration of its applicability for different SMR safety topics and various safety requirements.

2. CONSIDERATION OF DESIGN FEATURES IN THE SPECIFICATION OF SAFETY REQUIREMENTS FOR SMALL MODULAR REACTORS

2.1. DESIGN CHARACTERISTICS OF TYPICAL SMALL MODULAR REACTORS

For updating national design related safety requirements, it is important to take into account specific SMR design features and key differences from the large traditional WCRs. The majority of SMRs intended for production of electricity have an electric power below 300 MW_e, which is usually considered as an upper power bound for SMRs.

A SMR module is considered mainly as a nuclear steam supply system[1] [14]. A modular reactor is defined as a nuclear power reactor and its associated structures, systems and components, and the term 'modular' (see Section 2.2.4) means capability to allow two or more reactor modules to be built in the same NPP unit or in the same nuclear island [14]. Reactor modules are essentially identical. Generally, each reactor module can be operated independently from others.

In accordance with Ref. [9] with the overview of SMRs considered as potential for near or middle term deployment, the following four groups of SMR technologies can be recognized:

(a) Water cooled reactors (WCRs);
(b) High temperature gas cooled reactors (HTGRs);
(c) Liquid metal cooled fast neutron reactors (LMFRs);
(d) Molten salt reactors (MSRs).

[1] As stated in Appendix III of Ref. [14], "the definition of SMR 'module' may be better interpreted as 'nuclear installation' or nuclear steam supply system (safety classified part of the primary and secondary circuit for PWR) than as 'plant'."

As an example, Table 1 compares selected design features for SMR technologies based on a literature review.

TABLE 1. COMPARISON OF SELECTED DESIGN FEATURES FOR SMR TECHNOLOGIES (WITH ELECTRIC POWER RANGE UP TO 300 MW_E)

Parameter	**SMR type**			
	WCR	**HTGR**	**LMFR**	**MSR**
Range of thermal power per reactor module (MW_{th})	100–1000	100–625	30–950	20–750
Primary coolant	H_2O, pressurized or boiling	He	NA, Pb, Pb/Bi	Fluoride or chloride salt
Moderator	H_2O or D_2O	Graphite	N/A	Graphite, D_2O,
Fuel type / enrichment	UO_2, 3–20%	TRISO or hexagonal, UO_2, U, Th, Pu, usually up to 20%	UO_2, U, UC, Pu, mixed oxides	U, Th or Pu in fluoride or chloride salt, or TRISO, usually up to 20%
Configuration of reactor coolant system	Integral or compact (external steam generator) PWR or PHWR or BWR	Loop	Pool and loop, with integral or external steam generator	Integral, with internal or external heat exchanger
Cooling of reactor at power	Natural or forced circulation	Forced circulation	Forced circulation	Forced circulation
Approach to safety systems	Passive or active or both (hybrid)	Passive or hybrid	Passive or hybrid	Passive or hybrid
Core damage frequency	10^{-8}–10^{-6} $year^{-1}$	N/A	~10^{-8} $year^{-1}$	N/A, or <10^{-6} $year^{-1}$
Primary coolant temperature	285–345ºC	750–950ºC	420–535ºC	630–750ºC
Primary coolant pressure	12.25–16.5 MPa	4–9 MPa	0.1 MPa	0.1–1.2 MPa
Duration of fuel campaign	2–25 years	1.5–4 years, or on-line refuelling	5–30 years	7–8 years, or online refuelling
Design lifetime	40–80 years	40–60 years	30–60 years	30–80 years

Note: Values given in the table are typical values or ranges, not necessarily the maximum achievable values according to Ref. [9]. In addition, these values have not been evaluated by the IAEA.

From the brief information provided in Table 1, it is clear that design features of SMRs are often very different from those of existing large NPPs (even in the case of SMRs using light water technology). The design of SMRs is typically focused on broader use of inherent safety features such as:

(a) Strong negative reactivity feedback;
(b) Large thermal inertia;
(c) Use of accident tolerant fuels;
(d) Different coolants than water with high boiling points;
(e) Chemically inert coolants;
(f) Coolants with high retention capability for fission products;
(g) Design considerations on eliminating mechanisms potentially challenging safety functions, such as:

(i) Using integral configurations;
(ii) Using compact configurations;
(iii) Thermally resistant fuel coating materials;
(iv) Eliminating circulation loops;
(v) Reducing number of components;

(h) Increasing reliability of provisions by replacing active systems combined with operator actions by passive systems;
(i) Other various novel design safety features.

Certain design features requiring special attention in updating safety requirements are more closely discussed in Section 2.2. Although the specifics of a certain SMR technology might need further analysis, the representative features discussed below are sufficient to demonstrate the need for careful evaluation of applicability of existing safety requirements to SMRs.

2.2. SELECTED DESIGN FEATURES OF SMALL MODULAR REACTORS IMPORTANT FOR UPDATING OF REGULATORY DESIGN SAFETY REQUIREMENTS

The different reactor technologies and the variety of designs of SMRs might bring a wide range of design features that need to be considered in the review of existing regulatory safety requirements. The following Sections 2.2.1–2.2.15 highlight examples of some of these features.

2.2.1. Modularization

Modularization is recognized, according to Ref. [9], as the possibility to assemble several components of the reactor coolant system (RCS) in the manufacturing factory, thereby facilitating subsequent transport of such assembled modules to the construction site. Many SMR designs consider modularization as a solution to improve cost effectiveness in the manufacturing, construction, transportation and deployment of modules. In addition, modularization offers the opportunity to assemble the reactor module, which may include other components, such as steam generators, or circulation pumps, in the factory, which allows increased standardization, quality assurance and control of the whole implementation process and commissioning tests with a positive impact on the equipment quality and reliability. In addition, assembling the reactor module in the factory could reduce the complexity of the installation with a reduction of potential human errors.

On the other hand, transportation of modules, with nuclear fuel loaded or unloaded, for their installation on the site or for their refuelling in an off-site facility, opens new issues in particular due to the interfaces between safety, security and safeguards. In addition, the modularization of SMRs might challenge the regulatory process in relation to safety requirements for the design of NPPs in relation to proven engineering practices or the provision for construction. Challenges might also exist with respect to the performance of associated regulatory oversight activities, such as in the case of repeated fabrication of modules in the factory, in the transport of modules, in the re-licensing of modules with the same design, in the off-site assembling and testing of modules components, and in the decentralization of design activities and safety analyses.

2.2.2. Integral design

This SMR feature is considered mainly for water cooled SMRs but also for some non-water cooled SMRs [9]. The integration of several components of the RCS inside the reactor vessel allows simplification of the design because of fewer components such as pipes, valves and pumps, thus reducing the number of components that could fail and, in this way, eliminating some postulated initiating events. Examples are the elimination of loss of coolant accidents and loss of flow accidents in some SMR designs.

Integral design also allows for reducing the number of pressure vessel penetrations with a reduction in the probability of the vessel rupture. Simplification of the design also reduces the need for some normal operation activities such as maintenance, in-service inspections, tests and spare part replacement. However, new challenges might arise in relation to the requirements related to ensure radiation protection of personnel during these activities, and to the practical feasibility of those activities due to space constraints. Moreover, the potential increase in probability of failures resulting from necessary re-installation of components associated with the need for disassembling and assembling the modules might challenge the requirements in relation to the design of items important to safety.

Additionally, an integral design may require additional experiments and tests of the novel equipment as well as validation of the computer codes used for design or for deterministic safety analyses, as recommended in IAEA Safety Standards Series No. SSG-2 (Rev. 1), Deterministic Safety Analysis for Nuclear Power Plants [15].

Therefore, an integral design might challenge safety requirements for the design of NPPs in relation to the definition of postulated initiating events, as well as on the key activities related to items important to safety (see Requirement 29 of SSR-2/1 (Rev. 1) [11]).

2.2.3. Compact design

Reducing the size of the components (compact design) allows for reduction in the overall plant footprint and helps with ease of transportation and installation in remote areas. The compact design means a reduced size of the reactor core and, together with low power density, this might reduce source terms compared to large NPPs[2]. A low power density improves the thermal stability of the core and reduces the decay heat [14]. The small size of the SMR core also facilitates implementation of an integral design.

However, the compactness might also add complexity to normal operation due to more demanding needs for radiation protection.

Moreover, owing to the compact core, the hydraulic resistances become more important for core coolability, and further challenged by low driving forces in natural circulation mode of the heat transfer regime when relying solely on passive safety provisions.

Similar to an integral design, the compact design might challenge safety requirements on the design of NPPs in relation to the radiation protection in design and the definition of the postulated initiating events, as well as on the key activities related to items important to safety (see Requirement 29 of SSR-2/1 (Rev. 1) [11]). In addition, the compact design might also

[2] Low power density of the reactor core (NPPs) lead to a reduced source terms, considering that source terms depend on the characteristics (energy, inventory and time) of fission products and radionuclides in the core that could be released to the environment during the progression of a severe accident. Therefore, the source terms depend also on the operation cycle and the characteristics of the retention capabilities (confinement function) in the nuclear fuel, as well as in reactor coolant and containment systems.

impact the safety requirements related to the design of the RCS and the design limits of the fuel and the reactor core, respectively.

2.2.4. Modularity

A SMR module is considered mainly as a nuclear steam supply system [14]. Modularity is understood as the possibility for the subsequent placing of SMR modules (i.e. reactor modules) close to each other in one NPP [14]. This allows a gradual deployment and, thus, higher flexibility in operation. Gradual deployment facilitates operating experience feedback by considering the lessons learned from previous modules, regarding their design, manufacture, construction, installation, and commissioning. The cumulative experience from previous individual modules contributes to the reduction of potential fabrication and installation errors and, thus, to the reduction of the number of failures during operation.

On the other hand, modularity might affect the dependence and independence between design provisions due to the proximity of the modules. Sharing structures, systems and components (SSCs) by several modules, such as the spent fuel pool or the main control room, implies appropriate consideration of potential common cause failures (CCFs), including the mutual impact of several modules due to certain internal or external hazards. Those aspects might challenge Requirement 33 of SSR-2/1 (Rev. 1) [11] related to the independence of safety systems and safety features for DECs, for NPPs with multiple units.

2.2.5. Multiple barriers against releases of radioactive materials

For WCRs, the barriers implemented for confining the fission products in the nuclear fuel are typically:

(a) The fuel matrix;
(b) The fuel cladding;
(c) The reactor coolant pressure boundary and connected systems;
(d) The containment structure (e.g. for PWR and PHWR: large dry volume leaktight pressure containment structure; for BWR: leaktight containment with pressure suppression pool) and associated systems ensuring the confinement function.

The containment structure and associated containment systems in existing large NPPs perform the role of the final barrier, preventing a source of radioactive material from being released to the environment.

The concept of implementing multiple barriers to ensure an effective confinement of radioactive materials, as part of the DiD concept, continues to be essential for SMRs for the fulfilment of the fundamental safety objective. However, the barriers used in different SMR technologies may significantly differ from large WCRs. For instance, based on Ref. [9], the HTGR takes advantage of the properties of the ceramic materials to ensure the maximum retention of fission products (e.g. TRISO coated particle fuel). Other SMR designs, such as LFMRs, use steel cladding instead of Zr-alloy cladding. In other SMR designs, such as LMFRs and MSRs, a guard vessel is added surrounding the low pressure RCS, to be used as a leak jacket in case an accidental leak occurs in the RCS. Additionally, the reactor containment civil structure, as mainly considered for large WCRs, might have different roles depending on the SMR technology and its design.

Due to large differences in the physical nature of the barriers, the behaviour of the barriers and the means for protection of functioning of the barriers need to be considered specifically for each SMR design.

2.2.6. Passive design features and natural circulation

Passive design features and use of natural circulation are not exclusively considered for SMRs, since they have been already implemented in some types of large NPPs. According to Ref. [9], SMRs have the advantage of smaller reactor thermal power allowing extensive implementation of passive systems for performance of safety functions, including reactor core and containment cooling based on natural circulation phenomena. Some SMR designs use natural circulation for reactor cooling during normal operation, while others use it for residual heat removal, eliminating the need for active pumps. Relying on passive systems reduces the potential failures and errors by reducing the number of qualified active components and, consequently, certain operational activities such as maintenance and in-service inspections. However, the passive failure[3] of passive systems still needs to be considered and the justification for its exclusion might challenge requirements related to the single failure criterion. For example, natural circulation relies on small driving forces that might be affected by certain phenomena and might compete with the pressure losses resulting from ageing effects or clogging, which could jeopardize efficient reactor cooling and fuel integrity. In addition, effective and reliable activation and operation of passive systems needs to be further substantiated by research programmes, operating experience feedback, and performance tests. Specific considerations are needed concerning the application of single failure criterion, quantification of equipment reliability and their qualification. Moreover, performing certain normal operation activities on passive systems such as in-service inspections and tests, require specific considerations since such activities may alter the system configuration and, thus, influence its activation or performance.

2.2.7. Human factors

Modularity (understood as several modules in the same NPP unit) and extended use of passive design features have specific implications for reliability of human factors. Use of passive systems and specific design solutions, such as unmanned plant operation with an off-site control room, reduces the need for operator actions and, thus, the frequency of potential operator errors. In addition, incorporating passive safety systems and features together with smaller reactor power may allow for a longer coping time for operator actions required after the occurrence of accident conditions. Such features can contribute to higher human reliability.

Some SMRs consider operations with a reduced number of operators or generally reduced overall staffing level, unmanned or remote operation, and potentially longer operating time between maintenance and refuelling outages.

Another change may be the modified approach to refuelling in some SMRs. For example, pebble bed HTGRs utilize online refuelling and, consequently, there is no specific shutdown reactor operating mode for refuelling. This is a significant difference from the usual operational regimes of light water reactors.

---

[3] A failure affecting the integrity of the component (pressure) boundary which jeopardizes their effective function in a fluid system, such as by internal or external leaks, or by clogging due to chemical formations, particle cumulation or structural deformations.

Challenges due to different conditions for human performance need to be recognized, such as multiple reactor modules sharing a single control room, with the possibility of affecting all operational staff of several modules at the same time due to harsh working conditions induced by internal hazards, such as fire, explosion, and other CCFs. Also, the interface between the reactor and the reprocessing and chemical plants in some applications needs further consideration since the operator actions related to reprocessing and chemical plant can be risk significant or can trigger initiating events in the reactor.

2.2.8. Fuel enrichment and performance

Compared to large NPPs, some SMR designs implement lower core power density. This design feature together with the coolant inventory ensure higher thermal inertia and a higher thermal margin able to cope with accident conditions without fuel damage, and consequently allow longer grace periods for operator actions.

Some SMR designs utilize materials for fuel and its coatings (e.g. TRISO coated particle fuel) with intrinsic characteristics such as high temperature resistance and strong retention of fission products, resulting in increased capability to cope with accident conditions. Moreover, accident tolerant fuels are being developed and their use for SMRs may be expected in the future. New materials for fuel and coatings need substantiation of their design characteristics and performance in different plant states. Moreover, since some SMR designs propose online refuelling (see examples in Ref. [9]), the standards and procedures related to the quality assurance process as well as the planning and scope of regulatory oversight activities need to be adapted for ensuring compliance with fuel requirements.

Some MSR designs use the liquid fuel mixed with a molten salt. Such designs apply specific solutions for reactivity control and reactor shutdown as well as for the retention of fission products based on complex chemical control and fuel treatment facilities to sustain the chemical properties of the fuel salt. This is a significant difference from the current solid fuel experience and requires further substantiation.

Regarding fuel enrichment, some SMR designs [9] are proposing fuel enrichments up to 20%, which is higher than that for conventional large NPPs. While those high fuel enrichments (<20%) have the advantage for longer cycle operation without refuelling and operational flexibility, from the safety perspective requirements, higher fuel enrichment will require careful consideration of the safety analysis for criticality accidents, safety systems performance and safety margins during all plant states, including fuel handling and storage of spent fuel, to avoid fuel damage. Higher fuel enrichment also leads to higher potential source terms in case of accidental radiological release. Other considerations with regard to the higher fuel enrichment need also to be taken into account related to safeguards and security, and their interface with safety, in relation to transport of a factory fuelled and refuelled SMR module.

2.2.9. Safe reactor shutdown

Safe reactor shutdown is understood as the condition applicable for all plant states, aiming at ensuring, with margins, that specified fuel design limits are not exceeded. In order to achieve that condition, design means are considered to ensure both the reactivity control of the reactor core and the decay heat removal from the reactor core along with the residual heat from the RCS (see Requirements 46, 51 and 53 of SSR-2/1 (Rev. 1) [11]).

Several SMR designs (in particular WCRs and HTGRs) take advantage of strong negative fuel or moderator temperature feedback coefficients to control reactivity along with the use of control rods (see Ref. [9]).

The fast reactor designs – sodium cooled fast reactors (SFRs) and lead cooled fast reactors (LFRs) – place emphasis on the operational temperature of the coolant well below its boiling point. This feature together with the passive features (natural circulation) supplemented by the engineered systems (control rods) for controlling reactivity, allow for safe reactor shutdown and ensure fuel design limits, see Ref. [16].

In MSR, reactivity control is intended to be ensured by taking advantage of negative temperature feedback of the molten salt (density and Doppler effect) together with control rods, see Ref. [17]. In some MSRs the fuel dissolved in molten salt is circulating and at the same time transferring heat. In such MSRs coupling between neutronic and thermal effects requires adequate considerations to ensure there are no regions in the reactor core where reactivity control could not be achieved.

The means used for safe reactor shutdown to ensure reactivity control for some SMR technologies may result in different design solutions than that of WCRs, some of which may propose safe transient end states that are not shutdown states in the traditional sense as considered for large NPPs. For example, for some transients in some MSR designs, the safest state is at stable, low power levels and not during shutdown (see examples in Ref. [9]).

Systems for ensuring the safe shutdown condition for SMRs can be significantly different from traditional ones, and also use passive systems, as the reactor scram system in large NPPs. For example, according to Ref. [9] for some HTGRs, a large negative Doppler coefficient combined with the helium circulator trip can be used to shut down the reactor while relying on the large heat capacity provided by the graphite structures to passively remove the decay heat without jeopardizing the fuel design margins. Many SMRs rely on passive heat removal systems during shutdown as well as during operation.

Therefore, the reactivity control mechanism considered in the SMR designs might challenge the total validity of safety requirements for the design of NPPs in relation to the control of the reactor core, the reactor shutdown and the definition of fuel design safety limits and margins to achieve them.

2.2.10. Coolant

Many SMRs use coolants other than water. Further considerations are necessary related to the different coolant characteristics, the associated physical and chemical phenomena and different nature of impacts on physical barriers such as chemical, thermal and mechanical damaging mechanisms. Use of a coolant with high boiling point, as for LMFRs, has advantage of low risk of boiling crisis resulting in overheating of fuel elements, see Ref. [16]. For gaseous coolant, as in HTGRs, there is the advantage of no phase change [9]. Some coolants, in particular liquid metals (in LMFRs) or molten salts (in MSRs), have high retention capability for fission products, thus contributing to minimization of the source terms [9]. Use of fast neutron spectra in LMFRs has specific implications on coolant and structural material selection and characteristics, also leading to specific physical phenomena and damaging mechanisms [18]. Furthermore, some coolants (e.g. lead) are toxic and, therefore, in-service inspection, repair and maintenance of components, such as pumps, valves or steam generators or heat exchangers are challenging, see Ref. [19].

Therefore, the specific characteristics of coolants used in SMR designs might challenge the requirements related to the qualification process for items important to safety and the activities to ensure the reliability of items important to safety.

2.2.11. Coupled facilities

Some SMR designs consider coupling with other facilities such as for hydrogen production, heat production or other chemical processes, for example metallurgy or mining of raw material (co-generation). Such combined use of SMRs offers an advantage since, in addition to economic benefits, it adds flexibility for stable utilization of full reactor power.

On the other hand, coupling of an SMR with another facility adds complexity to the design due to potential interaction, both for ensuring smooth operation of the coupled facility as well as due to potential adverse interactions. Possible adverse impacts of the coupled facility on reactor modules need to be considered as external hazards, as potential initiators of additional paths for radioactive releases and a possible source of security issues related to the need for accessing the coupled facility for operating purposes.

2.2.12. Protection against internal and external hazards

As for large NPPs, protection against internal and external hazards is considered in the site evaluation and the design stage. Some SMRs may be located in close proximity of populated or industrial areas. In that perspective, for the protection against external hazards, site selection for SMRs is a challenge since external hazards could have a significant contribution to the overall plant risk. In addition, the design robustness needs to be adequate to ensure the minimization of the impact that external hazards might have on the reliable operation of passive safety systems. Some SMR designs consider underground or semi-buried locations of the nuclear island. This solution increases the level of protection against some external hazards. On the other hand, construction of the plant can become more complicated and vulnerability against other external hazards can be increased, such as flooding, or those induced by the close proximity to hazardous industrial facilities. In any case, the site evaluation needs to identify those specific hazards to be adequately considered as input for the SMR design.

Potential challenges for the protection against internal hazards for SMRs are the modularity feature as well as the use of a single main control room for control and operation of all SMR modules. For the modularity, it is recognized that the close proximity of modules could facilitate the propagation of internal hazards as well as the potential common impact from external hazards. Therefore, specific provisions and measures need to be considered at the design stage to minimize the impact of internal and external hazards on several modules, as common cause failures.

Further, the underground location of the SMR might have positive and negative implications related to security and safeguards that need also to be considered in relation to the design requirements. Examples of previous implications from the security perspective will be the implementation of physical protection and the capability to respond to attacks such as protection against airplane crash and difficulties to neutralize and mitigate insider threats. From the safeguards perspective, an example of implication will be the physical access and the remote communication to the devices for the control of nuclear material accountancy.

2.2.13. Confinement system

The philosophy for the retention of radioactive materials, as in large WCRs, is reinforced in the majority of SMRs aiming at ensuring a high confidence of the confinement function at the level of the first and second barriers [9]. Among that group, some SMR designs use TRISO fuel, and may put less emphasis on the design of engineering features – e.g. civil structures, see examples in Ref. [9], providing confinement such as the traditional leak-tight containment building used for large NPPs. Consequently, from the confinement function point of view, those SMR designs which use TRISO fuel, such as the HTGRs, might not require the same criteria of leaktightness for the structural containment building as it is required for large NPPs with PWRs or PHWRs.

Other SMR technologies, such as MSRs or LMFRs, may operate with the primary coolant close to atmospheric pressure and with significant retention capability of fission products in the coolant (see Refs [9] and [14]). Therefore, the requirements for the containment as a last barrier against releases of radioactive substances might differ from the respective requirements for large NPPs with PWR designs. On the other hand, for SFRs, a possibility of sodium fires can lead to requirements of different nature (in terms of withstanding overpressure and leak-tightness) on the containment building [20].

The containment structure, as a robust barrier to protect against external hazards for all SMR types, will likely remain unchanged. However, the design safety requirements might not be dominated by those requirements associated to the confinement function and therefore the design safety requirements for the containment structure might be different.

2.2.14. Specific design features for ageing management

Design safety requirements related to ageing management of items important to safety in NPPs need to be considered in the design stage of SMRs. The definition of those requirements is based on a comprehensive identification of potential ageing effects with regard to the internal and external environmental factors and degradation mechanisms that might affect items important to safety during their operation. Consequently, the accumulated operational experience plays a crucial role in the identification process. However, the majority of SMR designs have limited or no operational experience [21]. Furthermore, many SMR designs incorporate unique materials, such as LMFRs and MSRs, where limited test data is available and where further investigations are needed to properly evaluate the degradation mechanisms affecting items important to safety, such as the behaviour of halide (fluoride and chloride) salts.

In addition, new concepts of SMR designs proposing sealed cores, such as in some LMFR and HTGR designs, might need additional requirements to ensure adequate accessibility of the equipment during the entire lifetime to perform inspections in support of the ageing management programme.

Finally, regarding the design safety requirements related to qualification, further information is needed for substantiation of adequate margins in component design throughout the life of the facility.

2.2.15. Specific design features affecting radiation protection

Most SMR designs have a reduced fission product inventory and, thus, smaller source terms compared to large NPPs. However, some SMR designs using coolants other than water might face new hazards relevant for the source terms and for the radiation and health protection of workers. Examples of these hazards are graphite dust (especially in pebble bed HTGRs [22], [23]), fission products that penetrate the silicon carbide barrier in HTGRs [24], the mixture of fuel salt and coolant with fission products, or fission products separated in the off-gas tank in MSRs [25] and doses from neutron activation of impurities in lead (e.g. Ag-110m, Sb-124, Co-60) and sodium (Na-24 and Na-22) in LMFRs [26]. Particular attention needs to be paid to the implementation of provisions, in the design stage, for ensuring radiation and health protection of workers during maintenance, testing and in-service inspection activities in compliance with the design safety requirements. In some cases, such as for sodium activation, the increased source terms can be addressed by appropriate shielding of the primary circuit.

In addition to the coolant characteristics mentioned above, SMR designs using integral or compact features, as mentioned in Sections 2.2.2 and 2.2.3, might pose specific challenges for radiation protection during normal operation activities such as testing, in-service inspection or maintenance. Additionally, the impact of unmanned operations on radioactivity control and monitoring needs also to be considered.

3. METHODOLOGY FOR DEVELOPING REGULATORY DESIGN SAFETY REQUIREMENTS

3.1. OVERVIEW OF THE METHODOLOGY

A general approach for developing design safety requirements including adequate DiD for innovative reactors was proposed in IAEA-TECDOC-1570 [13]. The considerations made in that TECDOC were based on the IAEA safety standards valid before 2007, including IAEA Safety Standards Series No. NS-R-1[4]. While potentially applicable, the approach was not specifically intended to support the development of regulatory design requirements and considerations related to the wide range of advanced NPP designs, including the different SMR technologies.

Section 2 of this TECDOC discussed specific issues related to SMRs that may have an impact on the applicability of national regulations. The purpose of the current section is to introduce a general methodology that can be used by regulatory authorities for the development, updating or adapting of regulatory safety requirements for the design of advanced NPPs, which are considered here as a high level requirements as presented in SSR-2/1 (Rev. 1) [11]. Therefore, the methodology is exemplified, taking as starting point the safety requirements for the design of NPPs, which are established in SSR-2/1 (Rev. 1) [11]. Then, the methodology proposes the evaluation of those safety requirements taking into account the specific issues related to SMR designs, as presented in Section 2 with regard to current regulatory practices for the development, updating or adapting the set of national safety requirements in individual Member States for advanced NPP designs.

The implementation of the proposed methodology will thus differ among Member States depending on their regulatory system. The approach described in this section can be used to develop, update or adapt national safety requirements either for a particular single advanced NPP design with a specific technology, or for a range of different advanced NPP designs with different technologies, including the SMRs.

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No. NS-R-1, IAEA, Vienna (2000).

The starting point for the proposed methodology is the evaluation of the regulatory safety requirements for the design of NPPs. For illustration, this TECDOC considers the design safety approach established in SSR-2/1 (Rev. 1) [11], which represents an international consensus of Member States on what constitutes a high level of safety for protecting people and the environment from harmful effects of ionizing radiation. The current IAEA design safety approach has been developed to comply with the fundamental safety objective, established in IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [27]. Among the ten safety principles associated to the fundamental safety objective in SF-1 [27], the implications of the safety approach on the design of the NPP with regard to radiation protection and nuclear safety are mainly captured in five of them:

“Principle 5: Optimization of protection

Protection must be optimized to provide the highest level of safety that can reasonably be achieved.

“Principle 6: Limitation of risks to individuals

Measures for controlling radiation risks must ensure that no individual bears an unacceptable risk of harm.

“Principle 7: Protection of present and future generations

People and the environment, present and future, must be protected against radiation risks.

“Principle 8: Prevention of accidents

All practical efforts must be made to prevent and mitigate nuclear or radiation accidents.

“Principle 9: Emergency preparedness and response

Arrangements must be made for emergency preparedness and response for nuclear or radiation incidents.”

The general intent and purpose of these principles are to prevent, to control and to mitigate potential radioactive releases mainly based on a robust design ensuring the achievement of fundamental safety functions and the application of the DiD concept. Other aspects, from the safety perspective, contributing to the prevention, control and mitigation of potential radioactive releases are the implementation of the safety culture, the performance of a safety assessment as well as of surveillance and inspection activities. Those aspects are further expanded in the requirements presented on the SSR-2/1 (Rev. 1) [11] to derive the safety approach for the design of NPPs.

In addition, the plant design needs to integrate design provisions aiming at facilitating the implementation of emergency preparedness and response measures as considered at the fifth level of DiD, to ensure the readiness and the effectiveness for the mitigation of the radiological consequences of any accidents that might occur.

In particular Section 4 of SSR-2/1 (Rev. 1) [11] provides the basis to achieve the main aspects of the safety approach for NPP design, among them the following:

(a) Ensuring radiation protection of workers and the public be such that radiation doses do not surpass the established dose limits, that radiation doses are maintained as low as reasonably achievable during all operational states and the entire lifetime of the NPP, and that they are kept both below acceptable limits and as low as reasonably achievable in, and after, accident conditions;
(b) Ensuring and maintaining, for all plant states, the fundamental safety functions, which were defined in line with the fundamental safety objective, by identifying and designing the items important to safety required for their fulfilment;
(c) Considering the application of the DiD concept during the design of the NPP as well as striving to ensure, as far as practicable, the independence of safety design provisions considered at its different levels, while:
   (i) Considering available operating experience and knowledge improvements based on research results related to all plant states during the whole lifetime;
   (ii) Considering and complying with relevant national and international codes and standards selected for the design;
   (iii) Considering human capabilities and limitations as well as of factors impacting human performance during all plant states;
   (iv) Designing items important to safety at the NPP to ensure the safety functions performance with a high level of reliability;
   (v) Keeping the NPP operation within its operational limits and conditions for its whole operating life and monitoring its performance, aiming at reducing the likelihood of occurrences of deviations from normal operation;
   (vi) Applying sufficient and adequate safety margins to comply with the conservatively established regulatory acceptance criteria with respect to the onset of damage to the barrier integrity in accident conditions and consequently, with the fundamental safety objective.
(d) Considering for the design of NPPs the results of deterministic safety analysis, supplemented by the insights of probabilistic safety assessment and engineering judgement to ensure first an adequate prevention of accidents and second the mitigation of the accidents' consequences if they were to occur;
(e) Performing periodic and regular reviews of the safety of the NPP to ensure that the design safety level of the plant is maintained throughout the whole lifetime;
(f) Considering challenges in decommissioning and generation of radioactive waste from the design stage of the NPP as well as during its operation to achieve the minimum practicable in terms of both volume and activity of radioactive waste and discharges.

Considering that the safety requirements for the design of NPPs established in SSR-2/1 (Rev. 1) [11] were mainly developed on the basis of the experience with current large NPPs with WCRs, it is expected that SMR design features, discussed in Section 2, might challenge the applicability of existing design safety requirements. Furthermore, design safety requirements for existing WCRs might be incomplete for advanced NPP designs with other technologies than WCR, including SMRs designs, and require adaptation to propose design safety requirements being technology neutral, technology inclusive and technology specific, as proposed in Refs [28] and [29]. Because of the variety of technologies, designs and specific features considered by advanced NPP designs, the application of design safety requirements often needs to be graded to accommodate varied technological and regulatory considerations; for example, based on integrated risk informed, objective oriented, performance based approaches. The use of an integrated risk informed approach aims at ensuring the highest level of safety that can be reasonably achieved in a structured manner by covering all areas impacting the safety of the nuclear installation, such as probabilistic and deterministic factors, human and organizational aspects, and the interface between safety and security, as presented in Fig. 1. Considering the integrated risk informed decision making (IRIDM) approach for evaluating the appropriateness for development, updating or adapting regulatory design safety requirements has the advantage of being both technology neutral and applicable to all regulatory approaches, such as prescriptive and objective oriented approaches.

Further information on the application and factors to be considered in the IRIDM process and examples can be found in IAEA-TECDOC-1909, Considerations on Performing Integrated Risk Informed Decision Making [30].

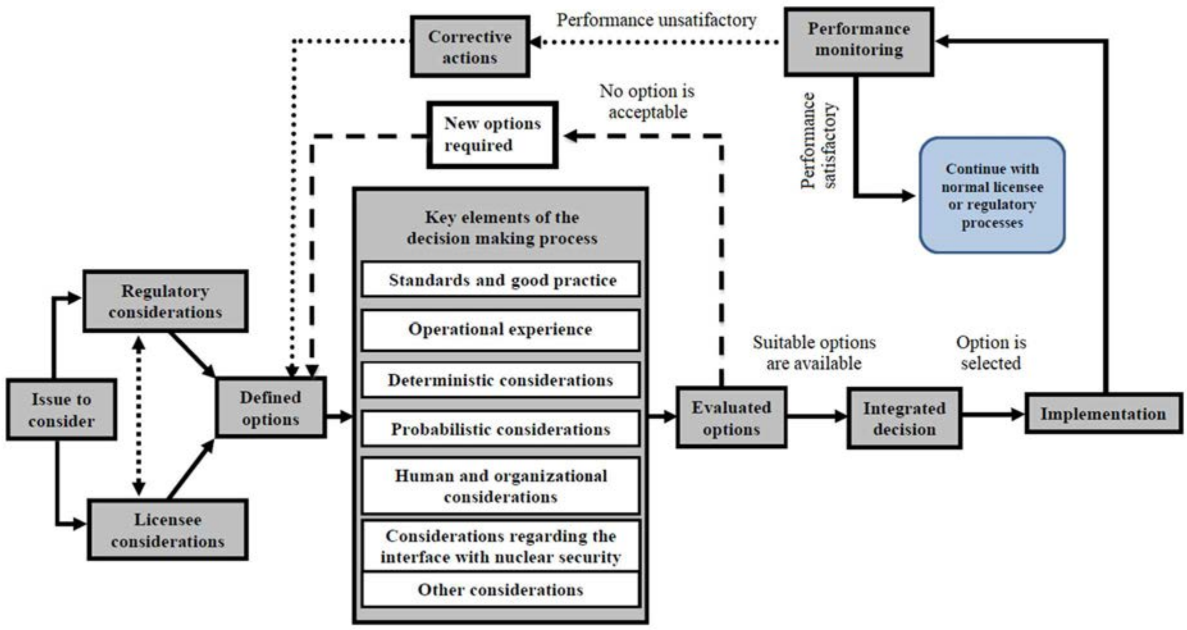

FIG. 1. Main components of the IRIDM process (taken from Ref. [30]).

The IRIDM approach is complemented by the usage of both objective oriented and performance based approaches. The objective oriented approach targets the definition of high level objectives such as design objectives to ensure dose limits and radiation protection for workers and the public, the objectives for safety, security and safeguards (including their interfaces), as well as design and operational economic targets. Defining those objectives allows for keeping the required goals or criteria for performing the evaluation for development, updating or adapting regulatory design safety requirements. In particular the objective oriented approach is intrinsically related to the application of the IRIDM approach at the first step, for consideration by the regulatory body and the licensee.

On the other hand, the performance based approach looks at defining desired and measurable outcomes without prescribing or imposing how they have to be obtained, as described in SECY-98-144 [31] and NUREG-1860 Vol. 1 [32]. Therefore, innovative design solutions or procedures could be considered based on their functional performance while achieving the high level objectives or goals as defined in the objective oriented approach, with or without consideration of risk insights. Figure 2 outlines the process for conducting the performance based regulatory process.

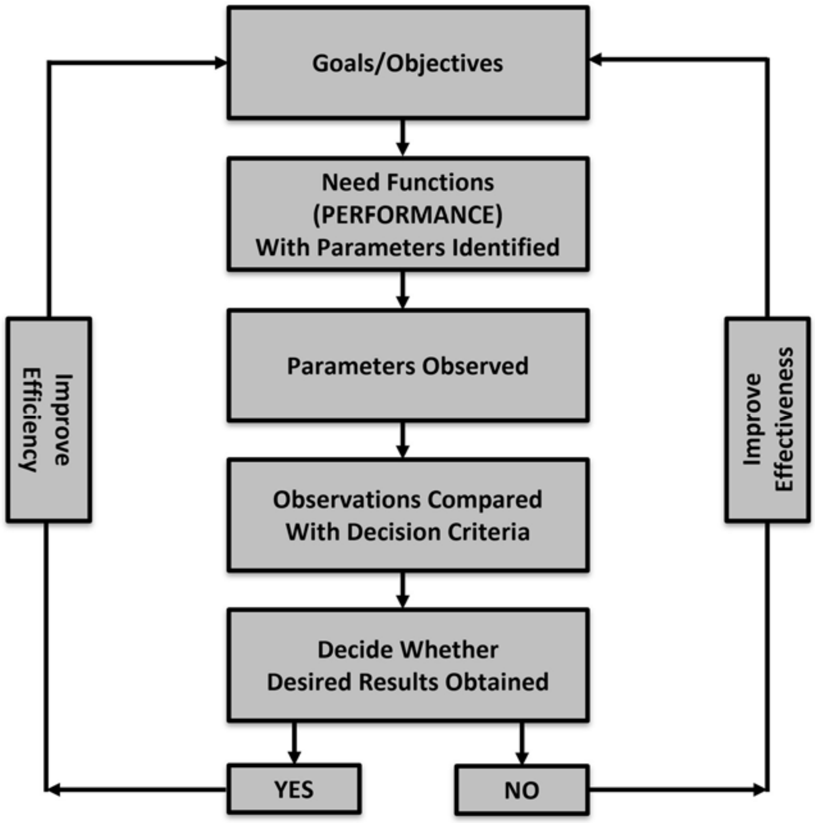

FIG. 2. Process to conduct performance based regulatory approach[5].

Consequently, the use of integrated risk informed, objective oriented, performance based approaches together have the advantage of ensuring a comprehensive and structured approach at developing, adapting or updating regulatory design safety requirements.

The methodology presented in the following Sections 3.2–3.5 introduces the steps needed for the development of the design safety requirements only. The overall process of establishment of a legislative framework and comprehensive regulatory system is not in the scope of this TECDOC. IAEA Safety Standards Series No. SSG-16 (Rev. 1), Establishing the Safety Infrastructure for a Nuclear Power Programme [33], provides an outline of all actions to be taken by States for the development of a national safety infrastructure.

[5] Source: https://www.nrc.gov/about-nrc/regulatory/risk-informed/concept/performance.html

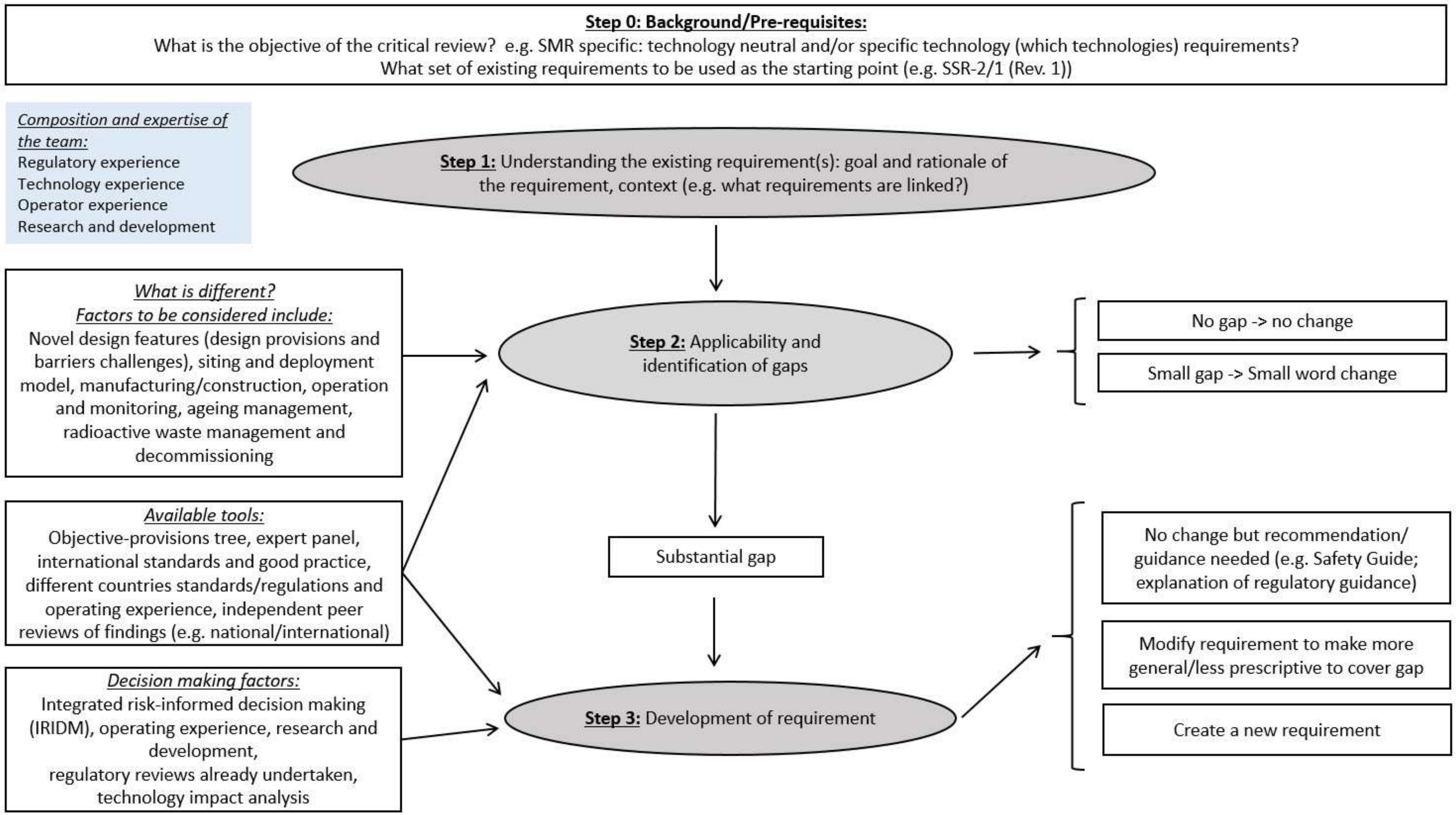

FIG. 3. Illustration of the review process.

The whole process of review of existing design safety requirements and its updating is subdivided into following four steps:

(a) Step 0 – Determine the objective(s) of the review (background and pre-requisites);
(b) Step 1 – Examine existing requirement to understand the rationale and goals;
(c) Step 2 – Evaluate applicability of the requirement and identify gaps;
(d) Step 3 – Determine whether to adapt the requirement or develop a new requirement or guidance.

The review process is illustrated in Fig. 3. The following sections discuss individual steps outlined in Fig. 3.

3.2. STEP 0 – DETERMINE THE OBJECTIVE(S) OF THE REVIEW (BACKGROUND AND PRE-REQUISITES)

For decisions regarding the objectives and scope of the review, factors such as the characteristics of a Member State's regulatory system and their national strategy for new builds, in terms of diversity of advanced NPP designs, are important. In some cases, Member States will seek to develop technology neutral safety requirements which would generally be high level objective oriented requirements. Alternatively, lower level technology inclusive, technology specific and design specific requirements respectively for a subset of reactor technologies or for a particular technology or for a specific design might be also developed. The end state of the final set of design safety requirements, i.e. the product of the review, needs to be clearly established at the onset of the review.

The starting point of the review is also variable and might differ depending on the status of a Member State's regulatory system. It needs to be emphasized that, if the intention is to develop technology neutral regulatory design safety requirements, a comprehensive understanding of the overall regulatory regime, national regulations and international good practice is needed, as well as a general understanding of the reactor technology under review. On the contrary, if the intention is to develop or to update technology specific regulatory design safety requirements, a more detailed understanding of the reactor technology design is highly advisable. No matter the intention, this step defines the starting point of the current regulatory safety approach, such as a prescriptive approach or objective oriented approach, and the definition of the potential outcome as a regulatory safety approach, such as IRIDM.

Examples of experiences in some Member States related to their efforts to adapt or update their regulatory system to be ready for the licensing process of SMRs can be found in Ref. [34].

Taking the structure of SSR-2/1 (Rev. 1) [11] as an example, the four levels of safety requirements for the design of items important to safety for NPPs can be found: technology neutral, technology inclusive, technology specific, and design specific.

3.3. STEP 1 – EXAMINE EXISTING REQUIREMENTS TO UNDERSTAND THE RATIONALE AND GOALS

To evaluate the applicability of a current design safety regulatory requirement, the rationale and goals behind the specific requirement need to first be understood. For example, the requirements are usually linked to higher level requirements such as fundamental safety objectives. Analysis of principal safety requirements from SSR-2/1 (Rev. 1) [11] can help in this regard. An important part of this evaluation is to determine how the requirement contributes to different DiD levels. Some requirements might be linked to only one DiD level, while others might be applicable to more than one or even to all levels. Also, the contribution of the requirement to the performance of safety functions needs to be understood. Similarly to the DiD levels, a requirement can be associated with one, or several safety functions. National regulatory frameworks might also include quantitative safety goals, for example dose limits in normal operation, frequency ranges for different plant states or probabilistic safety metrics. Finally, the contribution of the requirement under evaluation to meeting the safety goals needs also to be considered. Understanding the intent behind the given requirement is an important input for the next steps.

It is likely that for many requirements, the rationales and goals for the requirements will be closely mutually interlinked. The context of the requirement in the overall regulatory system is important; as such, interdependencies with other requirements are likely to be identified. Therefore, while each requirement can be revised individually, the requirements need to be read or revised in a holistic or integrated fashion, and not in isolation.

3.4. STEP 2 – EVALUATE THE APPLICABILITY OF THE REQUIREMENT AND IDENTIFY GAPS

This TECDOC proposes two different approaches – analytical approach and expert panel – that can be used for evaluating the applicability of a given design requirement and identifying differences in relation to reactor technologies. Either of these two approaches have already been used by several States, such as Canada, France, Japan, Russian Federation, the United Kingdom and the USA, in their review of design safety requirements regarding advanced NPP designs. The approaches can be also used in combination depending on the nature of the requirements under evaluation.

A possible analytical approach, which provides a systematic review of implementation of DiD, is the development of the objective–provisions trees, as exemplified in IAEA-TECDOC-1366, Considerations in the Development of Safety Requirements for Innovative Reactors: Application to Modular High Temperature Gas Cooled Reactors [35], for HTGRs. This top down approach using functional diagrams helps to identify the functions required to meet the top-level regulatory objectives for different plant states that may include quantitative safety goals. The other possibility, i.e. the use of an expert panel, also enables the review of a wide range of requirements (see Section 3.4.2). In addition, identified features in Section 2 can be used as guidance on important considerations to help to scope the gap analysis following these two approaches.

Whether analytical approach or an expert panel (or combination) is used, the sub-steps taken are expected to be the same, namely:

(a) Understanding the underlying rationale and goals of the design requirement as outlined in Step 1;
(b) Review and collection of relevant information to establish an adequate knowledge of the selected technology; this sub-step is essential for accomplishing next two sub-steps; important considerations and examples are presented in Section 2;
(c) Identification of gaps; the gaps are understood as those aspects of a requirement that are not applicable or proportional[6] to the technology (set of technologies) considered; and/or specific aspects or areas where existing requirements might not be sufficiently complete

[6] Proportional here is understood for those design requirements where their applicability is challenged by proposed technology improvements (e.g. confinement function).

to cover the technology under consideration. A summary of the important considerations to be accounted for in the gap analysis is presented in the following paragraphs;

(d) Evaluation aimed to determine the importance of any gap assessed with the conclusion whether the gaps require changes of the existing design requirements or development of an explanatory guidance is needed.

The evaluation and understanding of the reactor technology to be introduced and for which the requirements will be assessed, are needed to clarify what is different from NPP reactor technology covered by existing requirements. Even if the existing requirements (usually based on large NPPs with WCRs) seems to be reasonably technology neutral, there may be certain aspects that may need some adjustment to capture the unique characteristics of advanced NPP designs, and particularly for SMRs designs. Some of the differences to be considered in the gap analysis and in updating the requirements are briefly described in the text below as a summary of the examples presented in Section 2.

Given the different reactor technologies proposed for advanced NPP designs, and particularly for SMRs designs currently in different phases of design and construction (see Ref. [9]), the SMR design dependent features need to be fully understood during identification of gaps, such as:

(a) Configuration and robustness of physical barriers against radioactive releases might be different;
(b) Novel and specific technology related design features might be associated with new faults, new physical phenomena and new challenges to ensure the fulfilment of safety functions and integrity of barriers;
(c) The fuel behaviour in all plant states might be different from WCRs;
(d) Design of reactivity control and shutdown means might be different from WCRs, e.g. with more reliance on reactivity feedback mechanisms;
(e) Extended use of inherent safety and passive systems might create different challenges on the engineering SSCs, their performance, reliability and qualification;
(f) There might be different radiation protection challenges, such as in relation to in-service inspection and maintenance activities for integral and compact designs.

3.4.1. Issues of special importance for the identification and analysis of gaps

Based on the discussion in Section 2.2, an overview of issues of special importance to scope the gap analysis and for updating design safety requirements is presented below. The issues are presented in a simple list of items, subdivided into several categories corresponding to logical subdivision of the issues.

(a) Design related requirements

— Practical utilization of graded approach to safety;
— Compliance with the requirement of proven engineering practices;
— Ensuring independence of individual levels of DiD as far as practicable, in particular between levels 3 and 4;
— Design specific physical barriers against releases of radioactive substances;
— Definition and implementation of safety margins, demonstration of adequacy of safety margins;
— Consideration of the confinement properties of inherent fission products (retention capability of fuel and/or coolant);
— Requirements on the containment and its systems;

— Definition of DECs and severe accidents, and implementation of measures to prevent their occurrence or mitigate their consequences;
— Identification and justification of those plant event sequences that need to be practically eliminated since they might potentially lead to early or large radioactive releases;
— Applicability of current design codes and standards for SMRs.

(b) Internal and external hazards

— Robustness against external hazards including adequate margins against external hazards more serious than design basis hazards;
— Consideration of CCFs due to hazards;
— Combined impact of several external hazards events affecting some reactor modules, impacting the confinement function, and the spent fuel pool (if applicable, common for all reactor modules);
— Propagation of an internal hazard from one reactor module to another;
— Accident in one reactor module representing external hazard for other reactor modules;
— Considerations on specific hazards induced by reactor coolants other than water;
— Consideration on the hazards related to, or induced by, coupled facilities;
— Requirements for physical separation between reactor modules due to internal and external hazards.

(c) Passive design features

— Acceptability of passive heat removal in operating plant states and accident conditions;
— Application of single failure criterion for passive systems;
— Reliability of passive systems and its quantification;
— Safety classification of passive SSCs;
— Categorization of safety functions to be performed by passive systems;
— Consideration of active component initiation for operation of passive systems;
— Consideration of uncertainties in natural circulation (cooling) performance in certain conditions;
— Consideration of functional failures other than mechanical failures (e.g. as a result of small driving forces, phenomenological failures, higher level of uncertainties);
— Challenge to testing and qualification of passive systems.

(d) Modularity

— Accumulating the radionuclide inventory in several reactor modules;
— Sharing of safety systems and safety features for DECs between multiple reactor modules;
— Weaknesses due to multiple interfaces between the reactor modules;
— Specifics of security issues for multiple reactor module units;
— Specifics of human factors for multiple reactor modules sharing a single control room;
— Susceptibility to common cause events affecting several reactor modules (e.g. internal fire, explosion, flooding).

(e) Modularization, compact and integral design

- Increasing design margins to compensate for limited operating experience for the first of a kind reactor design;
- Accessibility and radiation protection issues in maintenance of integral and compact designs;
- Issues in periodic testing, inspections and maintenance;
- Potential damage to reactor modules during their transport;
- Issues related to interfaces between safety, security and safeguards.

(f) Accident management and emergency response

- Severity of accidents and emergency response to accidents occurring simultaneously on multiple reactor modules;
- Adequacy of technical and human resources for emergency response in multiple reactor module units;
- Possible disabling of operational staff of several reactor modules due to harsh working conditions in a shared control room;
- Requirements on adequate local infrastructure and external support capability;
- Determination of the size of the emergency planning zone corresponding to radiation hazard (size and probability).

(g) Safety assessment

- Identification of specific initiating events due to use of integral and compact designs, use of passive systems and other novel and specific technology related design features;
- Propagation of initiating events between the reactor modules in a multiple reactor module unit;
- Identification of potential paths of accidental release of radioactive material through containment bypass to a coupled facility;
- Multiphysics coupled analysis of neutronic, thermal-hydraulic, chemical, structural and radiological processes;
- Adequate validation of computer codes for unusual plant configurations (integral designs);
- Adequate experience in use of computer codes for SMRs and qualification of users;
- Modelling of passive system operation with due consideration of specific failure modes;
- Establishment of design specific acceptance criteria;
- Possibility of power oscillations in multichannel systems under natural circulation;
- Probabilistic safety assessment for multiple reactor module units;
- Integration of both deterministic and probabilistic elements in design and safety demonstration;
- Adequate analytical and experimental justification of novel design solutions;
- Independent verification of design and safety assessment by the operating organization.

3.4.2. Identification and analysis of gaps based on an objective–provisions tree

Analytical approaches as the objective–provisions tree can be used to complement the above considerations with a systematic review of the implementation of the DiD for a particular design. The information obtained by means of this approach enables a detailed identification of technical areas where existing design requirements might not be applicable and where there might be a need to create new requirements.

The objective–provisions tree approaches are discussed in greater detail in IAEA-TECDOC-1366 [35] and Safety Reports Series No. 46, Assessment of Defence in Depth for Nuclear Power Plants [36]. The objective–provisions tree approach is briefly explained below. Additional information and examples are provided in the Appendix.

The analysis of a specific requirement is illustrated starting with the consideration of Requirement 5 of SSR-2/1 (Rev. 1) [11], which states:

> **"The design of a nuclear power plant shall be such as to ensure that radiation doses to workers at the plant and to members of the public do not exceed the dose limits, that they are kept as low as reasonably achievable in operational states for the entire lifetime of the plant, and that they remain below acceptable limits and as low as reasonably achievable in, and following, accident conditions."**

In national regulatory frameworks, this requirement might be presented in a different manner, including quantitative safety goals. In national regulatory frameworks, the contribution of the specific requirement under review to the achievement of the fundamental safety objective and Requirement 5, here above, is evaluated. In addition, the contribution of the specific requirement at different levels of DiD needs to be analysed, considering the high level objective can be supported by more specific requirements (usually in form of acceptance criteria) for ensuring integrity of individual barriers. Table 2, reproduced from IAEA-TECDOC-1791, Considerations on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants [37], illustrates the linkage among safety objectives, plant states and acceptance criteria for different DiD levels.

TABLE 2. EXAMPLES OF ACCEPTANCE CRITERIA FOR DIFFERENT PLANT STATES AND DEFENCE IN DEPTH LEVELS (FROM REF. [37])

Level of defence	Objective	Associated plant state	Criteria for maintaining integrity of barriers	Indicative examples of criteria for limitation of radiological consequences
Level 1	Prevention of abnormal operation and failures	Normal operation	No failure of any of the physical barriers except minor operational leakages	Negligible radiological impact beyond immediate vicinity of plant. Acceptable effective dose limits are bounded by general radiation protection limit for the public (1 mSv/year commensurate with natural background), typically in the order of 0.1 mSv/year.
Level 2	Control of abnormal operation and detection of failures	Anticipated operational occurrence	No failure of any of the physical barriers except minor operational leakages	Negligible radiological impact beyond immediate vicinity of the plant. Acceptable effective dose limits are similar as for normal operation, limiting the impact per event and for the period of 1 year following the event (0.1 mSv/year).
Level 3a	Control of DBAs	Design basis accident (DBA)	No consequential damage of the RCS, maintaining containment integrity, limited damage of the fuel	No or only minor radiological impact beyond immediate vicinity of the plant, without the need for any off-site emergency actions. Acceptable effective dose limits are typically few mSv.
Level 3b	Control of DECs without significant fuel degradation	DECs without significant fuel degradation	No consequential damage of the RCS, maintaining containment integrity, limited damage of the fuel	The same or similar radiological acceptance criteria as for the most unlikely DBAs
Level 4	Control of DECs with core melt (mitigation of consequences of severe accidents)	DECs with core melt (severe accident)	Maintaining containment integrity both in an early as well as late phase, and practical elimination of fuel melt when the containment is disabled or by-passed	Radiological acceptance criteria ensuring that only emergency countermeasures that are of limited scope in terms of area and time are necessary
Level 5	Mitigation of radiological consequences of significant releases	Accidents with releases requiring implementation of emergency countermeasures	Containment integrity severely impacted, or containment disabled or bypassed	Off-site radiological impact necessitating emergency countermeasures

Figures 4 and 5 present the process of gap analysis using the objective–provisions tree. The bases for the approach are as follows:

- Safety needs to be ensured by implementing safety provisions considered for each level of DiD at any time;
- Each level of DiD has its own safety objectives aiming at ensuring the integrity of the barriers;
- In order to maintain the barriers' integrity, the fundamental safety functions or more detailed (derived) safety functions need to be performed;
- Safety functions can be challenged by different mechanisms affecting their performance;
- To prevent mechanisms impacting the safety functions, there is a need to implement safety provisions of different kinds;
- Provisions implemented at different levels of DiD need to be reasonably independent.

Each level of DiD has its particular objectives, considering the need for protection of the important barriers and the key features for this protection. To ensure fulfilment of objectives for each level of DiD, all fundamental safety functions or more detailed (derived) safety functions important for this level need to be accomplished.

Challenges are groups of mechanisms, processes or conditions of similar nature that might affect the intended performance of safety functions with similar effects (consequences). Mechanisms are particular processes or situations leading to consequences that might generate challenges to the performance of safety functions.

It is possible to detect, for each mechanism, some safety provisions, such as safety margins of SSCs, inherent plant safety characteristics related to feedback coefficients, or specific features for systems design, which can support the performance of the safety functions and prevent any impact of the mechanism on the related safety functions.

The identification of challenges for the accomplishment of fundamental safety functions as well as of the different provisions considered for preventing this impact for each level of DiD is a critical task for making an inventory of specific capabilities considered in each DiD level of a plant and subsequently the identification of potential gaps. Then, in the identification of the gaps, it has to be verified if the regulatory requirements are adequate to prevent mechanisms challenging integrity of barriers from taking place.

The overall concept of the objective tree is technology neutral, but the development of the tree, from the level of barriers, safety functions and further down to the challenges, mechanisms and provisions, generally is technology and design dependent.

The provisions corresponding to a functional requirement need to be considered in evaluating the safety requirements during the critical review. The comparison between the functional requirements derived by the objective–provisions tree and the requirements in the existing regulatory requirements can determine the gaps. In particular, there may be cases where a provision or the entire line of protection (i.e. set of provisions) is not covered by the reference requirements, or conversely, where provisions have not been identified in the development of the objective–provisions tree and regulatory requirements might suggest extending the list of safety functions. These situations can define areas where the development of safety requirements for key features of the innovative technology might need to be considered.

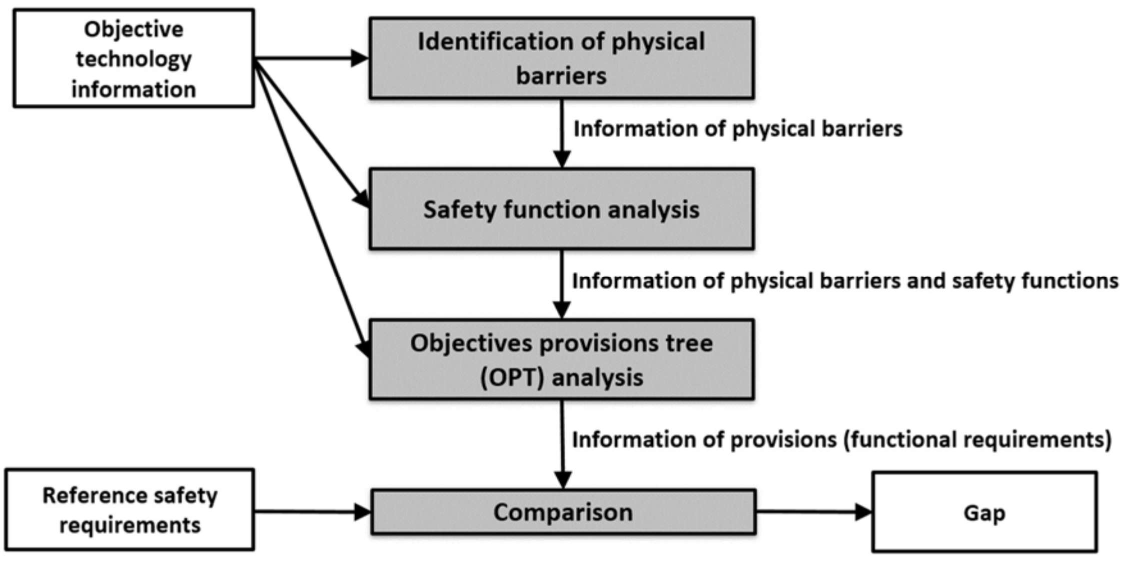

FIG. 4. Process of the gap analysis using the objective–provisions tree.

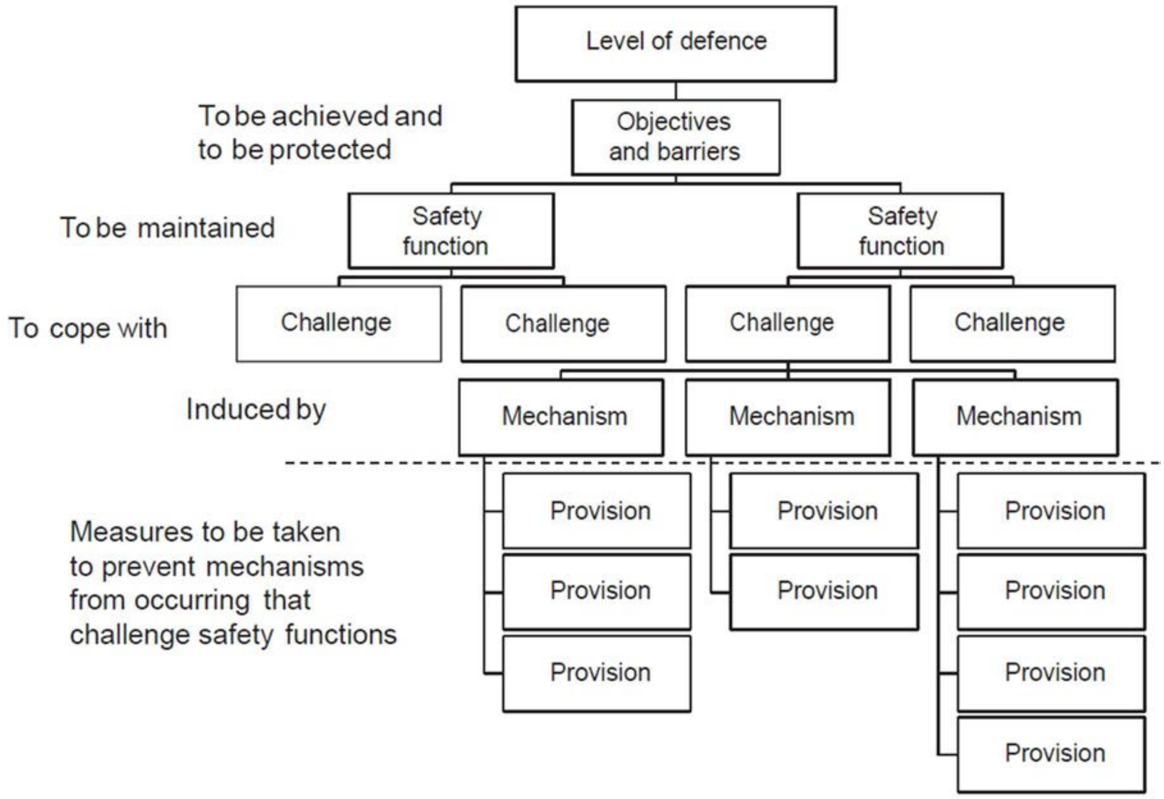

FIG. 5. Objective–provisions tree for the gap analysis.

The application of the approach for the assessment on how a particular novel design complies with the design requirements needs detailed information on both the technology and the design under consideration. Further, it might not be possible for the regulatory body to complete the regulatory review and assessment without close collaboration with other regulatory bodies facing similar challenges. Moreover, using an independent expert review panel as an advisory group to the regulatory body can be considered to ensure the independence of the outcome of the application of these approaches.

3.4.3. Identification and analysis of gaps based on an expert panel

An expert panel is another possibility used for exploring the applicability of a given requirement or a set of requirements to a particular reactor technology or design, such as the Advisory Committees consulted by the French Nuclear Safety Authority (ASN) [38] for defining the technical guidelines [39]. If an expert panel is used, experts involved need to cover a broad range of expertise, including research and development, design, operation and regulatory expertise. Evaluation and identification of gaps might involve inputs from a variety of sources, including safety analysis and gathering relevant international experience in the design and regulation of the given technology or design.

Another example of a commonly used expert panel process is the phenomena identification and ranking table (PIRT) approach. A PIRT process is conducted by eliciting a panel of experts on the key phenomena governing a technology and estimating both the relative importance and relative knowledge of each parameter in order to focus on key areas for understanding and development. Performance of a PIRT process can be used for a variety of developmental tasks, one of which is the code scaling, applicability and uncertainty (CSAU) evaluation methodology. Both the PIRT process and the CSAU methodology are described in Safety

Reports Series No. 52, Best Estimate Safety Analysis for Nuclear Power Plants: Uncertainty Evaluation [40].

The PIRT process does not significantly differ from the potential use of an expert panel in evaluating requirements. An expert panel would be expected to identify, categorize, and determine the relative importance of each part of the existing regulatory framework. The panel will first distil the large existing set of regulations into groups as a pre-requisite for input to the review process outlined in Fig. 3 in order to create a relevant set of regulations influenced by the gaps, categorized according to their relative importance and applicability to a given technology or design. Section 2 and the list of important issues identified in previous paragraphs can help to identify key areas to be considered by the expert panel.

A great deal of international experience already exists among Member States with regard to the variety of reactor technologies that include SMRs and, thus, involvement of international experts with relevant regulatory and operational experience in the expert panel may be helpful in incorporating lessons learned.

3.4.4. Outcomes of Step 2

As illustrated in Fig. 3, the outcome of the gap analysis will be one of the following conclusions:

(a) The requirement is applicable, there are no gaps and no changes to the requirement are needed.
(b) There are minor gaps that challenge the applicability of the requirement, e.g. some components such as fuel assemblies might not be applicable to some advanced nuclear power reactors technologies, and particularly for some SMR designs such as HTGRs. For this case, small (mainly editorial) changes can be made to address the gap.
(c) There are significant gaps that need further evaluation and that could challenge the applicability or proportionality of the requirement and need the development of a new requirement. For this conclusion, Step 3 described in Section 3.5 applies.

3.5. STEP 3 – DETERMINE WHETHER TO ADAPT THE REQUIREMENT OR DEVELOP A NEW REQUIREMENT OR GUIDANCE

As previously discussed, the information gathered in the first two steps of the review is used to determine gaps. If a significant gap is identified, the final step is to develop the new regulatory safety requirement. Both the analytical approach and the expert panel can provide detailed information to understand how a requirement could be modified to address specific gaps or applicability issues. One possibility how existing requirements could be modified is to focus on the compliance with performance criteria. This means to concentrate on the fulfilment of the fundamental safety functions rather than on availability and functioning of the specific system or component. For each identified challenge to a fundamental safety function (to be performed for a certain event or sequence related to the identified gap), a set of acceptance criteria could be established to define acceptable performance. These criteria can then be used as a surrogate to evaluate the efficacy of the existing requirement. If the existing requirement complies with the criteria, it may be considered as technology neutral, otherwise new or modified requirements and/or guidance may be needed, and could be addressed as follows:

(a) Produce recommendations and/or guidance for the application of the existing requirements to address the identified gaps. This option would not involve a modification of the requirements but rather the development of lower level guidance to capture the additional considerations;

(b) Modify the requirement to make it less prescriptive and more general (i.e. technology neutral) to cover the identified gap. An example of this case is presented in Section A.1 of the Appendix;

(c) Create new requirements to address the identified gaps. Examples of these cases are provided in Sections A.2, A.3 and A.4 of the Appendix.

The decisions on modification or development of new requirements also need to take into account the following:

(a) The uncertainties and limited available operational experience related to the areas of change. A conservative approach needs to be adopted unless there is sufficient confidence in the design capability to cope with the list of postulated initiating events and hazards.

(b) The level of regulatory experience with the reactor technologies under evaluation. The lack of regulatory experience could be compensated for by the adoption of an objective oriented approach. Further details on the evaluation of current regulatory experience in regulations and licensing of SMRs are provided in Ref. [34].

(c) Potential technology development. The new or revised requirements need to be sufficiently technology neutral to not preclude their applicability to further evolved advanced NPP designs with different reactor technologies, including SMR designs, that adopt different design solutions.

(d) The risk importance associated with the identified gaps and requirements under evaluation can be also used in the decision making, often referred to as integrated risk informed decision making (IRIDM) and performance based, by helping to confirm that conservative approaches are adopted to minimize risks. Further details on IRIDM can be found in IAEA-TECDOC-1909 [30].

In some cases, there may be existing technology dependent requirements that could be superseded (as not applicable) by the new requirements. Generally, if requirements are considered to be superseded, sufficient operating experience and regulatory experience, applicable to the specific technologies under consideration, need to be available. Operating experience and regulatory experience are needed to demonstrate with a high degree of confidence that the superseded requirement is not applicable to the technologies being considered.

3.6. CONCLUSIONS

Section 3 has presented an updated methodology considering an integrated risk informed, objective oriented, performance based assessment approach that can be used for the development of regulatory safety requirements for the design of advanced NPP designs with different technologies, including SMR designs. It is expected that requirements developed under this methodology will follow the same philosophy as in SSR-2/1 (Rev. 1) [11], where some requirements, such as Requirements 1 to 12, will be high level technology neutral while others will be technology inclusive and more specific, such as Requirements 43 to 82. The approach presented builds on the methodology of IAEA-TECDOC-1570 [13] and reflects recent experience of Member States in reviewing the applicability of their regulatory requirements to advanced NPP designs with different technologies, including SMR designs, which is also summarized in Ref. [34]. The approach has been illustrated, assuming that the applicability of design safety requirements in SSR-2/1 (Rev. 1) [11] to advanced NPP designs with different technologies, including SMR designs, is being evaluated. Section 2 presents examples of specific SMR design features that will assist with the understanding of Step 2 (b), by identifying specific aspects of SMRs that could impact the applicability of the requirements established in SSR-2/1 (Rev. 1) [11]. The Appendix carries forward some of the examples to illustrate the implementation of Step 2 (c) and Step 3 for particular requirements. In particular the cases illustrate where there is a substantial gap and different solutions are adopted, such as modifying a requirement to be more general (see Section A.1 of the Appendix) and creating a new requirement (see Section A.4 of the Appendix).

APPENDIX: REPRESENTATIVE EXAMPLES TO ILLUSTRATE THE METHODOLOGY AND CRITICAL REVIEW PROCESS

This appendix presents four case studies using the review process described in Section 3 for advanced NPP designs with different technologies, including SMR designs. For each case, an SMR-specific safety topic (i.e. containment, multi-module configuration, reactivity control, and air ingress) is presented in greater detail than described in previous sections. The examples use the relevant requirements of SSR-2/1 (Rev. 1) [11] and possibilities for revising them to be more technology neutral. Expert panels in these specific cases are replaced by the collective opinion of the experts involved in drafting this document. Each example introduces and discusses the SSR-2/1 (Rev. 1) [11] design safety requirements relevant to the safety topic and performs the review accordingly.

Regarding Step 0 ‘Background and pre-requisites’ of the review, it needs to be understood that the examples are aimed to assess the applicability of hypothetical regulatory requirements that are aligned with the requirements established in SSR-2/1 (Rev. 1) [11] and to identify the need to develop more technology neutral requirements or additional guidance documents. For simplification, Sections A.1–A.4 in this Annex will directly refer to specific requirements established in SSR-2/1 (Rev. 1) [11]. When found appropriate, the aim can be to create a new requirement that is either technology neutral or adequate for the selected technology.

The examples presented in this Annex demonstrate the ability to apply the review process to different requirements, set of requirements, or safety topics. Examples of modified or new requirements are provided with the understanding that the decision regarding national regulatory requirements is always a national responsibility.

A.1. ROLE OF THE CONTAINMENT IN PERFORMING THE SAFETY FUNCTION OF THE CONFINEMENT

A.1.1. Step 0 – Background and pre-requisites – Determine the objective(s) of the review

With respect to the fundamental safety functions for NPPs, Requirement 4 of SSR-2/1 (Rev. 1) [11] states:

> **“Fulfilment of the following fundamental safety functions for a nuclear power plant shall be ensured for all plant states: (i) control of reactivity, (ii) removal of heat from the reactor and from the fuel store, and (iii) confinement of radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases.”**

Traditionally, the third function has been performed by a series of safety systems and successive barriers including a containment structure. From the brief descriptions of various SMRs, it is seen that the containment structures in different designs are quite different. In the review, it is important to evaluate whether existing national requirements are adequate or need to be modified, and if so, to what level of detail.

A.1.2. Step 1 – Examine existing requirement to understand the rationale and goals

Requirements related to both the containment structure and containment system are established in SSR-2/1 (Rev. 1) [11], specifically in Requirements 54 (containment system for the reactor), 55 (control of radioactive releases from the containment), 56 (isolation of the containment), 57 (access to the containment) and 58 (control of containment conditions). These requirements address the confinement of radioactive substances from the reactor in all plant states within a demonstrably leaktight containment structure, protection of the reactor against effects made by natural external events and human induced events, as well as radiation shielding in all plant states. These overarching requirements contain more detailed sub-requirements based on the assumption that the containment is a leaktight building designed to confine releases (e.g. requirements for airlocks to access the structure and requirements on control of pressure, temperature and combustible gas conditions within the structure).

Many SMR designs aim at using novel design solutions to effectively perform the safety function of confining the radioactive material. The above requirements in SSR-2/1 (Rev. 1) [11] relevant to the containment appear to be not technology independent, however, the underlying purpose of the requirement, which is confinement of radioactive material in all plant states, remains valid.

Requirement 54 (containment system for the reactor) of SSR-2/1 (Rev. 1) [11]) states:

> **"A containment system shall be provided to ensure, or to contribute to, the fulfilment of the following safety functions at the nuclear power plant: (i) confinement of radioactive substances in operational states and in accident conditions; (ii) protection of the reactor against natural external events and human induced events; and (iii) radiation shielding in operational states and in accident conditions."**

Requirement 54 is sufficiently broad and does not exclude various design solutions, assuming that the term 'containment' would not be limited to the traditional containment of large WCRs (i.e. a robust and leaktight structure, capable to withstand dynamic and static loads such as pressure and temperature).

A.1.3. Step 2 – Evaluate applicability of existing requirement and identify gaps

As a general rule, instead of a prescriptive requirement of a traditional containment structure, more holistic performance oriented criteria for each requirement could be established. All plant SSCs have to be assessed against the requirements relevant for the containment, considering effective plant confinement or the approach known as 'functional containment'. This approach is thus becoming more consistent with the approach taken in SSR-2/1 (Rev. 1) [11] for other fundamental safety functions. For instance, Requirements 51 through 53 (removal of residual heat from the reactor core, emergency cooling of the reactor core and heat transfer to an ultimate heat sink (UHS), respectively) of SSR-2/1 (Rev. 1) [11] provide requirements for performing the fundamental safety function of removing heat, but they do not prescribe a method for heat removal.

Design features to ensure confinement of radioactive materials within the plant have to be assessed considering the full spectrum of plant states, which includes operational states and accident conditions, and the corresponding related acceptance criteria. For instance, in the case of large NPPs with WCRs, fuel design limits ensure the primary confinement for anticipated operational occurrences, while together with the design limits of the RCS, the containment structure has a secondary role in mitigating the consequences of these events. However, analyses on DBAs and DECs are required to demonstrate that doses are below the relevant regulatory dose criteria. For large WCRs, this results in the need of a leaktight pressure containment structure. Similar considerations are needed for other requirements.

For some SMR designs, specific design provisions might accomplish the functions set forth without using a traditional leaktight pressure containment structure. One example is TRISO-based fuel designs. TRISO consists of fuel kernels coated with layers of carbonaceous material dispersed within a matrix material. TRISO particles are designed such that each particle has its own effective containment structure and can retain radioactive materials. In effect, the barrier approach described earlier is retained, with a greater emphasis placed on the initial barrier, and then successively allowing less emphasis to be placed on subsequent barriers.

Of course, demonstration of the adequacy of the fuel to retain radioactive materials within the regulatory acceptance criteria will require greater analytical burden on the designer and may involve additional requirements on fuel design and operation (i.e. manufacturing tolerances and temperature limits) to ensure that the design meets the acceptance criteria for all plant states.

For MSRs, or designs utilizing fuel and coolant at low pressure such as LMFRs, a greater reliance could be placed on the reactor coolant boundary and other subsequent confinement structure (i.e. a leaktight guard vessel surrounding the RCS or compartments designed to retain radioactive materials) with relatively small reliance placed on the fuel for accidents of greater severity. Because the existing containment requirements are based on assumption of a pressurized coolant with substantial internal energy, specifics of MSRs or LMFRs are not well addressed by Requirements 55 through 58 of SSR-2/1 (Rev. 1) [11]. In these cases, a leaktight barrier might exist but it could not have the same structure as that used to protect against internal and external hazards. This approach might result in a relatively reduced design demonstration burden in qualifying the fuel but a comparably higher reliance on the performance and integrity of systems used to confine radioactive materials for given plant states.

A.1.4. Step 3 – Determine whether to adapt requirement, or develop new requirement or guidance

The discussion below is provided as a potential approach for re-evaluating the existing containment requirements. As stated previously, the existing requirements of SSR-2/1 (Rev. 1) [11] related to the containment could be understood as not completely technology neutral.

The use of a 'functional containment' approach involves a more holistic consideration of all requirements and their relation to all fundamental safety functions, not just those associated with containment. Specifically, the current requirements associated with the retention of radioactive material for operational states (normal operation and more frequent events, like anticipated operational occurrences) are largely reflected in the acceptance criteria associated with the fuel (e.g. specified acceptable fuel design limits) and RCS boundary. The containment is designed to retain radioactive materials under accident conditions. Under a 'functional containment' approach, each of the barriers against releases of radioactive materials (i.e. fuel, cladding, RCS boundary, and containment) plays a role in the functional containment of radioactive materials.

In that case, a proposed requirement might modify existing regulatory requirements which were oriented to water cooled technologies, as follows: "Each reactor needs to have a functional containment, consisting of multiple independent barriers. The functional containment needs to be able to confine the release of radioactive substances in all operational states and in accident conditions."

Using TRISO fuel as an example, the functional containment approach can focus on the performance of the fuel to retain radioactive materials. As such, the acceptance criteria associated with the fuel particle integrity (coupled with any other barriers) would have to account for a wider spectrum of plant states than fuel acceptance criteria for large WCRs. In theory, this could extend acceptance criteria to maintain fuel integrity for all plant states within the design envelope (i.e. including DECs), depending on the extent to which the fuel is credited to prevent releases of radioactive materials.

As described earlier, the use of a functional containment approach would not allow SMR designs to obviate Requirement 54 (Containment system for the reactor) in SSR-2/1 (Rev. 1) [11] aligned with existing national regulations. Protection of SSCs (items important to safety) potentially located inside the containment from internal and external events and radiation protection would still be required, but could be accomplished with the use of other, independent SSCs designed specifically for those purposes. As an example, characteristics related to design-basis flooding, wind or seismic loadings might be required of these structures and would thereby be incorporated into this functional containment approach, but not necessarily as part of the same system as designed to meet the proposed functional containment requirement. The total effective plant confinement, then, would be accomplished by a set of SSCs, with each system acting to perform its radioactive material confinement function for a specific set of plant states. For instance, a combination of the fuel and RCS might be relied upon to mitigate against internally initiated radiological hazards, with the reactor building or similar structure being relied upon to mitigate against external events. This approach has the advantage of being less prescriptive and more technology neutral, incentivizing designers to make safety improvements for specific components tailored to reduce the impact of individual hazards, rather than potentially relying on a single leaktight structure to mitigate against a full spectrum of various events.

A.2. CONTROL OF REACTIVITY

A.2.1. Step 0 – Background and pre-requisites – Determine the objective(s) of the review

SMR designs differ from traditional NPP using WCR technologies, different configurations, different operating conditions and materials. NPPs with non-WCRs use coolants and fuels that differ from WCRs in characteristics and mechanisms affecting the safety functions (see 2.2.10). The means used to control reactivity for these different technologies may result in different design solutions (see 2.2.9), some of which may propose safe transient end states that are not a ‘shutdown state’ in the traditional sense as currently described in Requirement 46 of SSR-2/1 (Rev. 1) [11]. Addressing this matter consistently across the variety of advanced NPP designs with different technologies will be important to establish appropriate requirements for reactivity control across the full spectrum of SMR designs, including transportable technology types.

A.2.2. Step 1 – Examine existing requirement to understand the rationale and goals

Requirements 45 and 46 of SSR-2/1 (Rev. 1) [11] are associated with reactivity control of the reactor core. In addition, reactivity control of fuel handling and storage systems is governed by Requirement 80 of SSR-2/1 (Rev. 1) [11]. This requirement is discussed below only partially due to the large differences between fuel storage systems across the spectrum of technologies covered by advanced NPP designs. The aforementioned requirements relate to the fulfilment of the first fundamental safety function.

Requirement 45, which relates to the control of reactivity in the reactor core, appears to be largely technology neutral, with the exception of the term ‘reactor core’, which is difficult to clearly define for MSRs. Paragraphs 6.4–6.6 subordinated to Requirement 45 contain terminology such as “pressure boundary of the reactor coolant systems”, which is not technology neutral as some designs have an RCS boundary that may either not be a pressure boundary or is independent of the credited method for transferring residual heat in accident conditions.

Requirement 46, which relates to reactor shutdown, is technology neutral, but how a 'means' of shutdown is defined leaves room for different interpretations. Many non-water cooled advanced NPP designs, including SMR designs (e.g. fast reactor designs) place more emphasis on inherent reactivity feedback mechanisms to supplement engineered systems for ensuring reactor shutdown (see Section 2.2.9). Due to design and size considerations, these features may be sufficient to control reactor power in a fashion that differs from traditional solutions (i.e. two engineered systems) dominantly by means of feedback due to geometric effects supplementing or providing entirely for one of the two required means of shutdown. Diverse, independent means for reactor shutdown need still to be provided to comply with related design requirements (para. 6.9 of SSR-2/1 (Rev. 1) [11]), while the question how a 'means' for reactor shutdown to ensure reactivity control is defined might require additional considerations for some advanced NPP designs with different technologies.

Requirement 80 of SSR-2/1 (Rev. 1) [11] relates to the fuel handling and storage systems. Again, at a general level, the requirement is sufficiently broad to be technology neutral. However, various elements of the associated paragraphs, such as to "provide for the identification of individual fuel assemblies" are not technology neutral, e.g. for a large pebble bed HTGR or an MSR. Similarly, application of the requirements in paras 6.68 and 6.68A, referring to WCRs, are indeed not technology neutral, but technology inclusive particularly for WCRs. Selection of a technology for a temporary spent fuel storage facility may also involve necessary changes to Requirement 80; some SMR designs utilize long lived cores with no refuelling, or have no refuelling outages and instead utilize online refuelling.

A.2.3. Step 2 – Evaluate applicability of existing requirement and identify gaps

The variety of technologies considered by advanced NPP designs, including SMRs, may necessitate less detailed requirements than those prescribed for WCR designs. Reactivity control for non-WCR designs with stronger negative reactivity feedback mechanisms (e.g. thermal expansion for some fast reactor designs) or those less prone to large reactivity excursions (e.g. some HTGR designs) might involve consideration of what constitutes a 'means' for reactor shutdown to ensure reactivity control. The review process of a specific technology could result in the conclusion that the relevant requirements of SSR-2/1 (Rev. 1) [11] could be not sufficiently technology neutral for a given technology considered by advanced NPP designs.

For some transients, it is possible that safe states at certain stages of the transient are not shutdown states, but instead they are stable, low power levels so that local heat transfer conditions are safe throughout the reactor. Additionally, defining a single reactivity state across the core (determining whether it is 'shutdown') can be difficult for some scenarios, and it is important to consider both local and system reactivity effects when evaluating reactivity control requirements. Consequently, the paragraphs associated with Requirements 45 and 46 of SSR-2/1 (Rev. 1) [11] might need to be augmented to consider design dependent aspects in order to ensure that the fundamental safety function (i.e. to control reactor power) is met.

MSR designs necessitate a separate thorough evaluation of the applicability, and identification of potential gaps, of reactivity control requirements. Due to the complex fuel-core geometric configuration (i.e. flow of fuel in a loop) and possibility of continuous online refuelling, reactivity control systems need to account for phenomena beyond traditional core neutronic behaviour and control rods insertion (i.e. delayed neutron movement beyond the ‘core’ region or their continuous insertion and depletion). Because the fuel itself is flowing and directly transferring heat, thermal and geometric feedback mechanisms present important considerations for controlling reactivity. These mechanisms may be the primary reactivity feedback mechanisms for some events, rather than engineered control features (i.e. rods and neutron poisons).

Additionally, when considering the case of a transportable, in particular a floating NPP, a wider variety of external events apply, and additional requirements may be warranted due to the nature of the transportation of a reactor. For WCRs, the primary means to control the neutron flux of the core is via the control rods, which have limited positional freedom (generally up and down). For floating NPPs, careful consideration of what conditions are applicable for each plant state needs to be made because the vessel is transported, located or sited on the surface of a body of water. This adds the possibility of inadvertent rocking, tipping, or even capsizing the vessel equipped with the floating NPP that needs to be considered as part of normal operation, abnormal operation and DBAs, and potential changes in the direction and force of the reactor’s relative gravity.

A.2.4. Step 3 – Determine whether to adapt requirement, or develop new requirement or guidance

Several potential gaps are highlighted in Step 2. This example will focus on the specific case of floating NPPs. Based on the discussion in Step 2, the following proposals include additional considerations related to the control of reactivity fundamental safety function for floating NPPs:

(a) Creating a new paragraph under Requirement 45 of SSR-2/1 (Rev. 1) [11] which could resemble the following: “The core, reactor elements, and control element actuators need to be designed in such a way that the inability to move in any direction from the electric motor and manual drive, ejection of control elements or spontaneous uncoupling of control elements with actuators in any position of the vessel is precluded, including when it capsized.”
(b) Requirement 46 of SSR-2/1 (Rev. 1) [11] could remain as it is or be further adapted. It is important to consider careening and trim differences of the vessel including capsize, as well as potentially imposing additional requirements related to the transportation mode. Additional requirements could stipulate that when the vessel is capsized, the absorber rods of the shutdown system have to be entered to the core, such that the core is subcritical for all postulated accident conditions and core configurations. Further, prior to transportation, the reactor has to be shut down and brought to a subcritical state. During transportation, supplementary shutdown systems need to be available to ensure the reactor remains subcritical during any accident condition.
(c) If the floating NPP is to have a fuel handling and storage system that includes storage devices for fuel assemblies, the provisions of Requirement 80 of SSR-2/1 (Rev. 1) [11] need to be followed for all operational conditions, including the full spectrum of external events (e.g. when the vessel is capsized and flooded). Additional requirements on shielding and geometry control of spent fuel storage need also be imposed.

Additional requirements for floating NPPs could be further developed, e.g. on conditions during transport if the vessel is not capable of self-propulsion. Additional measures to ensure subcritical reactor cores and necessary heat removal from the core might be appropriate in the interest of DiD.

A.3. MODULARITY FEATURE

A.3.1. Step 0 – Background and pre-requisites – Determine the objective(s) of the review

See Section 2.2.4 for the definitions related to modularity.

Existing regulations aligned with SSR-2/1 (Rev. 1) [11] may consider multiple facilities or multiple unit sites in the traditional sense of several reactors with their own reactor building and safety systems located on a same site. This example considers the characteristics and challenges of the modularity feature of SMRs and how this may impact existing regulatory requirements.

A.3.2. Step 1 – Examine existing requirement to understand the rationale and goals

The IAEA defines a NPP as (Glossary of Terms in Power Reactor Information System (PRIS) Reports [41]):

> "a thermal power station in which the heat source is one or more nuclear reactors. As in a conventional thermal power station the heat is used to generate the steam which drives a steam turbine connected to a generator which produces electricity."

The nuclear steam supply system consists of the reactor, cooling system and associated piping in the NPP that are used to generate the steam. SSR-2/1 (Rev. 1) [11] establishes requirements for the design of NPPs and generally refers to the 'plant' as the facility to which the requirements apply. Several requirements relate to units of a multiple unit NPP. In such cases, it can be inferred that 'unit' refers to the reactor and other parts of the nuclear steam supply system, which includes the primary and secondary circuits (in case of a traditional PWR). Similarly, caveated by design particulars, a reactor module of SMR could be seen as a nuclear steam supply system 'unit' for a multi-module SMR.

Specific requirements in SSR-2/1 (Rev. 1) [11] for multiple unit plants (on the same site) are discussed below.

Requirement 17 (Internal and external hazards) of SSR-2/1 (Rev. 1) [11] contains para. 5.15B which states:

> "For multiple unit plant sites, the design shall take due account of the potential for specific hazards to give rise to impacts on several or even all units on the site simultaneously."

This paragraph was added to the latest revision of SSR-2/1 (Rev. 1) [11], considering the main lessons and observations from the nuclear accident at the Fukushima Daiichi NPP. The requirement enforces the need to design for prevention and mitigation of accidents from a whole site perspective.

This means that the design of a multi-unit plant needs to consider any mutual impacts among units. This is reinforced by Requirement 10 (Assessment of engineering aspects) of IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [42], which includes requirements for safety assessment of multiple facilities or activities located at an individual site and cases where resources, such as cooling sources or electric power supply or field operators, for a facility are shared among other facilities at the site.

SSR-2/1 (Rev. 1) [11] states:

> **"Requirement 33: Safety systems, and safety features for design extension conditions, of units of a multiple unit nuclear power plant**
>
> **"Each unit of a multiple unit nuclear power plant shall have its own safety systems and shall have its own safety features for design extension conditions."**

This requirement of SSR-2/1 (Rev. 1) [11] also contains para. 5.63 which states:

> "To further enhance safety, means allowing interconnections between units of a multiple unit nuclear power plant shall be considered in the design."

A.3.3. Step 2 – Evaluate applicability of existing requirement and identify gaps

The modularity feature of SMRs imposes specific challenges. For example, some SMRs propose a single control room and a common heat sink for all reactor modules housed at a single nuclear island building. Such options are not commonly implemented in traditional NPPs, and the sharing of systems introduces the potential for additional CCFs, interactions between reactor modules and specific considerations regarding internal and external hazards.

As introduced in Step 1, Requirement 17 of SSR-2/1 (Rev. 1) [11] directly applies to the modularity feature of SMRs; however, consideration needs to be given to the interactions between the SMR modules, similarly as currently between the units. This either represents a non-negligible gap, or it could potentially be treated as a minor gap upon further evaluation.

Regarding Requirement 33 of SSR-2/1 (Rev. 1) [11] for the modularity feature of SMRs, it can be inferred that any reactor module, on its own, needs to have capability for safe reactor shutdown, residual heat removal, and prevention or mitigation of the consequences for DBAs and DECs. Further on, the possibility of mutual support from adjacent facilities and available systems needs to be considered. Thus, where possible, the design provisions might consider that in accident conditions one SMR module can support another without jeopardizing its own safety. The following discussion further elaborates on some of the challenges associated with the modularity feature of SMRs and identifies possible gaps that might not be captured by existing regulations.

A.3.3.1. Postulated initiating events

For the modularity feature of SMRs, there is the possibility that an initiating event in one SMR module simultaneously prompts initiating events in another SMR module. Such a possibility needs to be closely considered in the design phase. Examples could include the effects from loss of power supply, loss of UHS, or the failure of shared equipment. While this does not necessarily represent a gap in regulations, careful consideration is needed.

A.3.3.2. Safety assessment

The safety assessment has to account for the challenges related to the modularity feature of SMRs. The safety assessment has to verify that all common features and dependencies do not lead to unacceptable consequences. Specific probabilistic safety assessment guidance might need to be developed to adequately consider the coincident occurrence of accident sequences which could lead to radioactive releases involving several modules. In particular, it would be useful to provide guidance related to common cause events that affect more than one SMR module simultaneously.

A.3.3.3. Internal hazards

The modularity feature of SMRs could increase the vulnerability to internal common cause events such as fire, explosion, pipe whip, release of pressurized fluid or missile generation. Therefore, special consideration needs to be taken regarding the protection against effects of internal hazards with regard to the modularity feature of SMRs.

A.3.3.4. External hazards

For the modularity feature of SMRs, the design needs to consider the potential for external hazards impacting several or all SMR modules simultaneously since that could intensify challenges that operating personnel might face during an accident. Therefore, the consequences of an external hazard impacting one SMR module might affect a neighbouring module; and could adversely impact the available resources to be shared at the unit such as human, equipment and consumable resources. These potential challenges need to be identified, allowing to anticipate sufficient resources and to define the mitigation strategies. These potential issues might need consideration in existing regulations.

A.3.3.5. Shared systems or interconnections between several modules

Some SMR designs implementing the modularity feature propose to have some shared systems and interconnections. Consequently, the safety case needs to demonstrate that any connections, shared resources or dependencies among SMR modules do not jeopardize safety. The impact of having shared resources and dependencies among SMR modules needs to be evaluated with regard to the DiD levels and to their independence.

A.3.3.6. Independence of defence in depth levels

Together with the modularity feature, other specific features of SMRs such as compact design, close proximity, sharing of equipment or potential common failures might particularly challenge the independence of DiD levels. For example, the event or accident progression in one module could also impact adjacent SMR modules leading to the risk of affecting both their safety systems and safety features. The independence of safety features considered among DiD levels, as far as practicable, needs to be considered as an important requirement in assuring the effectiveness of DiD and this could represent a gap needing further consideration.

A.3.3.7. Shared safety features

When establishing the acceptance criteria for a shared confinement function, the designer needs to account for simultaneous challenges to confinement from all SMR modules housed in the same building. Similarly, for common heat sinks, requirements may be needed related to their design. For example, the cooling capacity of a common spent fuel pool or UHS needs to be considered. These could represent gaps in the current requirements.

In the development of new requirements, the additional sources of CCFs and potential interactions between SMR modules need to be considered. An example of a CCF that can affect several SMR modules is the impact that oxide deposits and coolant solidification can have on the safety of sodium cooled fast reactors (SFRs) and lead cooled fast reactors (LFRs).

A.3.3.8. Common control room for several reactor modules

Some SMR designs propose to control several SMR modules from a single control room. Although there is some relevant experience with large NPPs, this might not be directly applicable, since SMRs are often proposed to have smaller crews and/or operate in remote locations. Gaps might exist in existing regulations, and new requirements and/or guidance might be needed, given the unique human factors posed by a single control room design.

A.3.4. Step 3 – Determine whether to adapt requirement, or develop new requirement or guidance

In order to ensure that all hazards are identified and to ensure safety through DiD, regulatory requirements need to be considered holistically, to determine whether modification or addition of new (sub)requirement(s) to address the gaps posed by the modularity feature of SMRs and the potential sharing of systems is needed. For an example of how these requirements could be modified, see IAEA-TECDOC-1936 [12].

A.4. AIR INGRESS IN A HIGH TEMPERATURE GAS COOLED REACTOR

A.4.1. Step 0 – Background and pre-requisites – Determine the objective(s) of the review

For HTGRs, graphite is used for the core internals as a moderator, reflector and in some designs as core structural support. Air ingress into the reactor core in the case of the rupture of the reactor pressure boundary may cause oxidation of core graphite. This is a specific challenge potentially causing initiating events that needs to be considered in the design of HTGRs, in accordance with Requirement 16 (Postulated initiating events) of SSR-2/1 (Rev. 1) [11]. Given that there is no requirement that specifically addresses air ingress, it might be necessary to establish new safety requirements related to air ingress for HTGRs.

A.4.2. Step 1 – Examine existing requirement to understand the rationale and goals

Because air ingress is a HTGR specific challenge, it might not be explicitly considered in the existing requirements of SSR-2/1 (Rev. 1) [11]. However, provisions are needed to prevent or mitigate air ingress to ensure performance of the fundamental safety functions and integrity of physical barriers. There are several potential approaches to address this gap. The discussion below will highlight key technical considerations that would be important in updating regulatory requirements instead on focusing on how these could be addressed in SSR-2/1 (Rev. 1) [11].

A.4.3. Step 2 – Evaluate applicability of existing requirement and identify gaps

Gaps between the existing regulatory requirements and the needs for HTGRs can be identified and evaluated using the objective–provisions tree approach, as described in Section 3.4.2.

A.4.3.1. Identification of physical barriers

In the case of HTGRs, the physical barriers contributing to the confinement function are the coated fuel particle (CFP), the fuel element, the reactor coolant pressure boundary, and the reactor building.

A.4.3.2. Safety function analysis

Safety functions of the CFP that need to be performed to ensure the integrity and confinement function of the barrier are evaluated using the objective–provisions tree (see Fig. 6). It is a phenomena based analysis, and initiating events such as heat up and oxidation are postulated to identify safety functions for a wide range of designs. The safety functions to ensure the integrity of physical barriers are summarized in Table 3. The safety function to maintain core geometry is not applicable for the reactor building in this analysis because this safety function does not affect the integrity and confinement function of the reactor building. However, functional requirements to maintain the core geometry by the reactor building will be identified from the following analysis by the objective–provisions tree related to the safety function of maintaining core geometry to ensure the integrity of the CFP, as shown in the example presented in Fig. 6.

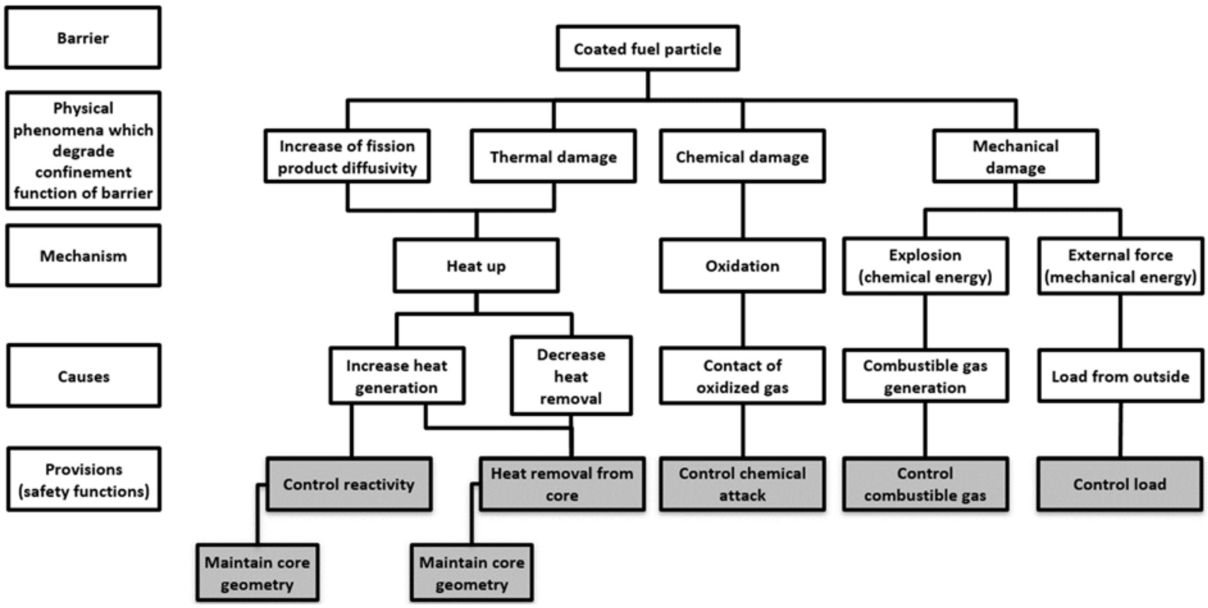

FIG. 6. Results of the safety function analysis to ensure the integrity of the coating layer of the coated fuel particle.

A.4.3.3. Objective–provisions tree analysis

An objective–provisions tree analysis needs to be done for all combination of SSCs for a given safety function that needs to be performed, in this case, to ensure the integrity and confinement function involving a physical barrier. The objective–provisions tree is developed for all combination of physical barriers and safety functions as listed in Table 3. However, only the portions of the objective–provisions tree related to the air ingress for each physical barrier are extracted and summarized in Fig. 7 as example of this case study. Even with the same event of air ingress, there are three sets of provisions due to different types of challenges. Many provisions are common for all challenges, such as prevention of air ingress, reactor shutdown, mitigation of core temperature increase, heat removal from core, heat transfer to an UHS, and mitigation of air ingress. However, there are specific provisions for each challenge.

TABLE 3. RELATIONSHIPS BETWEEN PHYSICAL BARRIERS AND SAFETY FUNCTIONS TO PROTECT THE INTEGRITY OF PHYSICAL BARRIERS

SAFETY FUNCTION	PHYSICAL BARRIER			
	COATED FUEL PARTICLE	FUEL ELEMENT	REACTOR COOLANT PRESSURE BOUNDARY	REACTOR BUILDING
Control reactivity	X	–	X	–
Heat removal from core	X	–	X	–
Control chemical attack	X	X	–	–
Control combustible gas	X	X	X	X
Control load	X	X	X	X
Maintain core geometry	X	–	X	–
Heat removal from reactor coolant pressure boundary	–	–	X	–

The detailed provisions included in the objective tree and corresponding functional requirements are summarized in Table 4. Each provision can be also categorized into the applicable level of DiD. The objective–provisions tree analysis demonstrates that all levels of DiD, i.e. from Level 1 to Level 4, are necessary to prevent or mitigate air ingress. The specific level of DiD is an example as it depends on reactor design and regulatory requirements in each Member State. For example, air ingress might be categorized as a DBA in some States and as a DEC in other States. Some provisions are uniquely specific to air ingress in HTGRs, such as mitigation of air ingress. On the other hand, some provisions such as monitoring, reactor shutdown and heat removal from the core will appear as common provisions in the objective–provisions tree for other challenges.

In Table 4, the left set (column) of provisions is developed to cope with the challenge of oxidation of the CFP by air ingress. Provisions specific to this challenge are ‘prevention of contact of oxidized gas to the CFP’ and ‘mitigation of CFP oxidation’. The middle column of provisions is developed to cope with the oxidation of core graphite by air ingress from the mechanisms of the heat generation by chemical reaction, oxidation of fuel element and damage of graphite core support structure. A specific provision to prevent these mechanisms is ‘mitigation of graphite oxidation by high quality graphite’. The right column of provisions is developed to cope with generation of combustible gas (i.e. carbon monoxide) due to graphite oxidation by air ingress. Provisions specific to these conditions are ‘mitigation of graphite oxidation by high quality graphite’ and ‘control of combustible gas concentration by the reactor building’.

TABLE 4. EXAMPLE OF PROVISIONS AND CORRESPONDING FUNCTIONAL REQUIREMENTS FOR AIR INGRESS

PROVISION	FUNCTIONAL REQUIREMENT(S)	DEFENCE IN DEPTH LEVEL
Prevention of air ingress by high quality of reactor coolant pressure boundary, monitoring, operational limit	To prevent air ingress (by high quality of the reactor coolant pressure boundary) To maintain operational limits and conditions To monitor plant parameter in normal operation To control the plant To protect deviations from normal operational states	1
Reactor shutdown by the reactor shutdown system (i.e. rods)	To shutdown reactor (by the reactor shutdown system)	2–3
Reactor shutdown by inherent reactivity feedback	To shutdown reactor (by inherent reactivity feedback)	4
Mitigation of core temperature increase by large core heat capacity and relatively low power density	To mitigate core heat up (by large core heat capacity and relatively low power density)	2–4
Heat removal from core to the reactor pressure vessel by conduction, convection and radiation	To remove heat from core to the reactor pressure vessel (by conduction, convection and radiation)	2–4
Heat transfer from the reactor pressure vessel to an ultimate heat sink by the reactor cavity cooling system	To transfer heat (from the reactor pressure vessel) to an ultimate heat sink (by the reactor cavity cooling system)	2–3
Heat transfer from the reactor pressure vessel to a different ultimate heat sink	To transfer heat (from the reactor pressure vessel) to a different ultimate heat sink	4
Mitigation of air ingress by the reactor building	To mitigate air ingress (by the reactor building)	3–4
Mitigation of graphite oxidation by employing of high quality (i.e. low oxidation rate) graphite	To mitigate oxidation of graphite (by employing of high quality graphite)	3–4
Prevention of contact of oxidized gas to the CFP by the fuel element	To prevent contact of oxidation gases to the CFP (by the fuel element)	3–4
Mitigation of CFP oxidation by the fuel element and material characteristics of the CFP itself	To mitigate oxidation of the CFP (by the fuel element and material characteristics of the CFP itself)	3–4
Control of combustible gas concentration by the reactor building	To control combustible gas concentration (by the reactor building)	3–4

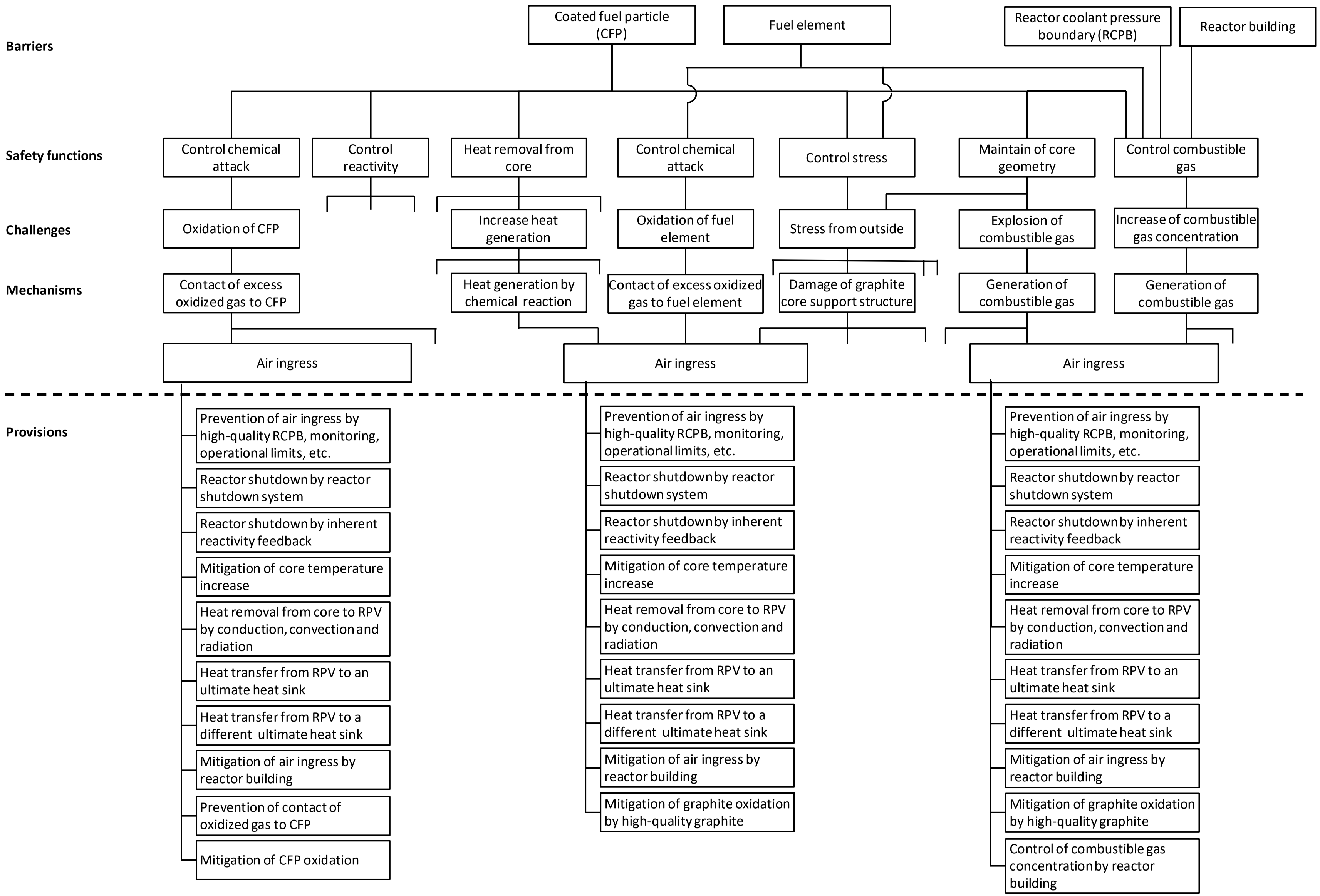

FIG. 7. Result of the objective–provisions tree analysis related to air ingress (RPV: reactor pressure vessel).

A.4.3.4. Gap analysis

The functional requirements identified by the objective–provisions tree need to be compared to existing regulatory design requirements to identify potential gaps. The provisions in Table 4 include possible means to cope with the challenges. The means are shown in the parentheses of the functional requirements column, in Table 4, as examples.

The following areas may therefore be considered as potential gaps:

(a) For the functional requirement "to prevent air ingress" by the reactor coolant pressure boundary, a HTGR specific regulatory requirement might be necessary because air ingress is a HTGR specific hazard.
(b) The functional requirement "to mitigate core heat up" by large core heat capacity and relatively low power density is a HTGR specific case. For the functional requirement "to remove heat from the core" and "to transfer heat to an UHS", if it is defined without detailed "means," they could be evaluated as technology neutral. However, if these requirements are interpreted taking into account more detailed specification of means for heat transfer, i.e. "to remove heat from the core to the reactor pressure vessel by conduction, convection and radiation", "to transfer heat from the reactor pressure vessel to an UHS" and "to transfer heat from the reactor pressure vessel to a different UHS", gaps exist.
(c) The remaining functional requirements, i.e. "to mitigate air ingress", "to mitigate oxidation of graphite", "to prevent contact of oxidation gases to coated fuel particle", "to mitigate oxidation of coated fuel particle" and "to control combustible gas concentration", are HTGR specific and recognized as potential substantial gaps in existing regulations.

A.4.4. Step 3 – Determine whether to adapt requirement, or develop new requirement or guidance

The substantial gaps identified in the gap analysis are to be considered in the development of regulatory safety requirements for design, as discussed further below.

The functional requirement "to prevent air ingress" by the reactor coolant pressure boundary is related to the RCS. Current regulations include requirements for the high quality design and construction of the RCS in line with Requirement 47 of SSR-2/1 (Rev. 1) [11]. However, preventing air ingress into the reactor core might not be considered as a subject of existing regulatory requirements for large NPPs with WCRs. The isolation function of the reactor coolant pressure boundary as it relates to HTGRs is less focused on preventing the loss of coolant as in the case of WCRs, but rather on preventing the air ingress into the reactor once a depressurization accident has occurred. Consequently, it is proposed to develop requirements or guidance that address the identified gaps.

Gaps are also associated to the functional requirement "to mitigate core heat up" by large core heat capacity and relatively low power density and "to mitigate oxidation of graphite" by employing of high quality graphite relate to the inherent design features of HTGRs. There is a possibility that a new regulatory requirement is needed for mitigation of relevant mechanisms, or for prevention of the mechanisms by adequate design features of the reactor.

The functional requirements "to prevent contact of oxidation gases to coated fuel particles" and "to mitigate oxidation of coated fuel particles" are related to the fuel. For existing regulations that reflect Requirement 43 of SSR 2/1 (Rev. 1) [11] in this topic, it may be necessary to consider an expansion. It seems difficult to address these gaps by developing guidance or modifying the requirement to make it more general because the gaps are significant and HTGR specific. An example of how existing regulatory requirements on fuel could be expanded to address these issues can be formulated as follows: “fuel needs to be designed to take chemical attack in all plant states into account”.

The functional requirements “to mitigate air ingress” and “to control combustible gas concentration” by the reactor building are related to regulatory requirements on the design of the containment and confinement functions. Though the role of containment in the confinement function is discussed in Section A.1 of the Appendix, these functional requirements are different from requirements to confine fission products because they need to consider ingress of air into the RCS. It may be necessary to extend existing requirements on the containment/confinement function to address the gaps. For example, a new requirement such as “design features to limit the availability of air for possible ingress into the reactor core in the event of a break in the helium pressure boundary need to be provided as necessary” could be proposed.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Advanced Reactors Information System (ARIS): https://aris.iaea.org/sites/overview.html

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Small and Medium Power Reactors, Conference Proceedings, IAEA (1960).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Small and Medium Power Reactors, Conference Proceedings, IAEA (1970).

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Proceedings of the 6th INPRO Dialogue Forum: Global Nuclear Energy Sustainability: Licensing and Safety Issues for Small- and Medium-sized Reactors (SMRs), Vienna, Austria, 29 July – 2 August 2013.

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Small Modular Reactor (SMR) Regulators' Forum, 2014 – 2021: https://www.iaea.org/topics/small-modular-reactors/smr-regulators-forum

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, International Conference on Topical Issues in Nuclear Installation Safety: Safety Demonstration of Advanced Water Cooled Nuclear Power Plants, Vienna, Austria, 6–9 June 2017.

[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Side Events at the 63rd IAEA General Conference, Vienna, Austria, 16 – 20 September 2019.

[8] INTERNATIONAL ATOMIC ENERGY AGENCY, Climate Change and Nuclear Power 2020, IAEA, Vienna (2020).

[9] INTERNATIONAL ATOMIC ENERGY AGENCY, Advances in Small Modular Reactor Technology Developments, a Supplement to IAEA Advanced Reactors Information System (ARIS), IAEA Booklet, 2020 Edition, IAEA, Vienna (2020).

[10] INTERNATIONAL ATOMIC ENERGY AGENCY, Deployment Indicators for Small Modular Reactors, IAEA-TECDOC-1854, IAEA, Vienna (2018).

[11] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), IAEA, Vienna (2016).

[12] INTERNATIONAL ATOMIC ENERGY AGENCY, Applicability of Design Safety Requirements to Small Modular Reactors Technologies Intended for Near Term Deployment – Light Water Reactors – High Temperature Gas Cooled Reactors, IAEA-TECDOC-1936, IAEA, Vienna (2020).

[13] INTERNATIONAL ATOMIC ENERGY AGENCY, Proposal for a Technology-Neutral Safety Approach for New Reactor Designs, IAEA-TECDOC-1570, IAEA, Vienna (2007).

[14] SMALL MODULAR REACTOR REGULATORS' FORUM, Pilot project report: Considering the Application of a Graded Approach, Defence-in-Depth and Emergency Planning Zone Size for Small Modular Reactors, IAEA, Vienna (2018).

[15] INTERNATIONAL ATOMIC ENERGY AGENCY, Deterministic Safety Analysis for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-2 (Rev. 1), IAEA, Vienna (2019).

[16] INTERNATIONAL ATOMIC ENERGY AGENCY, Passive Shutdown Systems for Fast Neutron Reactors, IAEA Nuclear Energy Series No. NR-T-1.16, IAEA, Vienna (2020).

[17] BROOKHAVEN NATIONAL LABORATORY, Phenomena Important in Molten Salt Reactor Simulations, Rep. BNL-114869-2017-INRE, Upton, NY (2018).

[18] INTERNATIONAL ATOMIC ENERGY AGENCY, Unusual occurrences during LMFR operation, IAEA-TECDOC-1180, IAEA, Vienna (2000).

[19] ARGONNE NATIONAL LABORATORY, Research and Development Roadmaps for Liquid Metal Cooled Fast Reactors – Draft for Public Comment, Rep. ANL/ART-88 Rev. 02, Argonne, IL (2018).

[20] OECD NUCLEAR ENERGY AGENCY, Experimental Facilities for Sodium Fast Reactor Safety Studies, Task Group on Advanced Reactor Experimental Facilities (TAREF), NEA/CSNI/R(2010)12, OECD, Paris (2011).

[21] INTERNATIONAL ATOMIC ENERGY AGENCY, Nuclear Power Reactors in the World, 2019 Edition, Reference Data Series No. 2, IAEA, Vienna (2019).

[22] SUN, Q., PENG, W., YU, S., WANG, K., A review of HTGR graphite dust transport research, Nucl. Eng. Des. **360** (2020) 110477.

[23] BARTH, G., LECRIVAIN, G., JAYARAJU, S.T., HAMPEL, U., Particle deposition and resuspension in gas-cooled reactors—Activity overview of the two European research projects THINS and ARCHER, Nucl. Eng. Des. **290** (2015) 127–134.

[24] KO, H., SZLUFARSKA, I., MORGAN, D., Cs diffusion in SiC high-energy grain boundaries, J. Appl. Phys. **122** (2017) 105901.

[25] OAK RIDGE NATIONAL LABORATORY, Modeling Molten Salt Reactor Fission Product Removal with SCALE, Rep. ORNL/TM-2019/1418, Oak Ridge, TN (2020).

[26] INTERNATIONAL ATOMIC ENERGY AGENCY, Liquid Metal Coolants for Fast Reactors Cooled by Sodium, Lead and Lead-Bismuth Eutectic, IAEA Nuclear Energy Series No. NP-T-1.6, IAEA, Vienna (2012).

[27] EUROPEAN ATOMIC ENERGY COMMUNITY, FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANIZATION, INTERNATIONAL MARITIME ORGANIZATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS ENVIRONMENT PROGRAMME, WORLD HEALTH ORGANIZATION, Fundamental Safety Principles, IAEA Safety Standards Series No. SF-1, IAEA, Vienna (2006).

[28] IDAHO NATIONAL LABORATORY, Guidance for Developing Principal Design Criteria for Advanced (Non-Light Water) Reactors, Rep. INL/EXT-14-31179, Revision 1, Idaho Falls, ID (2014).

[29] UNITED STATES NUCLEAR REGULATORY COMMISSION, Guidance for Developing Principal Design Criteria for Non-Light Water Reactors, Regulatory Guide 1.232, Revision 0, US NRC, Washington, DC, (2018).

[30] INTERNATIONAL ATOMIC ENERGY AGENCY, Consideration on Performing Integrated Risk Informed Decision Making, IAEA-TECDOC-1909, IAEA, Vienna (2020).

[31] UNITED STATES NUCLEAR REGULATORY COMMISSION, Staff Requirements – SECY-98-144 – White Paper on Risk-Informed and Performance-Based Regulation, US NRC, Washington, DC, (1999).

[32] UNITED STATES NUCLEAR REGULATORY COMMISSION, Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing – Main Report, NUREG-1860, Vol. 1, US NRC, Washington, DC (2007).

[33] INTERNATIONAL ATOMIC ENERGY AGENCY, Establishing the Safety Infrastructure for a Nuclear Power Programme, IAEA Safety Standards Series No. SSG-16 (Rev. 1), IAEA, Vienna (2020).

[34] INTERNATIONAL ATOMIC ENERGY AGENCY, Practical Experience in the Regulations and Licensing of SMRs, TECDOC, IAEA, Vienna (in preparation).
[35] INTERNATIONAL ATOMIC ENERGY AGENCY, Considerations in the Development of Safety Requirements for Innovative Reactors: Application to Modular High Temperature Gas Cooled Reactors, IAEA-TECDOC-1366, IAEA, Vienna (2003).
[36] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment of Defence in Depth for Nuclear Power Plants, Safety Reports Series No. 46 (Rev. 1), IAEA, Vienna (in preparation).
[37] INTERNATIONAL ATOMIC ENERGY AGENCY, Considerations on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants, IAEA-TECDOC-1791, IAEA, Vienna (2016).
[38] AUTORITÉ DE SÛRETÉ NUCLÉAIRE, Report on the state of nuclear safety and radiation protection in France in 2019, ASN, Paris (2019).
[39] AUTORITÉ DE SÛRETÉ NUCLÉAIRE, Technical guidelines for the design and construction of the next generation of nuclear power plants with pressurized water reactors, Report of the Advisory Committee for Reactors and German experts, ASN, Paris (2000).
[40] INTERNATIONAL ATOMIC ENERGY AGENCY, Best Estimate Safety Analysis for Nuclear Power Plants: Uncertainty Evaluation, Safety Reports Series No. 52, IAEA, Vienna (2008).
[41] INTERNATIONAL ATOMIC ENERGY AGENCY, Power Reactor Information System (PRIS), Glossary of Terms in PRIS Reports: https://pris.iaea.org/PRIS/Glossary.aspx
[42] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment for Facilities and Activities, IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), IAEA, Vienna (2016).

ABBREVIATIONS

BWR	boiling water reactor
CCF	common cause failure
CFP	coated fuel particle
CSAU	code scaling, applicability and uncertainty
DBA	design basis accident
DEC	design extension condition
DiD	defence in depth
HTGR	high temperature gas cooled reactor
IAEA	International Atomic Energy Agency
IRIDM	integrated risk informed decision making
LFR	lead cooled fast reactor
LMFR	liquid metal fast reactor
MSR	molten salt reactor
NPP	nuclear power plant
PHWR	pressurized heavy water reactor
PIRT	phenomena identification and ranking table
PWR	pressurized water reactor
RCS	reactor coolant system
SFR	sodium cooled fast reactor
SMR	small and medium sized or modular reactor
SSCs	structures, systems, and components
TRISO	tristructural-isotropic (nuclear fuel)
UHS	ultimate heat sink
WCR	water cooled reactor

CONTRIBUTORS TO DRAFTING AND REVIEW

Alderson, N.	CANDU Owners Group, Canada
Arnold, R.	International Atomic Energy Agency
Calle Vives, P.	Office for Nuclear Regulation, United Kingdom
Cook, S.	Canadian Nuclear Safety Commission, Canada
Deknopper, K.	Tractebel ENGIE, Belgium
Flauw, Y.	Institut de Radioprotection et de Sûreté Nucléaire, France
Luis Hernandez, J.	International Atomic Energy Agency
Misak, J.	ÚJV Řež, a. s., Czech Republic
Montaner-Calatrava, T.	Office for Nuclear Regulation, United Kingdom
Ohashi, H.	Japan Atomic Energy Agency, Japan
Ranval, W.	Electricité de France, France
Reiman, L.	Consultant, Finland
Sinegribov, S.	Scientific and Engineering Centre for Nuclear and Radiation Safety, Russian Federation
Travis, B.	Nuclear Regulatory Commission, United States of America
Vesecky, R.	ČEZ, a. s., Czech Republic

Consultants Meetings

Vienna, Austria: 2–6 September 2019, 3–7 February 2020

IAEA
International Atomic Energy Agency

No. 26

ORDERING LOCALLY

IAEA priced publications may be purchased from the sources listed below or from major local booksellers.

Orders for unpriced publications should be made directly to the IAEA. The contact details are given at the end of this list.

NORTH AMERICA

Bernan / Rowman & Littlefield

15250 NBN Way, Blue Ridge Summit, PA 17214, USA
Telephone: +1 800 462 6420 • Fax: +1 800 338 4550
Email: orders@rowman.com • Web site: www.rowman.com/bernan

REST OF WORLD

Please contact your preferred local supplier, or our lead distributor:

Eurospan Group

Gray's Inn House
127 Clerkenwell Road
London EC1R 5DB
United Kingdom

Trade orders and enquiries:

Telephone: +44 (0)176 760 4972 • Fax: +44 (0)176 760 1640
Email: eurospan@turpin-distribution.com

Individual orders:

www.eurospanbookstore.com/iaea

For further information:

Telephone: +44 (0)207 240 0856 • Fax: +44 (0)207 379 0609
Email: info@eurospangroup.com • Web site: www.eurospangroup.com

Orders for both priced and unpriced publications may be addressed directly to:

Marketing and Sales Unit
International Atomic Energy Agency
Vienna International Centre, PO Box 100, 1400 Vienna, Austria
Telephone: +43 1 2600 22529 or 22530 • Fax: +43 1 26007 22529
Email: sales.publications@iaea.org • Web site: www.iaea.org/publications

22-03880E

International Atomic Energy Agency
Vienna